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Design approaches for heating reactors

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FOREWORD

The largest fraction of all energy consumed by society is in the form of heat at a temperature below 150°C, and the use of nuclear energy to provide this sort of energy has been examined for the past several years. Accordingly, the IAEA has been involved in related activities ever since the various technical approaches were first discussed in the late 1970s within the Nuclear Power Programme. A number of meetings, such as Technical Committee meetings and Advisory Group meetings have been held, and their results have usually been published in IAEA-TECDOCs, the last of in 1991.

Substantial progress has been made in the meantime with regard to experimental verifications and operating experience for integral types of dedicated nuclear heating reactors and for implementing commercial size demonstration plants. This became evident at an Advisory Group meeting which was held in Beijing in June 1994, and which is the subject of the present publication.

Nuclear heating reactors for supplying low temperature heat have been developed by relatively few countries, notably the Russian Federation and China. Progress in the development of reactors for supplying low temperature heat is currently of added significance since such facilities are also very well suited for seawater desalination. This is a field which is gaining worldwide importance and in which the IAEA has taken an active role in promoting goal-oriented co-operation between the interested parties. It is hoped that the present TECDOC will make a contribution to the understanding of one of the candidate technologies for nuclear desalination.

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1. BACKGROUND AND PURPOSE OF THE ADVISORY GROUP MEETING

Interest in using nuclear energy for district heating dates back to the 1960s when, for example in Sweden, the Ågesta reactor successfully supplied 55 MW(th) to a suburb of Stockholm for about ten years. On account of the then very low oil prices this type of nuclear application was not expanded. Interest revived after the first oil crisis in 1973–1974 in a number of countries and led to the investigation of different types of dedicated nuclear heating reactors as well as to studies on extracting heat from the low pressure stage of the turbine of a normal nuclear power plant (nuclear co-generation). A successful example of the latter is the REFUNA Project in Switzerland where up to 100 MW(th) are being extracted from the Beznau NPP in winter.

Although development of dedicated nuclear heating reactors (NHRs) has been halted in European countries such as Germany, Switzerland and the Czech Republic, the basic rationale for nuclear district heating is still valid. It rests primarily on the significant contribution that this application of nuclear power can make to reducing and limiting the detrimental environmental consequences associated with ever increasing use of fossil fuels, primarily coal. The halt in development in the European countries is due to a mix of unfavourable circumstances, mainly low prices for natural gas and oil, but also political decisions such as a lack of commitment to nuclear energy and maintenance of a given rate of domestic coal consumption in Germany. Such circumstances do not necessarily apply to the Russian Federation and China. As long as competitiveness with local alternatives, evaluated for locally prevailing conditions, including infrastructure aspects, support the nuclear option in general, a sound case can also be made for nuclear district heating.

However, even though a large number of countries had been investigating nuclear district heating with a great variety of concepts in the 1980s, presently only a few countries are making substantial efforts toward realization of such projects. The most active heating reactor development programmes are currently under way in China and in the Russian Federation. It was thus considered appropriate to hold an Advisory Group meeting to review the technical status of NHR concepts that are characterized by:

- Integral design with operating pressure between 1 and 2.5 MPa (10 and 25 bar),
- Light water cooling with primary side natural circulation, either in boiling or non-boiling mode.

The restriction to this family of reactor type is not to be seen as a negative judgment on other concepts of heating reactors. Instead, it was felt that for efficiency the meeting should focus on a single family of NHR reactors with comparable features.

For similar reasons it was decided not to discuss nuclear co-generation which is also a promising approach to providing district heat and which is already being practised in many countries.

Finally, detailed aspects of costing and evaluation of nuclear district heating economics are seen to vary so much from country to country, and even between different locations within one given country, that it would be difficult to come to conclusions that would be generally valid.

The Advisory Group meeting was held at the Institute of Nuclear Energy and Technology, Beijing and attended by 28 participants from six countries to review and discuss the status of dedicated nuclear heating reactors. Representatives of potential Chinese customers and the regulatory authority were also among those participating.

The meeting was conducted in two parts, the first three days dedicated to presentations and the others comprising workshop sessions. The workshop was conducted in two parallel sessions which were devoted to safety and thermo-hydraulic analysis and mechanical design issues, respectively.

The papers presented were arranged in accordance with the structure, corresponding to the topics being addressed:

- Status of NHR development;
- NHR safety objectives;
- Thermo-hydraulic design and supporting experiments;
- Mechanical design and water chemistry;
- Results of safety analysis;
- Coupling of NHRs to heating grids or desalination plants;
- Experience with the nuclear heating reactor NHR-5.

This report consists of two parts. The first part is the report of the Advisory Group meeting summarizing the status of technology of nuclear heating reactors, the findings of the workshop and conclusions. The second part comprises the full papers presented at the meeting.

2. REPORT OF THE ADVISORY GROUP MEETING

2.1. STATUS OF NHR DEVELOPMENT

General

Sensible heat is an important form of primary energy consumed by mankind to satisfy its various needs for energy. Low temperature heat for heating homes, factories, hospitals, etc. in turn accounts for a substantial fraction of that sensible heat, with fossil fuels being the dominant source. The world's population is still growing, and a large fraction of it is still striving for an improved standard of living. Therefore, further growth of primary energy consumption, also in the form of low temperature heat, will inevitably take place. It is not surprising that the use of nuclear energy for district heating has been under consideration for a long time and that in a number of countries, mostly in eastern Europe, nuclear generated heat is actually being used. The two principal advantages of nuclear over fossil fuels, particularly coal, are environmental and logistical. Nuclear power is practically pollution free in the widest sense of the word, and does not require substantial rail and road transportation facilities.

Yet, in spite of these principal advantages, the utilization of low temperature nuclear heat for space heating is marginal from a global standpoint. The key reason is that consumption at the end point of use is spread over a wide area, and the cost of transporting hot water, the dominant form of low temperature heat, is expensive. It is far more expensive than, for instance, the transmission of energetically high grade electricity. In addition, in comparison to electricity, energy for low temperature heat can be supplied in many other ways, e.g. by oil via truck, gas via pipelines or bottles, with direct or indirect (heat pump) electric heating and, of course, with old fashioned coal stoves. These alternatives, all closely related to each other in overall economic terms, provide a strongly competitive environment in which nuclear district heating has a difficult position, particularly at present when, on the one hand, fossil fuels are inexpensive and, on the other hand, a rapid return on capital investment is also required in the field of supplying energy in its various forms.

However, whilst the factors mentioned previously have to be addressed wherever nuclear district heating is being considered, their point by point evaluation in far greater detail than in this survey can give very different results for different regions. In other words, it is not possible to come to general conclusions that are universally valid for all regions of the world. Each case has to be looked at individually, recognizing that other factors than the very important one of economics, e.g. national policies, can play an important role when assessing the prospects of nuclear district heating.

In view of its environmental advantage such district heating is desirable, and it is also economically feasible if one looks beyond the immediate future. Currently, countries like China and the Russian Federation, although very different from each other, have good opportunities, again for different reasons, to implement and expand district heating with reactors specifically designed for this purpose.

The situation in China

In order to resolve the future energy and environmental problems, China has given great attention to R&D for a NHR. This work has been conducted as one of the national key projects since the 1980s. A 5 MW(th) NHR (NHR-5) was completed by INET in 1989 and has been operated successfully for space heating since then. In the meantime, a commercially sized NHR with an output of 200 MW(th) (NHR-200) was developed by INET. A number of experiments have been carried out on the NHR-5 to study the safety features and comprehensive applications of the NHR.

In 1991, it was decided to construct an NHR-200 demonstration plant at Daqing city in northeastern China. The siting report, the environmental impact assessment report, and the feasibility study of the project were approved by the respective authorities in 1993. The first NHR-200 is expected to be put into operation in 2000. The NHR, as a new promising energy system with a number of advanced and innovative features, can achieve the specified safety goal and economic viability. At present, the urgent task is to implement the NHR-200 demonstration plant project as soon as possible.

The situation in the Russian Federation

Quite real perspectives for nuclear district heating exist in the Russian Federation according to the survey that was reported. NHR technology has already been developed in this country, and substantial practical experience has been gained in designing, licensing, equipment manufacturing, transportation and erection of AST-500 type nuclear district heating plants (NDHPs).

In spite of an almost three year delay in the construction of a large pilot NDHP (2 × 500 MW(th)) near city of Voronezh, permission was given recently by the local authorities to resume plant construction. This decision was preceded by a comprehensive review of the environmental impact assessment report and of the basic design documentation for the NDHP by an ad hoc public ecological commission.

A feasibility study is being carried out for the Khabarovsk NDHP (in the far east of the Russian Federation) with two advanced AST-500M NHRs. Deployment of NHRs is also under consideration for some other regions, particularly small and medium power nuclear co-generation plants on the basis of AST type reactors. Shortage of fossil fuels and insufficient capacity of existing district heating systems are the main motivating factors. However, a distrustful attitude of the general public is still negatively influencing decisions regarding the utilization of nuclear energy, especially for nuclear heating.

2.2. NHR SAFETY OBJECTIVES

Basic trends

The trends in safety objectives for district heating plants are affected by two influences. On the one hand, they follow in general the trends for large nuclear power plants where there is a strong move to include consideration of severe accidents in the licensing procedures. On the other hand, whilst it is true that the specific characteristics of district heating reactors (size, heat capacity, temperatures and pressures) are favourable for implementing a high degree of safety, it is argued that the latter is an absolute necessity because of the vicinity of the reactor to the population centers to be provided with heat; in practice, it must be ensured that there will be no releases of radioactivity.

Therefore, it must be investigated whether any possible accident sequences could lead to an unacceptable release. Such extended sequences have to be designed out (prevention), or additional mitigating features have to be provided. From the NHR concepts known, it can be expected that this challenge can be met, since the features of the NHR (low pressure, high specific water inventory) ease the technical task of assuring safety.

Chinese objectives

The safety objectives and design criteria for the NHR-200 demonstration plant to be built in northern China has been drawn up and reviewed by a team organized by China's National Nuclear Safety Administration (NNSA).

The major differences in criteria compared to the ones for conventional nuclear power plants are as follows. An operation condition of category V is added, which is a design basis accident (DBA) followed by an additional failure. The dose limit for individuals should then be smaller than 5 mSv as required by China's NNSA. Correspondingly, more rigorous requirements in terms of fuel element damage, DNBR and fuel temperature limits are specified. The reactor core has to be covered by coolant in all loss of coolant accident (LOCA) cases. A guard vessel plus a concrete structure is recognized as providing the equivalent function of a conventional containment. The diesel generator system, the component cooling water system and the HVAC system are all classified as non-safety systems.

Finally, a 250 m non-residential area and a 2 km area of restricted development surrounding a site are required.

2.3. THERMO-HYDRAULIC DESIGN AND SUPPORTING EXPERIMENTS

Phenomena to be considered

In the course of the Russian AST-500 type design development, a large scope of experimental investigations on the reactor plant thermo-hydraulics has been carried out to select and verify the design decisions and parameters, and also to validate safety. A complex of test facilities was constructed, including electrically heated models of the reactor with guard vessel simulators of various scales, for in-depth study of the reactor thermo-hydraulics both under normal and emergency conditions.

The peculiarities of the thermo-hydraulics of the AST-500 in normal operation and under emergency conditions are associated with the natural circulation of the coolant, reduced plant parameters and with the specific features of the integral reactor, such as built in steam gas pressurizer, in reactor heat exchangers for emergency residual heat removal, and guard vessels. Proceeding from these factors, the main attention during the experimental work was paid to studying the following effects and phenomena:

- natural circulation stability;
- ultimate values for core heat loads;
- thermo-hydraulics of the built in steam gas pressurizer;
- behaviour of non-condensable gases (gas transfer, gas distribution, gases dissolved in water);
- natural circulation modes under loss of coolant conditions;
- operation of emergency heat removal heat exchangers with steam condensation from a steam gas mixture;
- the reactor guard vessel system behaviour at LOCAs;
- passive safety systems initiation and functioning, and other items.

The calculation models and computer codes for the analysis of AST reactor plant thermo-hydraulics under normal operating and emergency conditions have been developed and validated experimentally. The verification of the codes confirmed the reliability of the calculated prediction of the AST reactor-hydraulics.

A set of calculations and experimental investigations carried out on the thermo-hydraulics of the AST-500 has allowed identification of two possible modes of operation: boiling and non-boiling. The non-boiling mode of operation was selected for the AST-500. The primary circuit is pressurized by a steam gas pressurizer using a helium–hydrogen mixture. The value of average primary coolant subcooling was chosen on the basis of the requirements of providing stable natural circulation and excluding the accumulation of radiolytic gases in the reactor. A special feature of the design is that the natural circulation tolerates significant local variations of fuel assembly power (about a factor of 2.5) thanks to the self-shaping of the coolant flow rate.

Two phase flow instabilities

It is well known that low pressure natural circulation systems may exhibit flow instabilities if not properly designed. This problem was addressed, with particular reference to the NHR-200 demonstration plant for Daqing city. As a lead in, it was stated that the key difference, compared to the 5 MW(th) NHR, NHR-5, is that the grid supply temperature had to be increased to 130°C at the request of the customer. As a consequence, the core outlet temperature has been increased to 210 instead of 193°C and correspondingly the pressure of the primary system has increased to 2.5 MPa. Although a higher power density than in the NHR-5 was chosen for the NHR-200, suitable safety margins are still maintained in normal operation and under accident conditions due to the excellent inherent safety characteristics and the utilization of passive safety systems.

Regarding the investigations of hydrodynamic stability of two phase flows in a low pressure natural circulation system, the following conclusions were drawn:

- There can be low quality instabilities in natural circulation systems as demonstrated by experiments. The test results obtained agree with computer code RETRAN-02 and NUFREQ analyses.

- It was found that there is a subcooled boiling instability in natural circulation systems. This means that even if the main stream of the exit coolant is subcooled, if there is subcooled boiling in the heated region a subcooled boiling instability could take place under certain conditions. In light of this point, for reactors operated in a PWR mode, or for swimming pool reactors in natural circulation, avoiding subcooled boiling instabilities should be a design criterion.

- Steam quality is a dominant factor for the hydrodynamic instability of natural circulation systems operated in a low quality region.

- For a natural circulation system operated under very low pressure, such as atmosphere, there are many different types of instabilities which could take place, even if the bulk of the coolant is subcooled.

To get a deeper insight into this situation, theoretical studies of the first kind of density wave instability (DWI) in natural circulation systems have been performed at INET. Time domain and frequency domain methods were used to analyse DWIs and to predict the instability threshold and to investigate parameter effects on the instability threshold, as well as different types of instabilities. Finally, based on the energy principle, a dimensionless criteria can be used for predicting the first kind of DWI in a low pressure natural circulation system.

2.4. NHR MECHANICAL DESIGN AND WATER CHEMISTRY

Basic considerations

Heating reactors must be tailored to the specific demands of heating grids, and in order to minimize the heat generation cost the design has above all to aim for low specific capital cost. In the currently dormant SIEMANS NHR-200 design, one measure to gain an optimized design is incorporation of the control rod drives (CRDs) in the core. This feature leads to a series of advantages. The height of the RPV and containment will not be dominated by the drive and the control rod stroke. A modular core cell design, consisting of cruciform control rod guide structures surrounded by four fuel assemblies, simplifies refueling since no RPV internal components need to be removed prior to fuel handling. Since the reactor core and all the CRDs are supported by the same support structure, optimal behaviour even under external events is expected. The advantages of the design are an inhomogeneous water distribution in the core, and a limitation to about 6 m for the core height plus control rod stroke. Accounting for mechanical overlapping, the maximum active core height is expected to be less than 2.7 m.

As with electricity producing nuclear power plants, in-service inspection, maintenance and repair, and decommissioning aspects have to be taken into account from the very beginning of the design. Good access to all connections and welds in the primary circuit are mandatory. In order to achieve high quality at low cost the design should aim for low stress levels in the components, and for clear shapes of the individual parts.

Features of the AST-500 designs

The AST-500 reactor plant, being the first in a family of integral PWRs developed in the Russian Federation by OKBM, represents a new generation reactor plant with passive safety features. It is mastered in production and is used as a part of the Voronezh NDHP that is currently under construction. The high level of safety of this type of reactor plant is recognized by the national review and regulatory bodies and was confirmed by an international review team (IAEA pre-OSART mission).

The basic engineering decisions used in the AST-500 were then borrowed and further developed into a series of AST type advanced NHRs, the AST-500M and the AST-600. The following principal objectives have been adopted and realized for these advanced designs:

- uprating of reactor power within the same RPV dimensions and increase of heat output up to 600 MW(th);

- extension of the service life of the RPV and the main equipment (up to 60 years for the RPV);

- fuel lifetime extension up to 8-10 years;

- reduction of mechanical and electrical equipment in safety systems;

- enhancement of plant immunity to common cause failures and personnel errors by wide usage of self actuated devices;

- standardization of the engineering decisions for all AST type NHRs to be built in series.

Features of the Chinese NHR-200

Whilst the Russian design uses conventional control rod drives mounted on top of the RPV, vessel internal hydraulic drives are used by the Chinese developers, much in the same way as for the German concept. In the integrated design of the NHR-200 for Daqing, an in-vessel spent fuel storage and an internal hydraulic control rod drive system are adopted. Around and close to the active core is the area for spent fuel storage. This leads inter alia to a higher average of burn up as some neutrons are supplied by fission in the spent fuel. The new type of hydraulic control rod drive system can be properly arranged inside the reactor pressure vessel. The pressure vessel is designed to meet LBB (leak-before-break) requirements. This precludes brittle fracture of the pressure vessel, and its safety is thus greatly improved.

Crack propagation

The safety properties of the pressure vessel for NHR-200 were analysed showing the large differences in crack propagation characteristics between a 15 MPa (150 bar) PWR vessel and the 2.5 MPa (25 bar) vessel of the NHR; the crack propagation rate in the NHR-200 vessel is much lower than that for a conventional PWR.

The RPV of the NHR-200 is designed to meet the leak-before-break (LBB) criterion. A brittle fracture could thus be excluded, and only short through-wall cracks could happen, but with a very low probability. Based on these analyses, the in-service inspection requirements should be different from those for PWRs or BWRs and can be simplified in some ways.

Water chemistry

Experience in water chemistry for PWRs, including Russian PWRs, and for BWRs can be used to establish water chemistry guidelines for NHRs, as the operating conditions of integral heating reactors extend into the regimes of either of these reactors. Accordingly, water chemistry guidelines should be defined together with the mechanical design and the material selection phase. Chemical control parameters should also be specified and appropriate instrumentation for continuous measurements and record keeping must be provided.

During operation, in-service inspection of components should be reviewed with respect to applied water chemistry, especially taking into account activity levels (general corrosion) and cracking incidents. Conductivity measurements should be included as a control parameter.

2.5. RESULTS OF SAFETY ANALYSES

AST-500

The principal decisions adopted for the AST-500 design, such as utilizing an integral PWR with natural circulation of the primary coolant, emergency heat removal systems operating on a passive principle, use of a guard vessel, reduced working parameters and a relatively low core power density provide a high level of reactor plant safety.

For DBAs, the safety criteria require that with some margin the plant can be left for at least 3 days without power supply and intervention by plant personnel. This is achieved through the use of passive devices and systems that are provided in the design. Long term core cooling is also provided for beyond DBAs with various postulated failures in the safety systems. Fuel overheating and loss of cladding integrity are therefore seen as highly improbable. The radiological consequences of such accidents are considerably lower than the limits specified by the regulatory codes, and they correspond to the respective design requirements.

Considerable time margins exist before core overheating and melting could possibly occur, and there is a wide range of possibilities for accident control to prevent severe core damage. The evaluation of the frequency of postulated accident scenarios resulting in severe core damage has resulted in values less than 1E-8 per reactor-year. The design decisions adopted for providing the safety of AST-500, and its in-depth protection after beyond DBAs allow categorization of the AST-500 reactor plant as an enhanced safety system.

NHR-200

In the safety assessment of the Chinese NHR-200 five operating conditions or categories are considered, and for each a specific dose limitation has been set. For category I (normal operation) the maximum allowable dose is 0.1 mSv/a; for category II (anticipated operation events) the dose limit is 0.02 mSv/event; for category III (rare accidents) 1.0 mSv/event; for category IV (limiting accidents) 5.0 mSv/event; and for category V (additional accident category) the limit is 5.0 mSv/event. The analytical methods used for assessing the safety correspond to those used in other countries.

The results for the five basic accident categories show that there is no fuel element damage in categories I, II and III. This is also the case for the majority of category IV and beyond DBAs as the core is always covered with water. In summary, there is no large radioactivity release in any classified accident.

2.6. NHR COUPLING TO HEATING GRIDS OR DESALINATION PLANT

Features of the Daqing heating grid

The key parameters of the district heating grid for the Daqing Nuclear Heating Plant are as follows:

- The NHR-200 will be installed as the main heat resource.
- The distance between the NHP and the boundary of Daqing city is about 2.5 km.
- The grid system consists of four loops.
- The heating period is 180 days per year, including 110 days with peak load fossil fired boilers; there are 12 boilers in the grid.
- Experiments with nuclear heating (NHR-5) have shown that load control by moving the control rods of the reactor is possible, but does not seem to be necessary. Because of the great heat inertia of the system, and on the basis of regulation by the temperature coefficient, the available load following characteristics are good enough (inherent load following).

Coupling to desalination plants

The AST-500 NHR has become a reference reactor plant for a whole series of integral enhanced safety reactors that have been developed recently at OKBM. The basic principles and engineering decisions realized in the reactor plant allow it to be used effectively as a heat source for an evaporating type desalination facility, which is based on the well proven DOUGTPA type apparatus with horizontal tube film evaporators. The rated output of this facility amounts to 220 000 m^3/day of desalinated water, with the potential for a 15% increase in output. The modes of AST-500 operation in combination with the desalination facility are similar to those for operating a heating grid.

The high level of radiological safety intrinsic to the AST-500 reactor makes it feasible to site it close to water sources, to freshwater consumers, and in the immediate proximity of the desalination facility. It is thus possible to form an integrated nuclear desalination complex.

The positive practical experience of the Russian Federation in creating and operating the nuclear seawater desalination plant based on the BN-350 reactor, as well as with the desalination apparatus of the DOUGTPA-type, together with the proven technology of the AST-500 NHR, substantially facilitate licensing and implementation of AST-based desalination complexes. Because of their excellent safety and economic characteristics they can be considered to be good prospects for deployment in many regions of the world suffering from a shortage of potable water.

2.7. EXPERIENCE WITH THE NHR NHR-5

The participants of the Advisory Group meeting were given the opportunity to visit the five MW(th) NHR developed and operated by INET for the specific purpose of studying the special characteristics of integral, low pressure reactors with natural circulation.

The NHR-5 has been operated for four winter seasons, and a number of experiments have been carried out since 1989 on its operational behaviour and safety features. The experiments related to such topics as self regulation with load changes of up to 60%, self stability upon an arbitrary insertion of a reactivity of 2 mk, simulated ATWS experiments initiated by loss of the main heat sink, and RHRS experiments involving the interruption of natural circulation in the primary circuit. All of the results obtained indicate that the NHR-5 has the expected inherent and passive safety features. In particular, changes in reactor pressure were always relatively small, e.g., less than 0.1 MPa (1 bar) when the external load dropped by 60%, a relatively severe disturbance.

The measurements of water chemistry have shown that nitrogen as cover gas is feasible for NHRs. Under the selected neutral water chemistry practice without soluble boron, oxygen is controlled by the chemical additive C_2H_4. The operating experience has also shown that the oxygen concentration can easily be kept below the specified 50 ppm.

3. TOPICS AND RESULTS OF THE WORKING GROUPS

Following the presentation of the papers, the participants broke into two groups dealing with safety and thermo-hydraulic analyses, and mechanical design issues, respectively. The topics for each group were, however, jointly defined in full plenum in order to reflect the questions of all participants. In addition, aspects of water chemistry for NHR conditions, and questions related to the operation of district heating grids were discussed in two small ad hoc groups. The findings of the workshop can be summarized as follows:

3.1. SAFETY APPROACHES AND RELATED THERMO-HYDRAULIC ANALYSES

The topics of the safety approaches and related thermohydraulic analyses were:

- Categorization of operational and accident conditions, with reference to the categorization proposed for the Chinese NHR-200:

  - Which accidents belong in category IV and which in V (beyond design base)?
  - Realistic or conservative assumptions for evaluation of category V accidents?
  - Qualifications of components for category V.

- Containment aspects:

  - Is a conventional containment necessary in addition to a guard vessel?

- Thermo-hydraulic aspects:

  - Thermo-hydraulic conditions in the presence of non-condensable gases.
  - Validation of codes for such conditions.
  - Deflagration conditions.

- Reactivity effects:

  - Electrical power supply aspects.

Categorization of operating and accident conditions

The main questions were related to what assumptions and design criteria have to be used for category V. This category is intended to comprise certain beyond DBAs which are expected to also play a role for NHRs, in particular since these reactors will be located relatively close to population centers. There is some analogy to the severe accident discussion for future large nuclear power plants for which safety enhancements are also being sought.

There is a basic commonality in safety objectives. All the NHR designs considered by the Advisory Group Meeting strive for a minimal off-site radiological impact, both during normal operation and after accidents, as the main safety objective. To achieve this goal, the NHR designs rely on enhanced capabilities to prevent and to manage anticipated emergency conditions through reliable prevention of reactor core uncovering, and thus fuel overheating. This is accomplished by virtue of the inherent safety features of NHRs, and through utilization of passive safety systems. As to the categorization, it was observed that the Chinese approach is a useful one since it is practised in many countries. However, differences in detail preclude generalization. Nevertheless, there was agreement that the distinction between DBAs and beyond DBAs should be retained and that the guidelines for treating the latter should not be as conservative as for the former.

A general approach which also recognizes the aspect of close location to population centers may be summarized as follows. DBAs are selected and defined in a comparable way to those used for future advanced nuclear power plants. The technical safety objective for the NHR requires, specifically, that there should be no fuel failure due to fuel overheating under DBA assumptions; and there should be no significant radiation consequences associated with DBAs.

Additional and extended accident conditions are therefore considered in the NHR designs. Their selection is based on technical reasoning that is, again, similar to that underlying the selection of severe accidents for future power reactors. However, the selection for the NHR recognizes, and reflects accordingly, the advantageous safety properties and features of the NHR. On account of these it can be shown that equivalent severe accident sequences also do not also lead to unacceptable fuel damage. It is because of these favorable safety properties that off site protection measures are not required for the NHR, and that additional freedom is gained in siting without increasing the risk to the general public.

Containment aspects

All NHR designs presented at the Advisory Group meeting employ a guard vessel which encloses the primary system. This solution effectively fulfills two key safety functions: prevention of core uncovering and confinement of potential radioactivity releases. Due to this innovative approach the source term is very low, and it may be shown to be technically possible to dispense with the traditional containment adopted for large NPPs as far as the radioactivity retention function is concerned. The other function, protection against external events, may be achieved in different ways. Either, as in one of the Russian approaches, by providing a strong concrete shell, similar to a traditional containment, or by a reinforced concrete reactor building which is designed to the specific site condition. Regarding the first option of providing a strong concrete containment shell, it was pointed out by some participants that this approach could be very beneficial from a public acceptance point of view.

Thermo-hydraulic aspects of safety

It was stressed by the Russian participants that western codes, particularly for describing long term LOCA sequences, may have to be slightly adapted to take account of the specifics of the NHR designs. One of these is the high amount of non-condensable gases in the primary circuit during normal operation. In the course of a LOCA they will reduce the heat transfer coefficient at the primary heat exchangers that serve as the heat sink. Whilst there seem to be no problems for DBAs, and most of beyond DBAs, a validation is needed to show adequate safety margins also for certain long-term conditions beyond the design basis.

Taking into account the complex character of thermo-hydraulic phenomena under LOCA conditions in NHRs (due to an unusually large amount of non-condensable gases in the primary system, as mentioned before), the Advisory Group meeting recommends that INET and OKBM carry out a joint benchmark on this topic, using their respective computer codes for an independent assessment of a jointly defined configuration and scenario.

The phenomena of accumulation of radiolytic gases in the reactor and the guard vessel during long term LOCAs (one day or more) were pointed out, and attention was drawn to the need of a special analysis of the gas mixture composition and the possibility of its deflagration.

Reactivity effects

In NHRs not only the heat removal problems are more relaxed compared to NPPs but also the problems caused by reactivity effects. There are several reasons for this:

- Because of the smaller burnup the total reactivity swing is smaller, and with the use of burnable poison chemical shim is not required. Therefore no boron dilution accidents are possible.

- Because of the large primary coolant water inventory the cooldown of the reactor is rather slow so that the associated reactivity additions have no steep ramps.

- For the NHRs considered, the reactivity insertion through any erroneous operational or accidental control rod movement is small.

Therefore, no reactivity insertions leading to unacceptable reactor conditions and to either DBAs or beyond DBAs could be identified.

Safety requirement for electrical power supply aspects

Due to the fact that NHRs are designed for passive heat removal, there are no large power consumers such as emergency charging pumps. The power required for instrumentation and control and for actuating certain valves can be supplied by a safety grade battery system with a capacity of the order of one day. Therefore, there is no need for safety grade diesel generators. Nevertheless, non-safety grade diesels are provided for backup.

However, it was pointed out by one participant that the licensing authorities may take the position that in view of the specific conditions of the site under consideration, the proposed approach may not be acceptable. Therefore at least one safety grade diesel should be provided. He did acknowledge, though, that its functional specifications need not be as stringent as those for normal NPPs.

3.2. REACTOR MECHANICAL DESIGN ISSUES

The topics addressed during the workshop on mechanical design of NHRs were:

- core arrangement and fuel channels,
- control rod drive mechanism,
- in-core instrumentation,
- refueling,
- pressure vessel and heat exchangers,
- in-service inspection, maintenance and repair.

Core arrangement and fuel channels

Whilst the AST has a PWR type core arrangement based on WWERs, the German NHR design is based on an advanced BWR core arrangement. INET has adopted PWR fuel assemblies and BWR control assemblies. This solution has a future potential for two phase flow in the primary circuit. The Siemens NHR core cell is a modular one and leads to advantages during refueling. All three designs are tailored to the specific NHR conditions. That is, they all show fuel channels. In two of the designs, those of INET and Siemens, it is planned that these channels not be affixed to the fuel assemblies. They remain in the core over the entire lifetime of the reactor. The expected irradiation induced growth is the limiting factor to assure a bowing free control rod path. Channel management and dedicated heat treatment (texture factor) have been identified as measures for limiting bowing.

Control rod drive mechanism (CRD)

There are advantages and disadvantages of hydraulic CRDs and their location relative to the core. Whilst integrating the drives into the core (Siemens NHR) leads to a very compact RPV, a location above the core requires more space. Furthermore, refueling becomes more complicated. A location on top of the RPV also leads to the question of why not to adopt well proven electrically driven CRDs, as applied for the AST. A detailed comparison between the two options is being planned.

In-core instrumentation

All the three reactor concepts presented at the Advisory Group meeting are designed with in-core instrumentation, but the quantities of detectors differ. The number of detectors may be reduced for future plants on the basis of increasing experience. Such later plants can perhaps eliminate in-core neutron flux detectors altogether.

Refueling

All concepts are designed for removal of fuel assemblies from the RPV to a spent fuel storage. The Siemens NHR and the INET design have interim in vessel fuel storage. If long term integrity of the fuel can be assured, in vessel storage of all spent fuel over the reactor lifetime has advantages since expensive external spent fuel structures (pools, vessels, handling equipment) can be minimized or avoided. However, the long term behaviour of the cladding of the spent fuel at elevated water temperatures has yet to be fully verified for this purpose.

Pressure vessel and heat exchangers

With regard to the reactor pressure vessel, its size and in-service inspection (ISI) are key issues. Since the pressure vessels for the three designs presented show different sizes for similar unit power, it was concluded that as far as the economics of the NHR are concerned, a reduction in vessel length is more advantageous than a reduction in diameter. Regarding ISI, see the pertinent paragraph below.

Different types of heat exchangers are used in the three designs. Whilst the AST has straight tubes, INET has a two header U tube heat exchanger. The SIEMENS concept is equipped with a U tube unit that has only one header. The mechanical features of the HEXs are comparable. With regard to natural circulation in the intermediate circuit, the flow direction in the two header U tube design can be affected by the temperatures in the legs of the loop.

ISI, maintenance and repair

All participants took the position that access for performing the related tasks should be maximized. This objective is thus reflected in the designs. However, consensus on certain details of the inspection requirements could not be reached because of differences in the approach taken by the designers and because of insufficient time during the workshop. It is important to consider the specific operating conditions of the NHR when deciding which of the inspection requirements established and practiced for NPPs in general will be applied to NHRs. An example would be the demonstration of leak-before-break as shown by the Chinese designers for their RPV.

3.3. WATER CHEMISTRY

Concerning the water chemistry, the main issues are the advantage/disadvantage which the lower operating temperature of NHRs provides with respect to present power reactors, and water chemistry related corrosion questions. It was found that water chemistry impacts upon operational safety in two ways. On the one hand, the integrity of pressure boundary materials may be affected, and activity transport and out of core radiation fields can impede inspection and maintenance, on the other hand. Therefore,

- Water chemistry should be established with regard to plant design and materials selection.
- The plant design should provide adequate sampling and/or monitoring possibilities to assure quick identification of normal parameters.
- Detailed chemistry procedures with intervention levels should be established.
- Review and surveillance of procedures should be implemented.

4. CONCLUSIONS AND RECOMMENDATIONS

4.1. GENERAL

The presentations and discussions have shown that substantial progress has been made during the past five years in the field of the NHRs. These are technically characterized by a low pressure, integrated primary system operating in natural circulation, which is housed in a guard vessel that hermetically encloses the primary system. The guard vessel fulfils the dual function of eliminating the possibility of a core being uncovered after a loss of coolant accident, and that of containing any radioactivity that may be released from the primary system in the course of an accident.

Prominent examples of this type of NHR are the Chinese NHR-200 and the different variants of the Russian AST-500 family. In China the basic design has started for a demonstration plant at Daqing, with the start of construction scheduled for 1997. In the Russian Federation, construction of the Voronezh AST-500 is expected to be resumed as soon as the current financial and public acceptance difficulties have been overcome. Additional plants are being planned for Khabarovsk and for Vladivostok. In China more than ten large cities are interested in a NHR, but they want to see a full size demonstration before making firm commitments for construction. Therefore, there is an urgent need to implement the NHR demonstration plant as soon as possible.

4.2. COMMONALITIES IN DESIGN AND SAFETY APPROACHES

The Advisory Group meeting confirmed the high degree of common positions in design approaches. This is exemplified by the above mentioned characteristic features of integral design, natural circulation on the primary side, and the utilization of a guard vessel to provide the standard containment function. Differences do exist, however, e.g. in the area of detailed design parameters, type of reactor control and system configuration. Building layout, and a number of other important technical areas also show different solutions. Despite these differences in detail, a very important commonality beyond selecting an integrated design can be identified as follows.

For technical and economic reasons it is necessary to locate NHRs close to population centres, much closer than is typical for electricity generating NPPs. There is a strong consensus that off site emergency measures for the ultimate protection of the general public in case of severe accidents are not acceptable for assuring adequate safety. Therefore, the designs discussed emphasize the prevention of overheating of the fuel which might entail a large release of fission products into the primary system of the reactor, and ultimately into the environment. In an analogy to the evolving safety objectives for future NPPs, additional failure postulates are under consideration for designing NHRs in a way that it can be technically demonstrated that an unacceptably large release of fission products, scaled to the proximity of a population center, is not possible.

The prime means for accomplishing this objective is to assure by design measures that the probability of an uncovering of the core is not realistically conceivable. The key feature for supporting this position is the provision of the guard vessel which assures that the loss of primary coolant in case of an assumed leak is terminated before the core becomes uncovered and decay heat removal is jeopardized. In this aspect there is no fundamental difference between the approaches discussed. In the view of the designers, the NHRs fulfill all top level safety objectives for existing and future reactors.

4.3. SUGGESTIONS FOR FUTURE ACTIVITIES

It would be desirable to come to a more detailed understanding of specific accident assumptions for NHRs. Specifically, how to identify and select in detail DBA conditions and those that go beyond, but have to be considered in the design. The same applies to how to deal with them and in what manner at the design level. As NHRs are seen to be evolving in line with future NPPs, compatibility has also to be established with the evolution of the general safety philosophy for these reactors. The suitable approach for such an activity still has to be established.

Another suggestion made was to do a benchmark analysis on the effects of non-condensable gases on the long term heat removal after a loss of coolant accident under beyond design basis assumptions.

PAPERS PRESENTED AT THE ADVISORY GROUP MEETING

DEDICATED LOW TEMPERATURE NUCLEAR DISTRICT HEATING PLANTS: RATIONALE AND PROSPECTS

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Abstract

Space heating accounts for a substantial fraction of the end-energy consumption in a large number of industrialized countries. Accordingly, efforts have been under way since many years to utilize nuclear energy as a source for district heating. The paper describes the key technical and institutional issues affecting the implementation of such technology. It is argued that the basic case for nuclear district heating is sound but that its introduction merits and drawbacks strongly depend on local circumstances.

1. INTRODUCTION

It is well known that the largest individual fraction of all energy consumed by society is in the form of sensible heat. Even though there are differences between different countries, mainly due to climatic circumstances, above statement is generally valid. On a world average, about 50% of all primary energy is used for heat, about 20% for transportation and the remainder of 30% for generating electricity. To a very large extent fossil fuels are used for meeting these needs for energy. In absolute terms, the world primary energy consumption in 1992 was just below 12 TWa/a. Even in more restrictive scenarios, the total consumption is expected to double until about the year 2020.

In contrast to the general thinking in, say, the early seventies, there is today little doubt that there will be no shortage of reasonably priced fossil fuel resources for the foreseeable future. But also in contrast to earlier thinking, concerns over the environmental consequences of such a massive use of fossil fuels have increased substantially and, perhaps more important so, are in principle shared by both the general public and the decision makers in practically all countries of the world. Whilst there is still scientific controversy about how serious the greenhouse effect related to burning of fossil fuels actually is, and whilst there is still political controversy about how to curb the use of such fuels how fast, there is, again, little dispute about the position that their use has to be limited. The key words in this context are the big and important conferences on environmental issues held at Toronto and Rio de Janeiro, respectively.

2. THE RECORD OF NUCLEAR POWER AND ITS CURRENT ISSUES

And this is where "we nuclear people" think that we have a good answer. The majority of energy experts the world-over, including those with a critical position towards nuclear energy, seem to agree that there is currently no alternative for a large scale substitution of fossil fuels at acceptable cost, with the latter two constraints being of decisive importance.

Nuclear energy, however, is not without controversy, as is well known. It would need another paper to review this complex issue because there are many facets to it which would need detailed individual evaluation. Here only a few remarks, almost as personal observations, will be made to fence-in, so to speak, our playing field.

The first one is that, by and large, nuclear does have an excellent record. According to the IAEA Power reactor Information System (PRIS), by the end of 1993 there were 430 reactor units connected to the grid with a total capacity of 337.8 MW(e). The accumulated experience amounts to about 6,900 reactor-years. During 1993, 9 reactors (8988 MW (e)) were connected to the grid in Canada, China, France, Japan, Russia and in the USA, and another 8 reactors with 5627 MW(e) are expected to achieve grid connection in 1994. Today, 17% of the world's electricity consumption is supplied by nuclear power. In absolute terms this is more than the world consumed in total at the time when the first nuclear reactor produced electricity.

As Fig 1 shows, we have accumulated the 6900 reactor-years of operation with, essentially, only two very severe accidents, of which only one resulted in very severe consequences to the environment. Fortunately, this latter one can, with some qualification, be considered as a singularity which is not representative for the vast majority of currently operating reactors. In other words, staying with the accepted design approaches and safety principles, a point addressed in another paper of this conference, to which most operating reactors have been designed gives confidence that the future record will be at least as good as the present one. At least as good implies better than today, as the goals for the future strive for even better performance, also in safety.

This then leads to the second point which is only now becoming apparent in a way that is still somewhat diffuse. Much of the criticism about nuclear energy seems to be shifting away from the concern about immediate personal injury associated with severe accidents to questions of collective "damage", mostly in terms of affecting the environment, rather than health injuries, to questions regarding long-term waste disposal, or rather, the predictability of its consequences, and, finally, to questions of misuse of civilian nuclear technology, namely proliferation. In either case, the (relative) position does not seem to be hopeless for nuclear, in fact, it looks very promising. In regard to environmental concerns and to long-term waste disposal, a comparison with fossil fuels would seem to give nuclear a decisive advantage. Again, this paper cannot deal with the proof, the statement has to remain as such. This would also apply to the question of non-proliferation. Diverting plutonium from the fuel cycle of commerical reactors and manufacturing it into crude, low-yield and rather unreliable explosive devices would just seem to be too cumbersome.

The third, and in many ways most important point is as follows. Although the above may give the impression that nuclear "has little to fear" because its benefits seem by far to outweigh its drawbacks, it should not be overlooked that economic viability, that is competitiveness, is the key criterion. If that is not fulfilled, there will be no nuclear power. This is a point that some within the nuclear community seem to neglect. They argue that in view of nuclear's benefits it would not matter much if it were "a little more expensive". To a certain extent they seem to be justified by the presumption, a topic for research, that even though energy cost rose in the past, its share in the "bread basket" has fallen. As a consequence, they argue, additional cost, whatever for, safety or special applications, should not be a point of great concern. The resulting question is not an easy one to answer. But prudence would require not to assume this "additional degree of freedom". To paraphrase a word ascribed to U.S. President Lincoln: you can fool the market for some of the time,

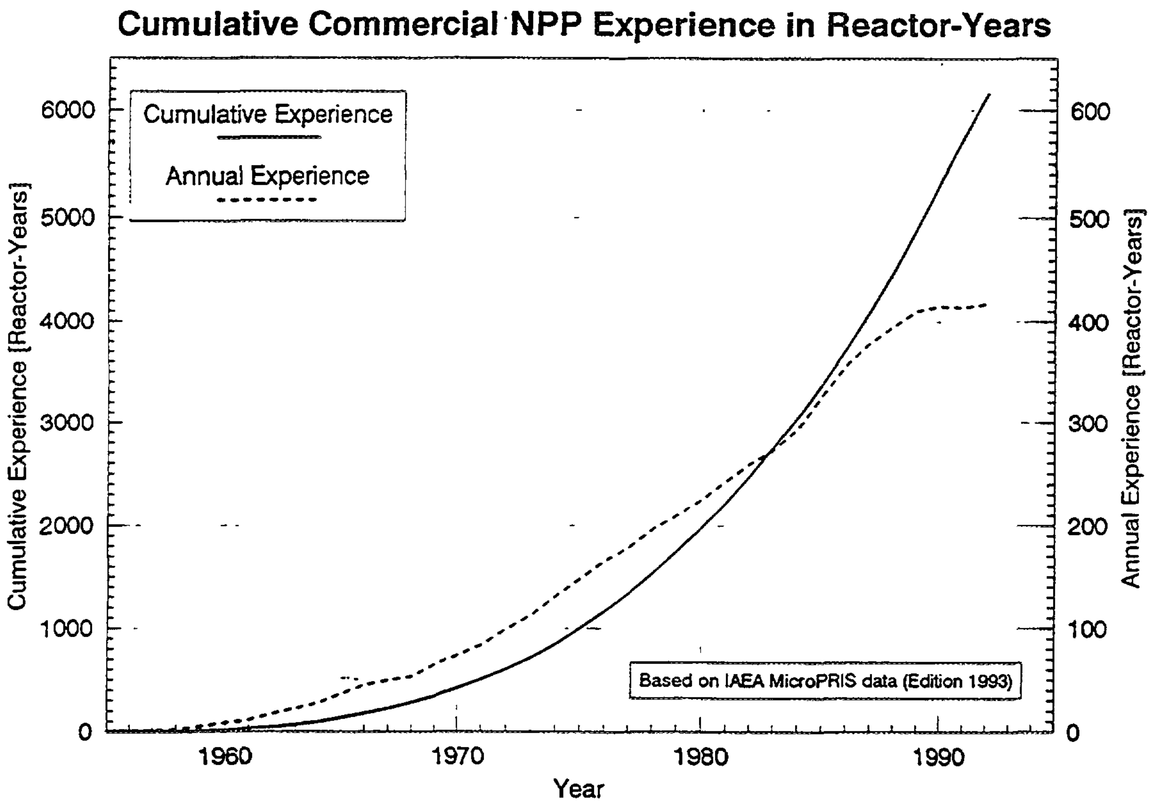


FIG. 1. Cumulative commercial NPP experience in reactor-years.

but you cannot fool the market all of the time. Nuclear has to maintain its economic advantage in spite of its environmental benefits. This then would be an overriding premise for the following discussions.

3. RATIONALE AND BOUNDARY CONDITIONS FOR NUCLEAR DISTRICT HEATING

3.1. The basic rationale

The idea of using nuclear energy for district heating is by no means a very new one, be it for economic or for environmental reasons. At least for the West, and for civilian applications, it dates back to the sixties when, e.g., in Sweden the Ågesta reactor, in operational terms, very successfully supplied 55 MWth to heat a suburb of Stockholm for a number of years. On account of the then very low oil prices, this type of nuclear application was not expanded, however. Renewed interest arose after the first oil crisis in 1973/1974 in a number of countries. This lead, inter alia, to the investigation of different types of dedicated nuclear heating reactors as well as to studies and, more important so, to the realization of projects where heat is extracted from the low pressure stage of a turbine fed with steam generated by a nuclear reactor (nuclear co-generation).

A most successful example for the latter approach is the REFUNA Project in Switzerland where about 100 MWth are being extracted in winter from the Beznau Nuclear Power Plant. REFUNA is also interesting in that it seems to show that district heating is not just a domain for areas with a very high agglomeration of dwellings as is characteristic for big cities with high-rise appartment buildings.

Nuclear co-generation for district heating is being practiced to a larger extent in many countries of the former Eastern block (e.g. Bilbino and Beloyarsk in Russia, or Bohunice and Bukovany in former Czechoslovakia). There is thus much precedent and experience. It is also complemented by the experience with supplying nuclear process heat for industrial applications in many countries (e.g. Douglas Point in Canada, Stade in Germany, or Goesgen in Switzerland).

The basic rationale was, and still is, always the same: fulfilling a need in the most competitive way and doing so in an environmentally beneficial manner. As was already pointed out earlier, this latter aspect is nowadays becoming more and more important. However, its proper assessment in terms of its always present twin condition, namely economic competitiveness, still suffers from not having a generally accepted methodology that helps in making the proper decisions.

If one studies the many proposals for dedicated nuclear district heating plants, as members of the more general class of small or very small reactors, one often comes across yet another rationale. It is seldom clearly spelled out, but it does seem to center around the following argument. Turning to small reactors, for purposes where they have to be small on account of the characteristic application, could help to give a needed boost to the revitalization of nuclear in general. In support of this rationale it is said that the seemingly less demanding technical requirements, including those related to assuring adequate safety, would open up the opportunity for many more entities to participate in the nuclear business and to contribute to its revival. Be this for smaller countries with a weaker infrastructure, or for smaller customers with limited needs, or be it for smaller companies with limited resources, or for appropriate combinations of these key elements is the essence of the argument. A final argument often quoted in support for smaller reactors derives from a general tendency in many countries to "do things at the local level by the local institutions" in an apparent diffuse dissatisfaction with "centralized activities".

Whilst many of the above arguments seem to have some merits, it is currently very difficult to assess their respective weights and to draw general conclusions.

3.2. Boundary conditions for dedicated heating reactors

In this context it is useful to recall what is already in existence in terms of fossil fueled district heating. Figures 2 and 3 show for some selected countries the installed capacity, absolute and per capita, of district heating. The figures would provide a first feeling for what nuclear could contribute and under what conditions.

In a general way the figures show two things. First, installed capacities are relatively large in total capacity. Second, there seem to be distinct differences between the countries looked at, and even between different regions for a given country. These differences already point to something which would have to be considered later under the heading of "Prospects". To anticipate the most general finding: it is difficult to generalize!

The first figure on installed capacity shows that the total capacities are by no means negligible. To work with the figure for Germany, almost 56,000 MWth installed is a large capacity even when considering that the average load factor is less than 2500 hours per year because of climatic conditions, as opposed to 6000 or 7000 hours per year for typical electricity generating plants. What this figure does not show, however, is that these 56000 MWth (peak) are supplied by more than 600 individual, that is non-connected, hot water, or

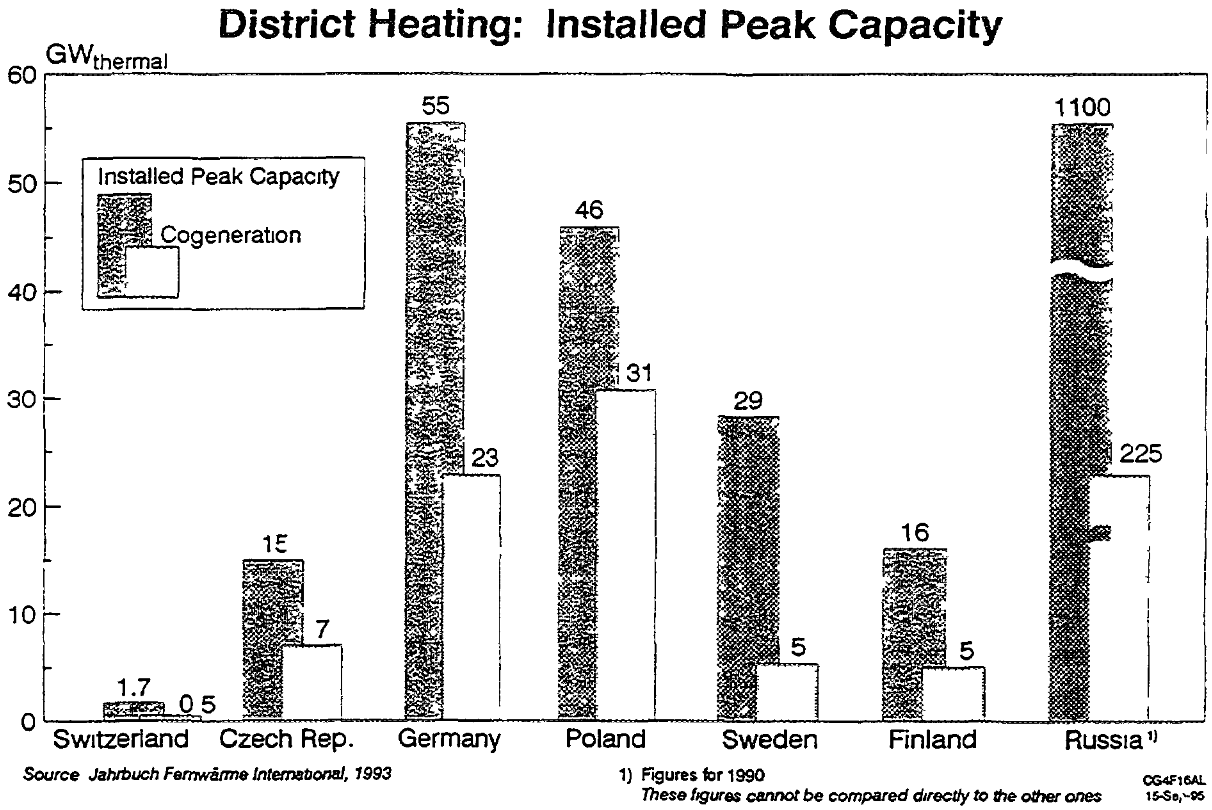


FIG. 2. District heating: installed peak capacity.

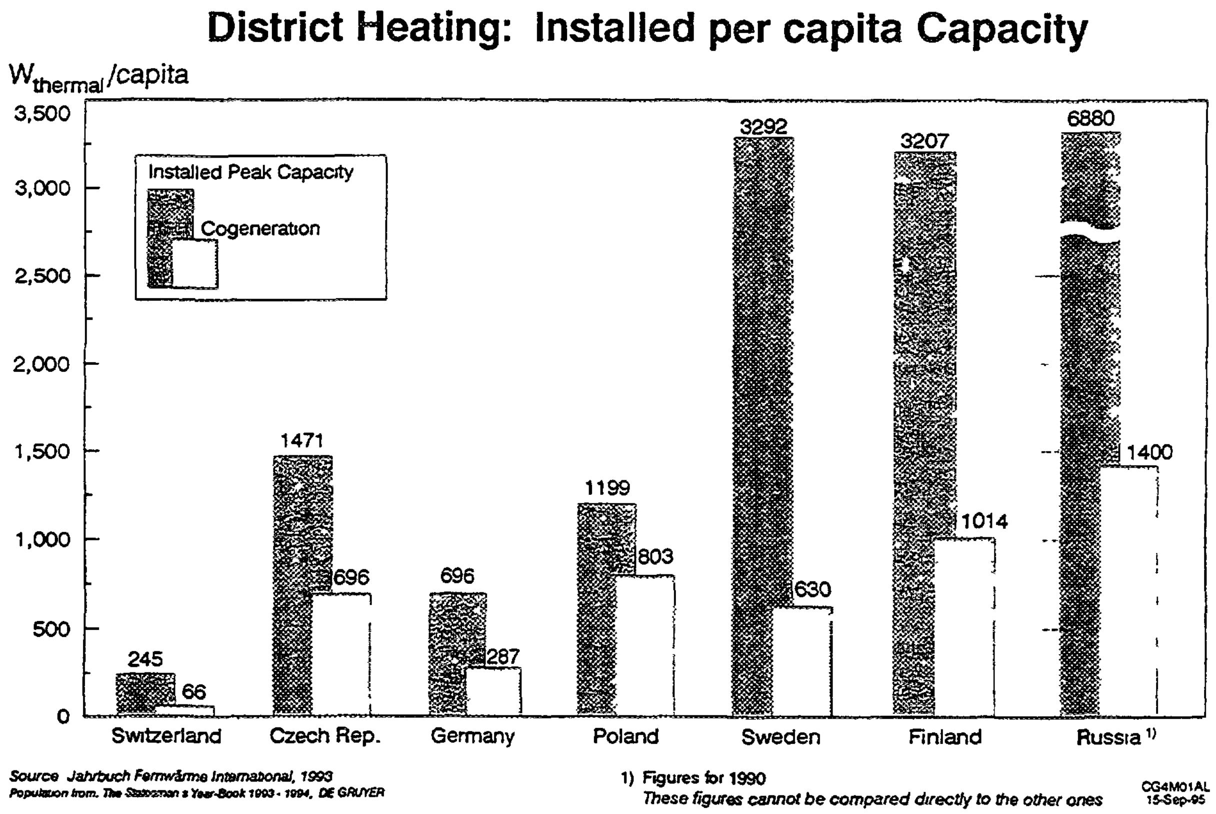


FIG. 3. District heating: installed per capita capacity.

in some cases, hot steam grids. This already points to a first very important aspect of the present discussion: since hot water is expensive to transport per unit useful energy, grids, and thus heat sources, have to be small.

But even that cannot really be generalized as it depends much on demographic factors that can vary largely in a given country. To stay with Germany, the city of Munich has four grids that, if combined, would amount to more than 1500 MWth, thus far more than the average of about 100 MWth per grid that can be inferred, on the average, from the figure given before. The same is true for other cities like, e.g. Hamburg and Berlin. In other words: there is a very large spread in installed capacities.

The city of Moscow, to look to another country as an example, is reported to have about 10,000 MWth of installed capacity. On the other extreme, detailed studies made in Switzerland (or in Canada) point to optimal grid, and thus reactor sizes of distinctively less than 50 MWth, typically 10 to 20 MWth. To repeat: generalization is very difficult from the point of demographics.

But this is not all. Matters become even more difficult as heat, particularly of low temperature as for space heating, can be produced in a number of ways, each relatively close to the other as far as overall economics are concerned. The spectrum ranges from centralized sources, either as pure heat boilers, to co-generation (see fig. 2) for good economic reasons, to "oil-via-truck", or "natural gas-via-pipe", or electric heating, either directly by resistance heating, or indirectly by heat pumps. Direct furnace-heating with solid fuels is almost negligible in many Western countries. For people thinking in terms of electricity generation, with the "relatively cheap big grid" these alternative and quite competitive means of providing residential heat is a circumstance that needs to be appreciated.

The main result of this discussion on the boundary conditions which affect district heating is this: beware of generalizations, even though this is a temptation in any overview. To further illustrate this point, look at fig.3 showing the per capita installed capacity of district heating. The somewhat surprising conclusion is that many countries with a relatively low average population density seem to have a high per-capita installed heating capacity, with the obvious commonality of belonging to "cold clima" zones (e.g. Finland, Sweden, Russia). Again, general conclusions are not easy to arrive at. It is obvious that a case-by-case analysis is needed for any individual heating district. This would seem to be the most important conclusion for the present meeting as will be emphasized later.

3.3. Current status

The preceding sections dealt primarily with general observations in an attempt to explain the many boundary conditions nuclear district heat applications have to face. It was stressed several times that, primarily on account of the multitude of such boundary conditions, it would be difficult to come to specific conclusions that are universally applicable.

A brief historical review of the many approaches reveals this situation. Starting in the optimistic sixties, reinforced and accelerated after the first oil crisis, a large number of approaches and concepts were proposed. Some of them have been investigated to a considerable detail, both analytically and experimentally. And, most important, a very few even reached the status of an integral test and/or construction of a full-sized prototype (China, Canada, Russia).

Above development work was not done in secrecy. In fact, many concepts were presented and discussed at the different fora the IAEA provides within its charter of responsibilities. Table 1 gives a condensed overview covering about the last 15 years. As most of the participants to this Advisory Group Meeting are experts in the field, there is no need to elaborate much on it.

This table is simply meant to lead to the question that we should not avoid, because if we do not ask it, others will. Why, except for China and Russia mainly, has the enthusiasm for dedicated heating reactors almost vanished in other countries? An answer to that question -if a good one can be found- is not only necessary for the next chapter of this paper which is entitled "Prospects", but also, and perhaps even more so, to help that the general nuclear pessimism, abounding in many parts of the Western world, does not creep into those countries that still have or want to maintain a strong nuclear program.

A tentative answer to above question, to be considered mostly as a personal observation of the author, may be given on the basis of the experience in Germany.

To anticipate the answer: the reasons for not actively continuing the Siemens NHR-200 were a mix of economic factors, political considerations, aspects of infrastructure, a certain negative feedback from the difficult situation "normal" nuclear power is currently exposed to, and to something that could perhaps best be described as unfortunate timing. It may be argued that, except for economics, none of the previous factors would have been sufficient alone to cause the current hiatus of the NHR, but in combination they did.

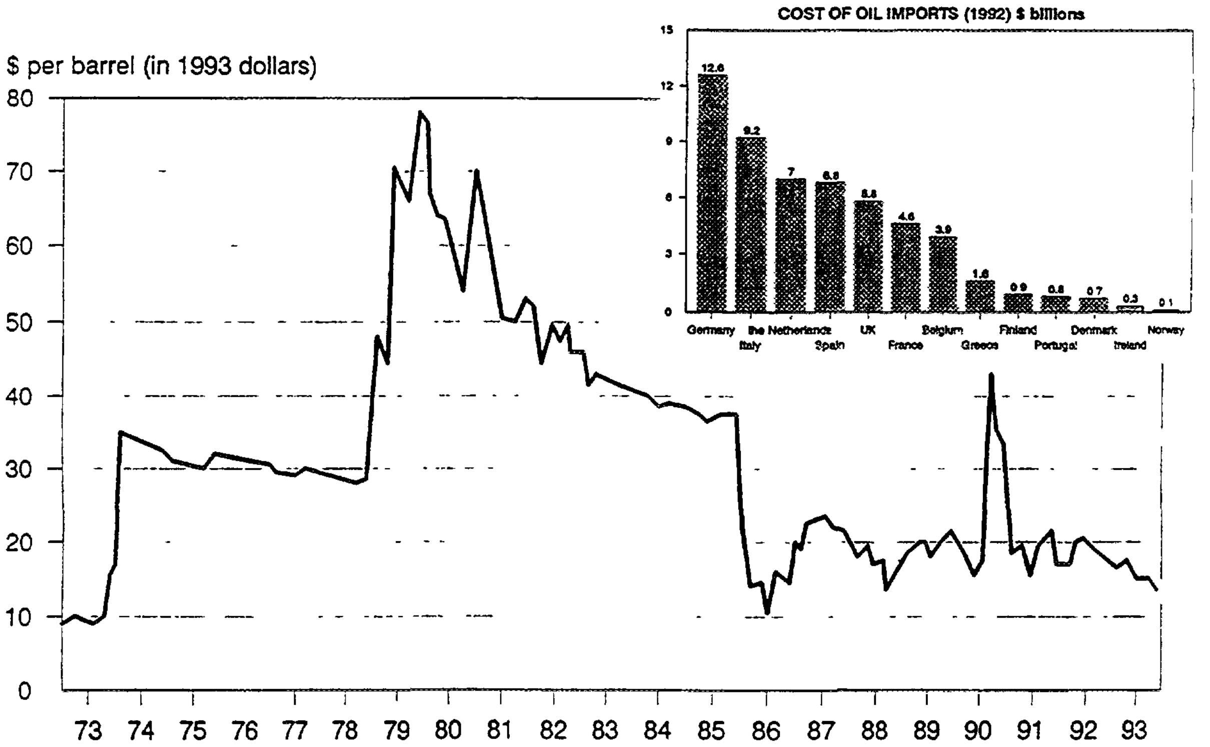
Two things are important in this context. Firstly, except again for economics, all above factors are very country (or even region) specific. Much in line with was said earlier, although within a different context, it is difficult to generalize. In particular, the German situation cannot be representative, for instance, for China or for Russia. The second important point is that both, developers and potential customers had (and still have) no doubts in the general technical feasibility and in the task of demonstrating adequate safety for the German NHR concept. This is an important aspect in the context of the current meeting.

All the factors mentioned so far presumably also exist as topics in other countries. However, both their specifics and their relative weights may be significantly different such that a generalization is not possible. This will be illustrated with some further remarks on the German situation.

To start with the most important factor of energy economics, the conditions in the late seventies were very favourable for dedicated nuclear heating in comparison to fossil alternatives. Coal was, and still is, notoriously expensive (several times world market price). Because of a national policy to keep German mines open, import of foreign coal is severely restricted. Oil was also very expensive and was generally expected to rise even further, quite in contrast to what actually happened (Fig. 4). Gas prices tended to follow those of oil and where thus no threat. In this context it should be noted that the comparison was made on the basis of what was earlier termed "oil-via-truck" and "natural gas-via-pipes" to the individual dwelling, as these modes are dominating in space heating in Germany. Finally, the developers were convinced that they could come up with a heating reactor that would be only marginally more expensive in terms of DM/kW-thermal than a big electricity producing NPP.

Table 1: International Meetings on District Heating Reactors

Meeting	Location/Publication	Date	Reactors Discussed
ENS/ANS/Finnish Nuclear Society, "Low Temperature Nuclear Heat"	Helsinki, Finland ----	August 1977	SECURE-200 (Asea Atom), Thermos-100 (CEA), CAS-50 (Technicatome), IPWR-200 (Interatom), CNSG-400 (B&W), DHAPP-500 (Kurchatov Institute), HTGR-Steamer (General Atomic)
IAEA-TCM, "Nuclear Heat Application"	Cracow, Poland **Topical Report**	December 1983	AST 300-500, NHR-200 (KWU), Slowpoke-10 (AECL), HHR-10 (EIR), Various HTRs for process heat application (FRG, USA, Japan)
IAEA-AGM, "Potential of Low Temperature Heat Application"	Würenlingen, Switzerland **TECDOC-397**	September 1985	AST-500, SHR 10-50 (EIR), Slowpoke-10, AST-300 (USSR & Czechoslovakia), CAS, Thermos, SECURE
IAEA-AGM, "Low Temperature Heat Applications: Nuclear Power Plants for District Heating"	Prague, Czechoslovakia **TECDOC-431**	June 1986 1987	Slowpoke-10, NHR 5-200 (INET, PRC), CAS, Thermos, NHR-200, SHR-10, Geyser-10 (SIN), GHR (BBC & EIR) AST-500, HERE-200 (UNI-Duisburg)
IAEA-AGM, "Small reactors for Low Temperature Heat Applications"	Winnipeg, Canada **TECDOC-463**	1987	Slowpoke, GHR, SHR, Geyser, [Triga Power System (TPS, General Atomic), small co-generation plant]
"Nuclear Applications for Steam and Hot Water Supply"	**TECDOC-615**	1991	Most of above concepts



Source. Datastream, International Energy Agency

FIG. 4. Evolution of oil prices in Europe.

As to the next factor, policies, one was already mentioned in the paragraph before. The second concerned new emission standards. Within about 10 years from the first idea of a NHR, all fossil fired power plants had to implement expensive backfitting measures. That affected also those plants that provide energy to the district heating grids. As there was also at that time a certain excess capacity in electricity generation, with the consequence that many coal fired co-generation plants were almost only used to supply heat, another rationale was seen for dedicated heating reactors: to replace those coal stations that had to be backfitted. But the heating reactor was not yet ready, and upgrading coal plants "won" the time race.

Regarding the point of infrastructure, it is important to know that Germany has no law that generally mandates which type of heating is to be provided for individual dwellings. This in turn limits the growth of district heating (since many years of the order of 2 to 3 percent per annum) in accordance with competitiveness, and thus would also put certain limits to an expansion of nuclear heating plants produced in series. Another infrastructure aspect is that many residential areas are relatively new, such there is little potential for a substantial re-construction of "old" heat sources which could positively influence the growth of district heating.

The German NHR development was also negatively affected by the slowdown of the nuclear business in general, which to explain would be beyond the scope of this paper. This caused needed development funds to dry up and made potential customers hesitate to engage themselves in a new and controversial technology. This in particular, because many district heating facilities are owned by small communal utilities which thus far have very little, or no nuclear experience, and which are much more exposed than private utilities to the split that runs through the big political parties on the application of nuclear power.

By unfortunate timing is in essence meant that the mandated implementation of stringent anti-pollution measures fell together with a slow-down of the nuclear business and an increased public controversy over this energy source. When taken together, all the factors mentioned before turned out to be very detrimental to NHR development in Germany.

4. PROSPECTS

From what was said so far, two things would seem to be obvious. First, environmental concerns, particularly regarding an unrestricted expansion of the use of fossil fuels, almost inevitably pre-programmed, not only supports, but also necessitates expansion of nuclear power into the dominant sector of providing sensible heat, much of it at relatively low temperatures. It would also seem that the groundwork for such an expansion should be taken up as soon as possible in view of the relatively long time constant associated with establishing a proven nuclear system. The basic rationale: environmentally clean energy at acceptably low cost is not only still valid, but is gaining weight in view of the increasing concerns about global warming.

The second point is that laying groundwork and subsequent expansion is a process that cannot be accomplished in more or less identical ways in all countries where (nuclear) district heating is a meaningful option. The key factors that have to be considered in each individual case simply vary too much with good reasons.

This should have become obvious from looking at the German case and comparing it to the situation in China or Russia. This comparison was not made since it was neither the purpose of the paper, nor is the author sufficiently knowledgeable on the specifics of these two latter countries. In addition, it would seem too presumptious to do so on this occasion where the respective experts are present.

However, this author has had the privilege to intensively work together with the specialists of the host of this meeting, INET, on the NHR development for China. And from the many discussions he had, he is convinced that the NHR does have a future in China, because circumstances are very different from Germany. This would also seem to be the case for Russia, even though the author has had only few occasions to discuss the prospects of heating reactors in that country.

5. CONCLUSION

The conclusion can be brief. The case for nuclear district heating is sound if one looks beyond the immediate future. Its practical implementation depends on a number of factors that can differ largely from country to country, requiring thus a case-by-case approach. As it currently stands, countries like China and Russia, although very different from each other, have good opportunities, again for different reasons, to further the idea of district heating with reactors specifically designed for this purpose.

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RESEARCH AND DEVELOPMENT OF THE CHINESE NUCLEAR HEATING REACTOR

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Abstract

The paper presents the significance of nuclear heat application in China as well as the development status, main design features and safety concepts of the nuclear heating reactor exploited by INET.

1. SIGNIFICANCE OF NUCLEAR HEAT APPLICATION IN CHINA

The energy supply has been one of the major issues impacting highly on the socio-economic development in China. According to a study on the future energy strategy, only about 80% of the energy demand in China could be satisfied with coal, oil, natural gas and hydropower by 2050.

Furthermore, China's energy system is characterized by the predominance of coal consumption. Since coal resources are highly concentrated in the North and Northwest of China, while the industry and population centers are mostly located in the Northeast, East and Southeast of China, the coal transportation with so great an amount poses an enormous problem in both economy and technique. For example, at present coal transportation requires 40% of railway freight and 30% of waterway freight. Moreover, according to the monitoring of data in some cities, more than 60% of the observed SO_2 was from coal burning, and the environmental pollution caused by massive coal burning will become unallowably serious in a number of cities if no action is taken immediately.

Because the energy consumption as sensible heat at temperatures less than 150°C accounts for about 25% of the total energy consumption in China, therefore, in order to mitigate the energy shortage, the environmental pollution caused by coal burning and the overburden on coal transportation, great attention has been given to the R & D for the nuclear heating reactor (NHR). The development of the NHR has been one of the national key projects in science & technology in China since 1980s. The NHR could be used for district heating, air conditioning, seawater desalination and other industrial processes. Therefore the NHR could substitute nuclear energy for fossil fuels and change the energy composition in China. This will be of significance in both the socio-economic development and the environmental protection.

2. DEVELOPMENT STATUS OF THE NHR

Research work on possible application of nuclear heat was initiated in the early eighties. During 1983-1984, INET used its existing pool type test reactor to provide space heat for the nearby buildings. Meanwhile, two types of NHR, i.e. deep pool type NHR and vessel type NHR, have been developed by INET. Based on the heating grid conditions in China and the comparison among various design concepts of the NHR, the vessel type NHR has been selected as the main development direction. As a result, construction of a 5 MWt experimental NHR (NHR-5) started in 1986 at INET. The reactor was completed in 1989 and has been operated successfully for space heating since then. In the meantime, a number of experiments have been carried out to demonstrate the operating and safety features of the NHR [1].

In order to investigate the comprehensive uses of the NHR, some experiments, such as electricity generation with low pressure steam under co-generation mode, and air conditioning for large building areas using the Lithium-Bromide absorption process, have also been performed at the NHR-5. In addition, a nuclear seawater desalination experiment is under way.

By its proven excellent performance, the NHR has made more and more impressions on Chinese society. Up to now, nearly 20 cities and utilities are very interested in introducing the NHR into their local energy system. Therefore, a commercial size NHR with an output of 200 MWt (NHR-200) has been developed.

With the purpose of the NHR commercialization, the R & D program for the NHR, which contains 18 topics in total, has been classified again as one of the national key projects in the Eighth Five-Year Development Plan (1991-1995), and it has been decided to build a NHR-200 demonstration plant in Daqing city in Northeast China. The siting report, the environmental impact assessment report, and the feasibility study [2] for the project have been completed and approved by the respective authorities. The basic design of the NHR-200 has recently been finished, and its detailed design is being carried out at present. It is expected that the first NHR-200 will be put into operation in 1999 and several 2x200MWt nuclear heating plants could be constructed in China in late 1999.

It is worth to mention that during the last few years the INET has closely cooperated with Siemens-KWU [3], Germany, and with the former EIR, Switzerland, on the R & D for the NHR. Recently the INET also has information and personnel exchange on the matter with the respective institutions in Russia and other countries.

3. TECHNICAL DESCRIPTION OF THE NHR

3.1. Main considerations in the design

Some special considerations related to the nuclear heat application should be reflected in the NHR design.

Since sensible heat is rather expensive to transport over a long distance, the NHR should be sited in close vicinity to the populated area served, closer than for power reactors. This means that the safety requirements for the NHR should be more restrictive than that for a present-day nuclear power plant. For example, the NHR should be designed to be so safe that a core melt down should be excluded, and that an emergency plan for evacuation of persons living nearby should never be necessary.

Moreover, due to the heat grid size limitation, the NHR, designed mainly for the purpose of the district heating, is relatively small in thermal power output, and its effective full power hours are generally fewer than that of a power generation plant. These circumstances force the NHR to face an economic impediment.

It is clear that the task for achieving this safety goal and for lowering the capital investment as well as the heat generating cost has become a major concern in the NHR design. Among other things, the NHR should be designed with inherent characteristics, say, with excellent self-protection abilities, including self-control and self-limitation in power. Its safety systems should function under a passive principle based on natural physical laws. Furthermore, more attention should be given to the elimination of complex systems and certain components so that the plant could be simple, reliable, easily be constructed and maintained. In addition, a comprehensive application of the NHR should also be kept in mind for improvement in its load factor and availability.

3.2. Technical description

As a consequence of the above, the NHR has been designed with a number of advanced and innovative features to achieve its safety goal and economic viability.

Fig.1 and Fig.2 show the reactor structures of NHR-5 and NHR-200 respectively. The main parameters can be found in Table 1. Their essential design features are the same. The NHR is a vessel type light water reactor with an integrated arrangement, natural circulation, self-pressurized performance, and a dual vessel structure. The core is located at the bottom of the reactor pressure vessel (RPV). The primary heat exchangers (PHEs) are arranged on the periphery in the upper part of the RPV. The system pressure is maintained by inert gas and steam. A containment fits tightly around the RPV so that the core will not become uncovered under any postulated coolant leakage within it. Reactor coolant is circulated by density differences between the hot and cold regions inside the RPV. There is a long riser on the core outlet to enlarge the natural circulation capacity.

Gadolinium oxide as a burnable poison is used to control the reactivity along with B_4C control rods. The reactor coolant does not contain boron acid during normal operation.

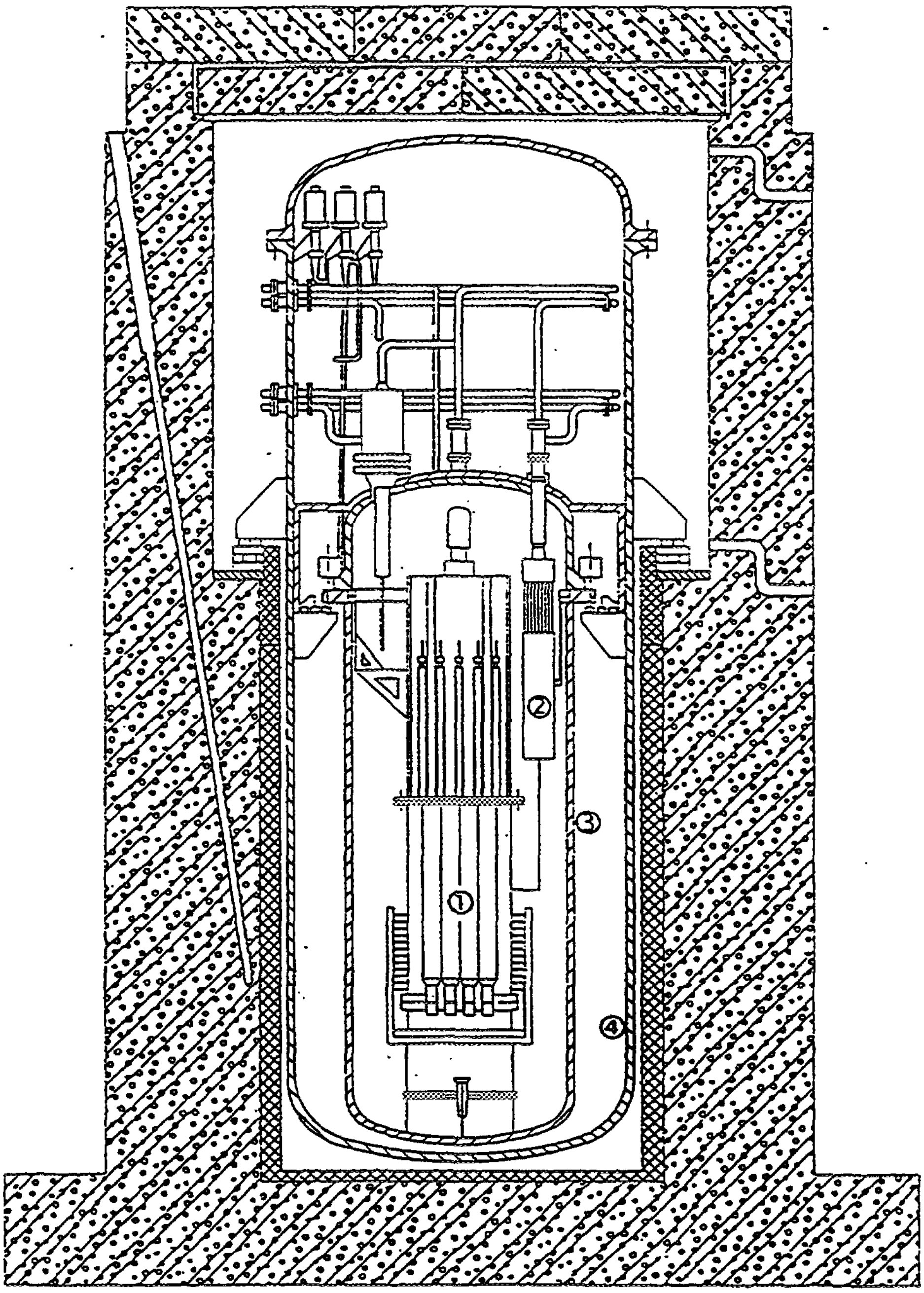
A hydraulic control rod drive system is adopted in the NHR, which is designed on the "fail-safe" principle, i.e. control rods will drop into the reactor core automatically under loss of power supply, depressurization, pipe break and pump shut down events.

Spent fuel assemblies are stored in racks around the active core. This solution greatly simplifies the refuelling equipment and eliminates the necessary space in the reactor building.

A simplified schematic flow diagram of the NHR is shown in Fig.3. The nuclear heat supply system (NHSS) contains triple loops. Primary coolant absorbs heat from the reactor core, then passes through the riser and enters the PHEs, where its heat content is transferred to the intermediate circuits. Finally, heat is delivered to the heating grid via the intermediate heat exchangers. An intermediate circuit is needed in the NHR to keep the heating grid free of radioactivity.

There is no emergency core cooling system in the NHR. The residual heat removal system (RHRS) is the most important safety system for the NHR. It is designed with passive characteristics. The decay heat will be dispersed to the ultimate heat sink by natural circulation.

A boron acid injection system, as a secondary reactor shutdown system, will be operated by gravity when an ATWS accident occurs.



①core
②primary heat exchanger
③reactor vessel
④containment

Fig.1 The NHR-5 structure with dual vessel

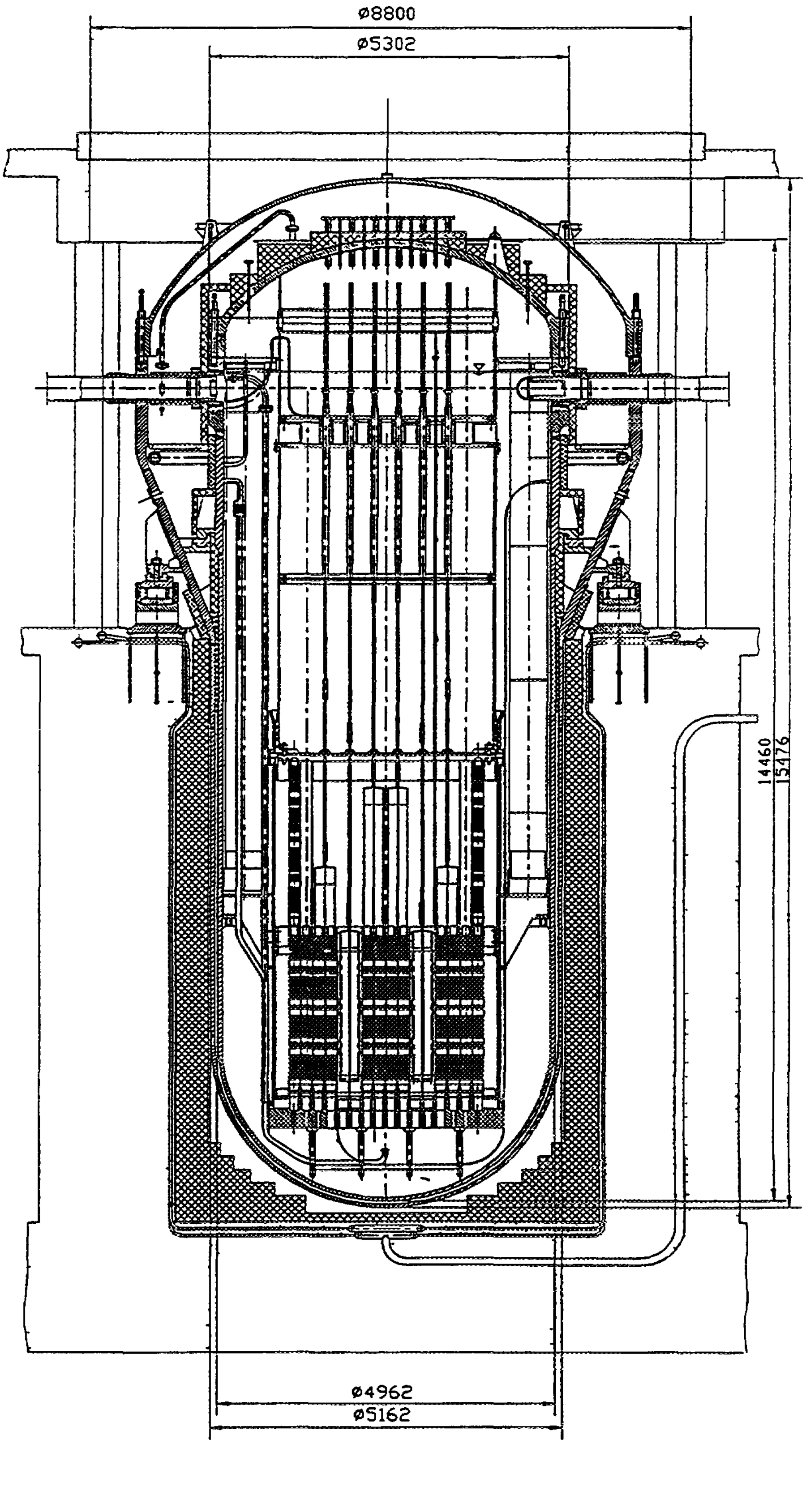


Fig.2 The NHR-200 primary system arrangement

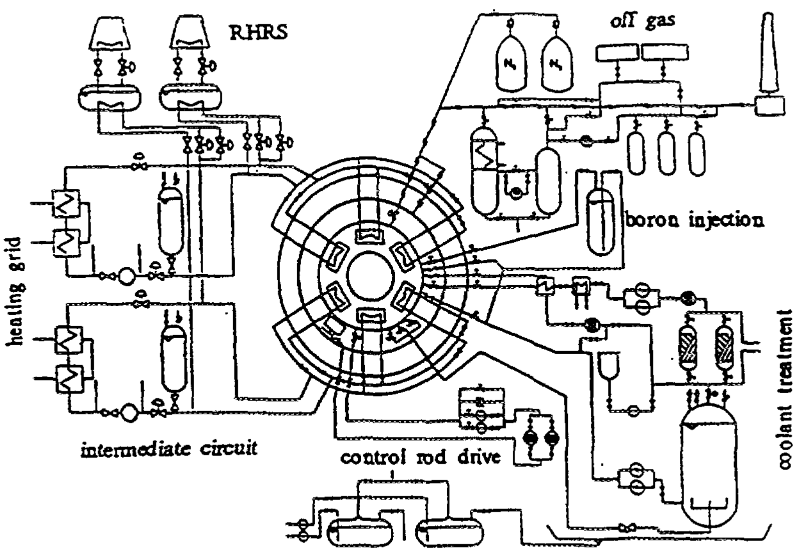


Fig.3 Schematic system diagram

The NHSS and all its auxiliary system, including the rad-waste processing systems, are housed in one building complex.

TABLE 1: MAIN DESIGN DATA OF NHR

Reactor	NHR-5	NHR-200
Thermal power, MW	5	200
Primary system pres. MPa	1.5	2.5
Core inlet/outlet temp.°C	146/186	140/210
Average linear heat rate, kw/m	5.6	7.67
Volumetric power density, kw/l	26	36.23
Number of fuel assemblies	16	96
Number of control rods	13	32
Active core height, m	0.69	1.9
Active core diameter, m	0.57	1.9
Inventory of UO_2, t	0.51	14.87
Enrichment of initial core, %	3	1.8/2.4/3.0
Refueling enrichment, %	3	3
Intermediate circuit pres., MPa	1.7	3.0
Intermediate circuit temp. °C	102/142	95/145
Heating grid temp. °C	90/60	130/80

4. SAFETY CONCEPT AND FEATURES OF THE NHR

The safety concept of the NHR is fundamentally based on its excellent inherent characteristics, allowing passive safety instead of engineered safety features.

The NHR is operated under low pressure, low temperature, low power density and low radioactivity content in the primary coolant. A huge subcooled water inventory results in a high thermal inertia in the primary system. A large negative temperature reactivity coefficient has been achieved in the core nuclear design. Furthermore, the innovative hydraulic control rod drive mechanism and a great inertia in the plant lead to exclude the rod ejection event and any other large uncontrolled reactivity additions. Therefore any transients and accidents can be very well counteracted. The core decay heat is transferred to the atmosphere by natural circulation. This makes core cooling more reliable. Moreover, the integrated arrangement, low operating pressure, low neutron fluency to the RPV and all in-vessel penetrations (only with small diameter) located in the upper part of the RPV lead to a low probability, and less seriousness, of loss of coolant accidents. The loss of primary coolant is limited to the extent that the core will never be uncovered, so an emergency core cooling system is not necessary for the NHR.

Intensive safety analyses have been conducted to evaluate the overall performance of the NHR. The results from the analyses of any design basis accidents are summarized as follows:

1) The minimum DNBR is always greater than the safe limit. The fuel element cladding is not overheated and damaged.
2) Peak pressure in the primary system is far below its design pressure and the integrity of coolant pressure boundary will be properly maintained. The safety valve on the RPV will not open until a "loss of main heat sink ATWS" occurs.
3) The reactor core will never be uncovered. Proper fuel cooling is ensured.
4) The maximum fuel enthalpy is much lower than the safe limit.
5) A release of radioactivity is much less than the prescribed limit.

Analyses on beyond design basis accidents have also been performed. The results have indicated again that a core melt down in the NHR could be excluded.

The overall excellent safety characteristics of the NHR have also been demonstrated by the experimental and operational data from the NHR-5.

5. SUMMARY

Adequate energy supply and environmental protection have become a major concern in the socio-economy development. Based on China's practice, a vigorous action has been taken to speed up the exploitation of nuclear energy for heat application.

The NHR with a number of advanced design and safety features can serve as a safe, clean and economic energy source. As its R&D program is well advanced, the NHR being a new promising energy system, would play an important role in resolving the future energy and environmental problems in China.

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REVIEW OF NHR ACTIVITIES IN THE RUSSIAN FEDERATION

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Abstract

NHR development activities in the ex-USSR were initiated in the 1970s mainly due to a growing deficiency of organic fuels needed for heating large cities in the European part of the country Construction of two pilot nuclear district heating plants with AST-500 NHRs was started in the early 1980s, and by 1989 the first unit in Gorky NDHP was nearly 90% completed. Current activity in this field is concentrated on upgrading the AST-500 design and on the development on this basis of a whole series of heating-only and co-generation reactor plants of unit power ranging from 30 to 600 MW A brief description of the AST-500 reference NHR design features is given, as well as of the R&D activities that have been carried out for the design decisions and safety validation

1. INTRODUCTION

The fraction of fossil fuels being consumed in the Russian Federation for the generation of low grade heat for industrial and district heating purposes amounts to 30-40% of the total consumption, where the most qualified fuels such as natural gas and oil comprise the main part. Therefore, the use of nuclear fuel for heating and hot water supply could become a significant contributor to the improvement of the fuel consumption balance of the country. It would thus relax the problems related to the remoteness of the most energy-consuming regions from the main fuel production areas, which are mainly located in the East of Russia.

Besides, it was realized that the substitution of fossil fuel by nuclear one would give an important social-economic effect through attenuation of the adverse influence of the traditional power technology on the environment, particularly in large cities.

By the early 80s favourable conditions existed in the ex-USSR for the introduction of nuclear reactors into the district heating grids. Such conditions were promoted by the following factors:

- positive experience that had been gained by the national power industry in centralized district heating using large energy sources and heat-distributing grids;
- a stable tendency to increase the heat load concentration and unit power of district heating sources;
- deficiency in fossil fuels and large expenditures for its long-distance transportation;
- mastering of large-scale manufacture of equipment for NPPs by the national industry;
- experience gained in construction and operation of powerful nuclear power plants with steam extraction for heating of nearby satellite towns.

All these factors stimulated the start of the first NHR of AST-500 type development and pilot heating plants construction [1].

The analysis performed to assess both the current and prospective levels of municipal heating loads and their concentration showed that in the European regions of Russia there are several tens of large heat-consuming centers with heating loads of more than 1000 GCal/hr. For the majority of the regions, the anticipated increment of heating load for the nearest future allowed to use twin-unit district heating stations of a total thermal power around 860 GCal/hr. A feasibility study showed the competitiveness of such a plant with fossil-fuelled heating stations.

Pilot twin-unit nuclear district heating stations with 500 MW(t) AST-500 reactors were intended to provide heat for the highly-populated districts of the large industrial cities of Gorky, Voronezh and Arkhangelsk. All these cities suffered from an acute shortage of fossil fuel which decelerated the municipal activity in the field of domestic building.

A minor construction activity is now under way for the Voronezh NDHP only, while the Gorky NDHP has been cancelled in 1991 due to strong protests of concerned local population. Recently, the designs for the advanced commercial NHRs AST-500M and AST-600 were developed, and a feasibility study for the construction of such reactors is being carried out for the Khabarovsk NDHP (Far East of Russia).

According to the recently issued Concept of Nuclear Power Development in the Russian Federation [10], the pilot twin AST-500 NDHP should be realized until 2000 (construction work completion and power start-up). In the same period of time, pilot small power electricity and heat co-generation nuclear plants are planned to be constructed to satisfy the energy needs of remote regions in the Far North and the Far East of the country.

2. AST-500 REFERENCE DESIGN CONCEPT AND PRINCIPAL DECISIONS

The AST-500 reactor plant principal flow diagram, the main engineering decisions and the main parameters are given in Table 1 [2,3].

The AST-500 NHR conception was preceded by the development of NHR-specific requirements for the safety of such nuclear plants. They were adopted as a supplement to the basic National Code for NPP safety. These requirements envisage the necessity to exclude fuel melt at loss of integrity of any reactor-related pressurised system and to account for external impacts like air crash and shock wave. Besides, stringent requirements to radiological safety were introduced, including a collective dose limit specification [4, 5].

A PWR-type nuclear reactor with positive intrinsic properties, particularly as it concerns nuclear safety, such as neutron power self-control and self-limitation, was adopted for the AST-500 design.

The chain reaction self-control capability in virtue of the negative temperature, power and void reactivity feedbacks was enhanced additionally in the AST-500 thanks to a rejection of soluble boron reactivity control and to the use of burnable poison in the core.

The feature of the NHR as a source of low-grade energy gave the opportunity to adopt much lower parameters for the primary system (T=200°C, P=2.0 MPa) compared to the traditional light water reactors. Conforming to the specific heating purposes, the required unit power of NHRs can be considerably lower than that generally accepted in contemporary nuclear power (up to 500 MW), depending on the specific heat load characteristics of the heating grid connected to a given NHR. Along with the reduction of coolant parameters, the

TABLE 1: AST-TYPE NHP DEVELOPMENT HISTORY IN THE EX-USSR

First phase (1977-1986)

* AST-500 Basic Design development, R&D activity
* Pilot AST-500 units construction in Gorky and Voronezh

Second phase (1986-1989)

* AST-500 design updating
* Peak of construction activity on both Sites
* Gorky NHP independent safety review (IAEA Pre-OSART mission)

Third phase (after 1989)

* Voronezh NHP design updating
* AST-500 advanced design development
* Series of AST-type NHP designs development
* Public acceptance and financing problems growing

NHP CURRENT ACTIVITY

* Feasibility study for Khabarovsk NHP (2 x AST-500M)
* Voronezh-NHP licensing (environmental impact review)
* No construction activity
* Minor design and R&D activity on advanced heating reactors
* AST-based co-generation (electricity and heat) NPPs development
* Activity to influencing public acceptance

NHRs PREMISSES IN THE RF

* DH - major contributor to fossil fuels consumption
* Considerable remoteness of fuel production fields
* Availability of powerful district heating grids
* Growth of heating loads and their concentration
* Positive experience in DH from operational NPPs

NHRs Motivation in the RF

* Large - scale substitution of fossil fuels
* Environmental benefits
* Scale down of fuel transport flows
* Competitive generating cost
* Improvements in working conditions and standard of life

NHRs IN DISTRICT HEATING GRIDS

Heat power appr. 50% of max. heating load

* 4500-5500 eff. hr operation per year
* At least two NHRs on a site
* Load - follow mode of operation
* Direct water rated temp. - 150°C

AST-500 BASIC DESIGN OBJECTIVES

* Simplicity and Reliability
* Invulnerability to Multiple Failures
* Immunity to Personnel Errors
* Large Grace Period
* Corium Confinement in RPV
* Exclusion of Population Evacuation

NHP Specific Safety Requirements

* Fuel melt exclusion
* External impacts consideration
* Reduced design limits for fuel failure
* Minimized radiological impacts

AST-TYPE NHR BASIC DESIGN FEATURES

* Integral PWR in guard vessel
* Natural convection of primary coolant
* 3-circuit heat transport scheme
* Low coolant parameters
* Low core power density

HEAT REMOVAL PRINCIPLE FOR PRIMARY CIRCUIT OVERPRESSURE PROTECTION

* Passive self-actuated safety systems
* Proven technology (PWR, operating or tested prototypes)

AST-TYPE NHPs BASIC DESIGN DATA				
	Parameter	Design Concept		
		AST-500M	AST-200	AST-30B
1.	Heat output, MW	500(600)	200	30
2.	Primary coolant			
	- pressure, MPa	2.0	2.0	0.54
	- inlet/outlet temperature, °C	131/208	147/208	70/144
3.	Secondary coolant			
	- pressure, MPa	1.2	1.2	1.08
	- IHX inlet/outlet temperature, °C	88/160	113/155	58/109
4.	Grid circuit			
	- pressure, MPa	2.0	2.0	1.57
	- GHX inlet/outlet temperature, °C	70/150	70/150	50/95

AST-500 Design Validation

* NHR - related experimental base
* NHR thermal - hydraulics
* natural circulation
* reactor hydrodynamics
* boiling crisis
* two-phase flows
* coolant outflow at LOCAs
* NHR safety at accidents
* RPV/GV structural mechanics and seismic stability
* RCS chemistry
* NHR neutron physics
* Functional tests of equipment and components

Design Objectives for Advanced AST Development

* Gaining better economics
* Simplification and standardization of design decisions
* Improving reactor equipment reliability
* Attaining longer plant lifetime
* Improving stability to personnel errors, internal and external impacts

core power density was reduced by 3-4 times down to 27 kW/l and the fuel specific heat rating down to 10 kW/kg. Such combination of parameters assures that the energy accumulated in the core is low which, in turn, assures that operating transients are slow. Besides, it also reduces considerably in-core accumulated radioactivity and delays the release of radioactive products from the relatively "cold" fuel ($T < 500°C$).

Natural convection of primary coolant in all operational modes is the essential feature of the AST-500 reactor. It ensures the independence of both the primary system due to complete elimination of active means for forced circulation of coolant and power consumers. It enables to exclude the complicate transient modes which are characteristic for reactors having forced coolant circulation and which are caused by start-ups and stops of reactor coolant pumps, which results in unfavourable thermal impacts upon the reactor structures.

To solve the complex safety problems, an integral design was adopted for the NHR, with arrangement of the primary heat exchangers and the pressurizer immediately in the reactor pressure vessel [6-9]. This design eliminates large-diameter primary pipelines. Besides, all auxiliary reactor-related pipes are arranged in the upper part of the reactor vessel. The height of water above the core up to the outlet nozzles is approximately 8 m, the water volume for evaporation is approximately 130 m^3.

The enlarged water gap between the core and the reactor pressure vessel decreases significantly the fast neutron fluence on the vessel down to approximately $10^{16} n/cm^2$, thus eliminating the problem of radiation embrittlement of the vessel's steel.

The large specific water inventory per unit power in an integral reactor provides for accumulation of large amounts of heat, and determines a considerable inertia in case of accidental events associated with loss of heat removal from the reactor. Due to the reactor heat accumulation capability, the pressure limit would be achieved only after 2 hours into an accident. The availability of such a time margin allows to eliminate any automatic actions for the actuation of the AST-500 heat removal systems.

The essentially new engineering decision for the AST-500 was to use a guard vessel that houses the entire reactor unit. Its main function is to keep the core covered with water and thus to exclude fuel element melting in reactor vessel loss-of-integrity accidents. The guard vessel (GV) eliminates the need for an emergency make-up system. At the same time the GV serves as a radioactivity confinement system which, in contrast to conventional containments, localizes radioactive products in a small volume in the immediate vicinity of the reactor.

The passive heat removal principle was used and validated in-depth for reactor overpressure protection. The capability to reduce the coolant temperature by the heat removal means available, and in a such way to decrease effectively the reactor pressure due to the close thermal coupling between the primary and secondary circuits through the built-in heat exchangers, provides the reactor with a reliable overpressure protection without the use of safety valves. So, the necessity of blowing off primary coolant in transients was eliminated.

The emergency residual heat removal system operates under natural convection of water in all circuits up to the ultimate heat sink, with no need of power for operation over several days. One of the ERHR channels is capable to provide a pressure reduction in reactor in accidents. As reliability is concerned, this system is not inferior to the reactor emergency protection system.

Pilot-operated relief valves installed on the secondary system pressurizers are actuated both to a primary pressure signal and directly by the action of the secondary medium pressure. The number of valves and the flow rate characteristics were chosen to ensure a primary system overpressure protection by removal of heat through the PORVs. So, a back-up heat removal channel is provided.

A twin isolation valve system is provided to limit discharges of primary coolant in case of any pipeline rupture, or loss of integrity of primary system components located beyond the GV boundary. The valve type was chosen to provide their closure without external energy supply. They close due to the force of a precompressed spring following a compressed air blow-off.

A remotely-operated boron injection system is provided for bringing the reactor to a subcritical state under cold, clean conditions, should a large number of control rods fail to insert into the core (back-up safety system).

For assured protection of the heat consumers, a three-circuit flow scheme is used for heat transport from the reactor. A pressure barrier is provided between the secondary and the grid circuits, thereby preventing radioactive products to ingress into the heating grid at primary/secondary HXs loss-of-integrity accidents.

3. NHR-RELATED EXPERIMENTAL BASE

All components and systems of the AST-500 reactor plant were tested comprehensively using appropriate rigs and test facilities created by the reactor plant designer (OKBM) and his sub-contractors. Their characteristics and operating processes were studied in depth on various models, analogues and prototypes [11, 12]. Many institutions were involved in these activities.

To check on a system level, and to validate the thermal-hydraulic characteristics of the NHR coolant natural convection circuit, as well as to investigate the plant emergency modes, special test facilities were created, including multi-channel thermophysical rigs, at OKBM, large-scale circulation loops at CKTII and at the Kurchatov Institute of Atomic Energy (IAE). The tests were performed in the entire range of the plant parameters' variation. The mechanisms of reactor coolant natural convection were studied depending on the power of the core simulators, pressure level, etc.

The reactor hydrodynamics were investigated at the Central Air-Hydrodynamic Institute, using a 1:4 mock-up of the AST-500 and air for the simulation of coolant flow.

Experimental studies of boiling crisis have been carried out independently on various thermal-physical rigs at OKBM, IAE and CKTII. The tests have been performed with six experimental assemblies representing the real fuel rods with bundles of electrically heated simulators.

The analysis of the thermal-technical reliability of the AST-500 reactor core which was carried out according to a special technique proved the availability of large design margins for fuel assembly heat power.

Investigations of two-phase flow and pressurizer operation, as well as of coolant outflow after primary circuit loss-of-integrity accidents, were carried out by the Power Research Institute.

The basic experimental work for the NHR safety validation concerning primary circuit loss-of-integrity accidents was performed at CKTII using a reactor model allowing to study thermal and hydraulic processes in the primary circuit at accidents. These experiments gave data on the variation of all essential thermal-hydraulic parameters characterizing the emergency process development during AST-500 primary circuit LOCAs. Comparison of the obtained experimental results with analytical data has shown that the mathematical models and the calculation codes used ensure a sufficient accuracy for design purposes.

A study of the reactor pressure vessel rupture mechanism has been performed by the Machinery Research Institute, CNIIMASH, and by OKBM.

The development and optimization of the water chemistry technology for the AST-500 primary circuit also required a large number of experiments. So, radiation-chemical processes have been investigated at the in-pile experimental loop of the MR test reactor at the Kurchatov Institute. The tests were carried out under both non-boiling and boiling conditions. They allowed to validate the need of primary circuit make-up with hydrogen to suppress coolant radiolysis and oxygen concentration build up. Physical simulation and investigation of gas-exchange processes in the primary circuit with a built-in steam-gas pressurizer have been carried out on special test rigs at OKBM. Based on the experiments, the technique and calculation codes were developed for the analysis of the gas distribution in a pressurizer and of the steady-state gas concentration in primary coolant.

Experimental studies of the neutron-physics characteristics of the AST-500 core were performed initially on small critical facilities using physical models which were composed of full-scale fuel assemblies, and then with a full-scale core model at the core manufacturer's site. On the models "cold" and "hot" critical experiments were performed for verification of the calculation codes which allowed to carry out the necessary check calculations. The analysis has shown that the complex of neutron-physics computer codes allows to define with high engineering accuracy the basic physical characteristics of the AST-500 core models.

According to the program of the AST-500 core acceptance tests, experiments were carried out for the determination of the basic neutron-physics characteristics of the initial core at the full-scale critical facility under conditions practically completely corresponding to the real ones. The experimental data analysis has shown that the basic neutron-physics characteristics of the AST-500 core determining the nuclear safety correspond to the design specifications and satisfy the regulatory requirements for nuclear safety.

In the phase of reactor plant basic design development, investigations of the reactor unit seismic stability have been performed using a 1:4 model. The tests confirmed the reactor seismic stability up to a magnitude 8 earthquake (to MSK-64 scale).

All reactor-related equipment items have been tested comprehensively during their acceptance tests before their delivery to the construction site.

Resulting from those extensive R&D activities the safety limits and margins have been comprehensively investigated, particularly for: critical heat flux, multichannel hydraulic instability, single channel flow stability, xenon instability etc. The design margins were proven adequate. A considerable empirical data base has been gained for AST-500 design validation and optimization.

4. CONCLUSION

Resulting from the NHR activities in the RF, the basic design of the AST-500 has been developed, comprehensively tested and mastered in manufacturing. Appropriate construction and installation technologies were proven on an industrial scale with the two pilot NDHPs in Gorky and Voronezh.

The IAEA Pre-OSART mission to the Gorky NDHP has confirmed the sound basis of the plant design and its enhanced safety. That conclusion covered the AST-500 design basis, specific design solutions, their experimental validation, as well as the reactor-related equipment manufacture, erection and construction quality.

Significant upgrading potential of the reference design approaches was revealed in the course of subsequent optimization and R&D activities, which allowed to recently develop the advanced NHR AST-500M with better economics and safety. On this basis a whole series of safe and reliable nuclear reactors of a wide unit power range can be developed for application in various district heating and co-generation systems.

It is noteworthy that the NHR development has played a significant role in forming an advanced safety concept, and design approaches have been implemented for the development of a new generation of enhanced safety nuclear power reactors.

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TRENDS IN SAFETY OBJECTIVES FOR NUCLEAR DISTRICT HEATING PLANTS

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Abstract

Safety objectives for dedicated nuclear heating plants are strongly influenced on the one hand by what is accepted for electricity nuclear stations, and on the other hand by the requirement that for economical reasons heating reactors have to be located close to population centers. The paper discusses the related trends and comes to the conclusion that on account of the specific technical characteristics of nuclear heating plants adequate safety can be provided even for highly populated sites.

1. NHR SITING CLOSE OF POPULATION CENTERS

From an ecological and energy policy standpoint, smaller nuclear facilities to provide heat for district heating are quite attractive. With respect to the demand and the associated heat losses in the hot water lines, these district heating reactors have to be located close to population centers. One of the important criteria to evaluate a nuclear facility is safety. Safety is here meant as the measure of protection of the population against the consequences of accidents occurring at the nuclear facility. Therefore in most countries a major protective goal is remote siting of reactors away from highly populated areas. There are no signs of a relaxation in this restrictive siting practice in many countries.

Accordingly, no special safety requirement for nuclear facilities close to population centers has yet been formulated by any agency, except for the Swiss safety authorities. The valid licensing regulations are generally concerned with normal operation and design basis accidents. However for reactors to be built in populated areas, where a large number of people could be affected by the consequences of severe accidents with very little time for countermeasures, such events have to be included in the licensing procedure, as this is the actual trend for all future reactors.

The probability of occurrence of a severe accident in any given reactor is very small. However, due to the potentially large number of smaller district heating reactors deployed, the cumulative probability of occurrence is large enough to also require for this reason that the severe accidents be included in the licensing procedure.

Distinctive DHR features in respect to safety

The types of risks of a district heating reactor are the same as for conventional nuclear power plants, but the associated consequences are distinctly different. Most consequences are significantly reduced, but a few are enhanced. The severity of the consequences of an accident is influenced by the smaller power size and the operating mode, but also by the proximity of its location to population centers.

The fundamental difference between (large) electricity generating reactors and NHRs is dictated by the application:

- The production of hot water is in the temperature range of 100-150°C. The corresponding pressures to contain the water are up to 20 bars. These pressures and temperatures are well below those of electricity generation plants.

- Because of the temperature and pressure losses and the high installation costs of hot water pipe lines, the heat source, the NHR, has to be located close to the consumer, implying that it be close to the population centers.

- Because of the limited hot water distribution capabilities, the amount of heat requirement is limited. Whereas electricity generation plants have capacities up to 4000 MW_{th}, NHR are much smaller; their capacity seldom exceeds a few hundred MW_{th}.

- The purpose of the NHR is to supply heat, in the form of hot water, through a large piping network to many households and industrial services. There is a danger that in case of a leak due to an accident, radioactively contaminated water could be distributed widely through the heating pipelines.

Consequences for siting

These difference in characteristics compared to the electricity generating plant, namely power, temperature and pressure level, hot water pipes and the siting, have significant implications on the safety.

- The smaller power level leads to a much smaller radioactivity-content (source term) in the reactor. However this content (millions of Curies) is still very much larger than what could be released to the environment without significant harm (a few Curies).

- The strongest implication of a NHR siting close to population centers is the very short time span available after an accidental release of radioactivity until the population would be affected. It is inconceivable that effective evacuation measures could be implemented in sufficient time after the moment of release. Some sheltering might be possible.

- A radioactivity release in a populated area without the implementation of protection measures would likely lead to a significant number of early fatalities. As Kroeger [1] has shown the number of early fatalities is about proportional to the population density, which for an urban area can be 10 to 100 times larger than for the traditional sites of power plants near areas of low population density. Because of the actually conservatively assumed linear dose/consequence relationship the number of late fatalities does not depend on the siting.

- In order to prevent a potential contamination of all heated buildings in an agglomeration, a leak from the primary reactor coolant through the intermediate circuit to the heating network has to be excluded under all circumstances.

- The considerably lower temperatures and pressures in a NHR than in an electricity producing plant can represent significant safety advantages; on the one hand the materials of the vessels and components have large margins to their temperature and pressure dependent failure-limits, and on the other hand the stored energy in the components and coolant is much smaller than in a large plant, and therefore more easily manageable in case of a accident.

- Because of the smaller power level the amount of cooling water compared to the heat source is in NHR significantly larger. This larger heat capacity represents a larger grace period for interventions in case of a malfunction.

These described special features of the NHR with their safety implication indicate that the safety objectives and criteria in use for electricity producing reactors do not take into account, at least not sufficiently, the specific advantages and disadvantages of the NHR. Since the safety objectives of nuclear power plants are documented in a number of guidelines, which have to be of course the bases to which any extra guidelines for NHR's would be added. Some of the more general trends are therefore described in the following.

2. ESTABLISHED INTERNATIONAL GUIDELINES

Since a NHR is a facility for the peaceful application of nuclear energy, it falls under the internationally accepted guidelines, both formal as well as informal ones, that aim at assuring that for achieving safety the best of what existing and evolving technology has to offer is implemented. The top level among the formal guidelines is presented in the recent IAEA publication: "The Safety of Nuclear Installations", issued in 1993 [2]. Other guidelines with a more formal character are:

- "Basic Safety Principles for Nuclear Power Plants", INSAG-3 published in 1988 [3] and its "companion" document "The Safety of Nuclear Power", INSAG-5, published in 1992 [4].
- IAEA "Code on the Safety of Plant Design", 1988 [5], Safety Series No. 50-C-D (Rev. 1), 1988 [5].

Of a less formal nature, but of significance nevertheless on account of the underlying high degree of peer-consensus, are such IAEA-documents as:

- "Safety of Nuclear Installations : Future Direction", 1990 [6], and
- "Objectives for the Development of Advanced Nuclear Plants", 1992 [7],
- Also relevant to the DHR is the discussions and results of finding consensus in the preparation of the new TECDOCs concerning safety principles for the design of future nuclear power plants.

An excellent overview of the current trend of thinking is given in the proceedings of the international conference in 1991 [8] on "The Safety of Nuclear Power: Strategy for the Future". This overview is interesting in as much as it also addresses new "wishes" regarding safety that surpass that which is established practice. Space does not allow the extension of this varied survey, in particular with reference to the approaches taken in the countries with all ongoing nuclear power programme.

3. THE ESSENCE OF CURRENT SAFETY OBJECTIVES

Invariably the top level safety documents stress two parallel, equally important safety objectives as is exemplified by the subsequent quotes from IAEA's Safety Fundamentals, and which are also in line with what is spelled out in INSAG-3:

Radiation Protection Objective: To ensure that in all operational states radiation exposure within the installation or due to any planned release of radioactive material from the installation is kept below prescribed limits and as low as reasonably achievable, and to ensure mitigation of the radiological consequences of any accidents.

Technical Safety Objective: To take all reasonably practicable measures to prevent accidents in nuclear installations and to mitigate their consequences should they occur; to ensure with a high level of confidence that, for all possible accidents taken into account in the design of the installation, including those of very low probability, any radiological consequences would be minor and below prescribed limits; and to ensure that the likelihood of accidents with serious radiological consequences is extremely low.

As this paper deals with plant design, only the technical safety objective will be addressed further below. In practical terms the implementation of this objective is based on the widely accepted principle of defense-in-depth.

The essence of defense-in-depth is the fundamental position that equipment can fail and that operators can err. Therefore, successive layers of protection have to be provided in a reactor plant to prevent so-called initiating events from escalating into accidents with potentially serious consequences, e.g. a large release of fission products from the reactor fuel. This concept is essentially what experts call deterministic and conservative, since for the selection of the protective features failure probability arguments play only a limited role, and since operating conditions are maintained at levels below values determined by large margins below potential failure limits.

The layers of protection can be physical barriers, like the sealed metal tubes that contain the nuclear fuel, or backup functions, like replenishing coolant a for compensating a leakage, or mechanical and electronic interlocks for preventing the operator from giving commands that could lead the plant into a forbidden operating mode, e.g. requesting the reactor fuel to produce more heat than the coolant can safety carry away.

Safety, then, is the result of accomplishing by technical means the following two essential tasks: First, to design, build, operate and maintain the reactor plant in such a way that neither equipment failures, nor operator errors, nor external events such as earthquakes, can lead to overheating of the nuclear fuel and, as a consequence, to a subsequent release of dangerous amounts of radioactivity to the reactor cooling system. Second, to provide and maintain a strong and leak tight containment shell around the reactor cooling system in order to retain the bulk of radioactivity which might be released in an accident sequence that is not terminated within the reactor plant itself, as it should be in compliance with the first task. Defense-in-depth is not a new principle. It has been around in many countries, in fact, since the beginning of the use of nuclear power for generating electricity. All its important elements were already contained, in different words though, in a paper entitled "The Safety of Nuclear Reactors" presented at the first Geneva Conference in 1955 by the then chairman of the US-ACRS, C.R. McCullogh [8]. In response to a question from the audience he summarized the ACRS position as follows:

> "... a policy ... which attempts to balance the four factors: first, careful design of the machine in all of its parts; second, adequate administrative control; third, containment and fourth, location. I think you will agree on reflection that location and containment can balance one with the other...."

This fundamental principle has served the nuclear industry very well and will thus also have to be applied to the design of nuclear district heating reactors.

4. EVOLVING SAFETY OBJECTIVES FOR FUTURE POWER PLANTS

Even though the majority of currently operating plants have an impressive safety record, and can thus be kept in operation without undue risk to the general public, efforts are being undertaken worldwide to enhance the safety of plants even further.

These efforts at enhancement are prompted by several factors. First is the well-known tendency of any industrial activity to improve itself in subsequent generations. Specifically this results in the desire to incorporate lessons learned from the many reactor-years of operating experience to date and to address additional safety issues identified through R+D, design and other analysis, such as probabilistic safety analysis. Second is the desire to maintain the current low level of risk to the public even in the event that nuclear capacity is greatly enlarged in the future. Third is the desire to limit the likelihood and consequences of severe accidents in future plants so as to minimize the potential for large off-site radiological consequences. This is a natural result of the desire for a nuclear power plant to be a "good neighbour", regardless of where it is located, by minimizing the potential for any disruptive effects on surrounding people or their environment.

Despite the fact that no consensus has been reached on the wording of this new document, the safety objectives for future plants will be based on the objectives and principles from INSAG-3 and will include in addition objectives on limited radiological impacts for severe accidents.

5. APPLICATION TO NHRs

It has been pointed out before that there are fundamental differences between (large) electricity generating reactors and NHR's, the most important one being the proximity of NHRs to population centers. This in turn means that evacuation as a potential last layer of protection within defense-in-depth is not a feasible option for providing adequate public protection. The consequences of even a severe accident must then be limited to the site of the NHR itself and this limitation must be provided by proper technical means.

While this specific technical safety objective is, as was mentioned, also under intensive discussion for the advanced nuclear plants for electricity generation (see e.g. the respective section in TECDOC-682 [7]) it would appear, for the reasons previously outlined, before, that it will have to be considered a requirement for heating reactors. Even though there is little precedent in terms of actual licensing, there is reason to anticipate such licensing in the future.

It might be worth mentioning that the Swiss Safety Authorities imposed for NHR one additional objective (above those for nuclear power plant) namely: The reactor system must be designed such that a large release of radioactive material into the environment is impossible under all credible circumstances. In the (not yet approved) safety objectives for future reactors the INSAG-3 objectives were expanded to say: to ensure that the consequences of all realistically conceivable severe accidents have no significant offsite radiological impact.

In practical terms it follows that so-called severe accidents have to already be addressed at the plant design stage. This is in principle not different from the objectives for the advanced power plants, except that the potential severe accident sequences for a NHR may be different. An approach that says, that since a NHR is a light water reactor, like many NPPs, therefore one should to look at a low pressure core melt scenario like that considered for LWR NPPs would be too simplistic on account of the technical differences of NHR. Rather, it is necessary to clearly identify potential sequences taking into account the aforementioned differences.

This would entail a two-tier approach based on the deterministically selected design base which is extended by additional failure postulates to find out whether there are sequences that could lead to an unacceptably large release. Then either such extended sequences have to be designed out (prevention), or additional mitigative features have to be provided.

An additional objective for NHR's has to be that the contamination of the district heating system during normal operation, or during any events, must be prevented.

6. CONCLUSION

The task at hand for this meeting is to scrutinize the design as it evolves and to make sure that the appropriate measures are indeed implemented. From the NHR concepts known it can be expected that the challenge can be met. This confidence results largely from the fact that the special, application-related, features of the NHR (low pressure, high specific water inventory) ease the technical task of assuring safety in comparison with the big reactors, which provide essential and extremely useful background for the safety design of the heating reactor.

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SAFETY OBJECTIVES AND DESIGN CRITERIA FOR THE NHR-200

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Abstract

The construction of a nuclear district heating reactor (NHR) demonstration plant with a thermal power of 200 MW has been decided for the northeast of China. To facilitate the design and licensability a set of design criteria were developed for the NHR, based on existing general criteria for NPP but amended with regard to the unique features of NHR-200. Some key points are discussed in this paper.

1. GENERAL SAFETY REQUIREMENT

For a nuclear district heating reactor (NHR) it is necessary to locate it near the user due to the necessary way of heat transport (hot water or low pressure steam). This means that a NHR is surrounded by a populous area. Using evacuation as an essential element in the ultimate protection of the public can thus become impractical. The safety for a NHR has to be ensured with its excellent inherent features and passive safety. In all credible accidents the radioactive release from a heating reactor has to be reduced to such low levels that off-site emergency actions, including sheltering, evacuation, relocation and field decontamination will be not necessary. In the Technical Report on "The Design of Nuclear Heating Plant" [1] issued by the Chinese National Nuclear Safety Administration (NNSA) it is stated explicitly that no off-site emergency actions such as sheltering, evacuation etc. are allowed for a NHR. In other words, the maximum accident should be no more serious than a level 4 event by the International Nuclear Event Scale. For a typical site in northern China a maximum individual dose of 5 mSv will result from an activity release of 4.7×10^{13} Bq I-131 equivalent via a stack of 50m height. It is indicated that a release of 3.7×10^{13} Bq I-131 can be the limitation for the maximum credible accident.

It is a fact that the existing reactors have become much safer due to various measures, with backfitting and upgrading, especially learning from TMI and Chernobyl accidents. Also, there is a trend that future plants will be, or will have to be, better in terms of CDF than the best of the existing ones, to be achieved by evolutionary design or/and innovative improvements. For convenience, the following figures of CDF can give an idea of what safety has been achieved and of what is the target for the next generation of NPP.

	CDF (1/reactor•year)
The best of the existing NPP	10^{-4}-10^{-5}
The NPP coming on-line by the year 2000 or later	10^{-5}-10^{-6}
The innovative designs	$<10^{-7}$

For a NHR, the safety requirement in terms of CDF has reached the top of the safety target if a figure of much less than 10^{-7} of CDF is achieved.

On the other hand, there is a serious challenge to the economy for a NHR. The capacity of a NHR can not be as big as that of NPP due to the limitations of heat transport. The economic thermal power is in the rang of 200-500MWt. Moreover, the load factor is also much lower than that of a normal NPP.

It is obvious that to meet the safety requirements, and to lower the capital investment are the major concerns in the design of a NHR. The only solution is to have a design with inherent safety characteristics and passive safety as much as possible instead of the complex engineering safety features. In addition, the high end-user efficiency of district heat application instead of electricity generation with low end-user efficiency is also a key point to improve the economy for a NHR.

2. DESIGN CRITERIA

The design criteria for conventional nuclear power plants already exist. Most of them are suitable for the NHR, but some of them have to be modified due to more stringent safety requirements and unusual design approaches. In order to have a design basis and to enable the safety regulatory body (NNSA) to evaluate systematically the design of the NHR-200, a set of design criteria for NHR is being drawn up and reviewed by a team organized by NNSA. Since no large scale operational experience is available at this time, the design criteria are not a complete nor a official set of regulations. It will be issued as a technical document. Some of the major points of this criteria are discussed as follows.

(1) Operation categories

Usually 4 operation categories are classified for a conventional NPP. Among them Category III and IV are accident conditions. The accidents of Category IV are the most serious ones in the sense of DBA. But in recent years addressing beyond design basic accidents has been required by various regulatory bodies. The French H procedures are perhaps the most comprehensive ones.

In order to meet the enhanced safety requirements as well as the position of the Chinese regulatory body, an operation Category V is added beyond the 4 categories in our design criteria. Generally, the operation Category V is such an accident condition after a DBA (Category III or IV). Apart from the assumption of a single failure, there is an additional failure assumed to occur. Therefore, Category V consists of events with lower frequency than those for Category IV. The typical events are: loss of off-site power followed by an assumed failure to scram combined with a stuck open safety valve; break of reactor vessel in its lower part or pipe break of coolant purification system followed by failure of isolation due to failure of two isolation valves; intermediate loop break followed by failure of isolation. The deterministic analysis is conducted with realistic parameters. The measures for mitigation of such accidents should be reliable. But a grace period can be taken credit of. The acceptance criteria of this Category is that the release of radioactive substance into the environment is not allowed to disrupt normal life beyond the non-residential area (the plant).

The limiting doses for individuals of the population during each operational Category are much less than that for a conventional NPP. They are listed as follows:

- Category I (Normal operation) 0.1 mSv/a
- Category II (Anticipative operation events) 0.2 mSv/event
- Category III (Rare accident) 1.0 mSv/event
- Category IV (Limited accident) 5.0 mSv/event
- Category V (Additional Operation Category) 5.0 mSv/event

(2) Thermohydraulic design criteria

In order to reduce the radioactive release from the fuel elements in case of an accident, the thermohydraulic design criteria for a NHR are more rigorous than those for conventional NPPs.

a. In respect to fuel element damage the differences between NPP and NHR are listed below:

	NPP	NHR
Category I and II	No additional fuel damage	No additional fuel damage
Category III	Fuel damage should be limited in a small part of all fuel elements	No additional fuel damage
Category IV	Fuel damage possibly occurs with a large amount of all fuel elements	Fuel damage should be limited to a small part of all fuel elements
Category V	NA	Small as above

b. Correspondingly the DNBR must stay above the limit value in operation Category I and II for conventional NPP, but also in Category III for the NHR. The same requirement for fuel temperature is that the maximum temperature at the center of the fuel element at the hot spot never reach the melting temperature in the operation Category I and II for conventional NPPs, but also in Category III for the NHR.

c. For conventional NPPs, the average temperature of the fuel clad at the hot spot has to be below the embittlement temperature of 1204°C in case of a LOCA. But for the NHR it is required that the reactor core is always covered by coolant in case of LOCAs. Thus the temperature will be far less than the above limit.

(3) Containment

As a final barrier against fission product release, a containment system is one of the important Engineered Safety Features (ESF) in a current nuclear power plant. It consists of a containment structure and several systems to maintain the integrity of contaminant during accident scenarios. This system is very expensive. Recently, along with the development of advanced nuclear power plant, especially for more innovative reactor designs, the concept of a containment is also getting development. For example, a vented confinement concept [2] is provided for a small or middle size modular high temperature gas cooled reactor instead of the gas-tight pressurized containment for the current generation LWRs due to the exclusion of the possibility of a fission product release from coated particle fuel elements in case of an accident. Another example, for the Safe Integral Reactor (SIR) developed by the UK and the USA [3], the integrated arrangement of the primary coolant system makes it possible that the containment is a compact one.

Based upon the definition given by 10CFR50, the primary reactor containment means the structure or vessel that encloses the components of the reactor coolant pressure boundary, and serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment. For the systems connected to the reactor vessel, the reactor coolant pressure boundary is up to and including the second isolating valve. Since the NHR-200 is an integrated arrangement of the primary coolant system, a small compact containment is adopted which meets the above definition. Moreover, this containment has the further important function that it ensures the reactor core being always covered by coolant in all pipe break accidents, even in the case of a small break in the lower part of the reactor vessel.

The reactor building serves as a secondary confinement. The main function of this structure is the protection against external events. It also provides a subatmospheric enclosure to collect the leakage from the compact containment during LOCA.

(4) Special credit for NHR-200

Since the safety systems adopted for the NHR-200 have many differences compared with those in normal NPPs, these differences have to be reflected regarding the requirements for the support systems. Some special credits are discussed as follows.

a) Emergency power system

For a NPP, apart from two independent off-site power supply connections, there is an emergency power system with two or three separate trains equipped with a quick-starting diesel unit for each to supply power to all redundant safety-related systems, such as ECCS, containment cooling and spray system, residual heat removal system and the related support systems in case of loss of off-site power. But for a NHR some safety-related systems, such as ECCS, containment cooling and spray systems are not necessary, and some safety-related systems, such as the shut down and residual heat removal systems are passive systems. Therefore, the emergency diesel generators are no more necessary. Nevertheless, there are two diesel generators in the design of NHR-200, but they are not classified as safety related. In order to enhance the reliability of stand-by power supply, one of these two diesel generators is classified as seismic Category 1.

b) Component cooling water (CCW)

In the design of NHR-200 the loads of CCW are cooling of coolant purification system, condenser of liquid waste treatment system, and cooling of the control rod drive system. Among them there are no safety-related systems, therefore, the CCW is classified as a non-safety related.

c) Heating, ventilation and air conditioning system (HVAC) for the control room

Since a large release of radioactive material into the reactor building is impossible under all credible circumstances, the HVAC is not classified as safety related.

3. SPECIAL RADIOLOGICAL PROTECTION ISSUES FOR NHR

Since a NHR has to be located near the user, some radiological protection issues which are different from the case of NPPs have to be investigated. They are discussed as follows.

1) Site criteria

The features of current site criteria for NPP shown by the American code 10 CFR100 are as following:

1) The evaluation is based on an assumption of a core melt accident.
2) In case of the maximum hypothetical accident, emergency actions, including sheltering, evacuation and field decontamination, are adopted in order to assure that no individual receives a whole body dose in excess of 0.25Sv which would result in acute injury. Also, the accumulative dose received by the population is limited to a reasonable value.
3) An exclusion area and a low population zone are necessary. Also, a distance from the reactor to population center has to be kept in order to meet the above requirement.

More than 6000 reactor-years of NPP operating experience with only two significant accident shows that the above principles are correct. NPPs are quite safe but core melt with subsequent release of appreciable quantities of fission products is still considered credible (The frequency of core melt is considered as 10^{-5} per reactor year) [4]. Preparation of an off-site emergency action including evacuation is necessary. A site for NPP should be far from population center, saying larger than 25 km.

For the NHR it is impossible to require a large area adjacent to the plant with sparse population or a site far from a city. It means that distance is not a protecting factor any longer. The actions of evacuation, relocation and field decontamination are no longer practical emergency measures to protect the population from over-exposure. The public is protected only by the safety features of the NHR. Safety is achieved by adopting more inherent safety features, they have been presented in many papers [5, 6]. The frequency of core melt for a NHR is much less than 10^{-7} per reactor-year [6, 7] which can be considered negligible in practice. In this way core melt is no longer considered as a design basis. The site criteria for NPPs can not be used in case of the NHR. For a NHR the recommended dose limit for the maximum design basic accident is 5 mSv without any emergency action.

Regarding high population, high utilization factor of land and the unit power of 200-500MW for one NHR, which is one magnitude smaller than that for a NPP, a non-residential zone of 250m in radius and a physical isolation zone of 2km in radius are proposed. During the lifetime of the NHR, development in the physical isolation zone should be restricted in terms of population and large scale public facilities. This physical isolation zone is only functioning as an isolation between the plant and the public to reduce the interference with each other during normal operation as well as during abnormal conditions.

2) Liquid effluents

For a NHR site it is better to have no restriction on liquid effluent release. In most cases, there is no suitable receiver of liquid effluent near a proposed site for a NHR.

Therefore, the principle of treatment and disposal of liquid waste for a NHR should be different from that for a NPP. For a NPP the distinguishing features are: large amounts of liquid waste (more than 10,000 m^3/a), a moderate degree of decontamination (depends on the amount of salt content, evaporation or demineralizing approaches that are used respectively; the target of decontamination is 10^{-8}-10^{-7}Ci/l). After treatment, the liquid effluent is mixed with circulating cooling water and released to a river or sea.

For a NHR the amount of liquid waste is much less than that for a NPP (300 m^3/a is expected). The proposed principle is increasing the decontamination factor (the target of concentration after treatment is 10^{-10}Ci/l), and then reusing the decontaminated water as much as possible. For the remains of usage it can be used as a make-up of plant cooling water or evaporated in a natural evaporative pond, even drained to a city sewage network.

3) Protection against pollution of the heating grid

Differing with a nuclear power plant, the NHR will be connected with the user through the heating grid. Therefore, protection against pollution of the heating grid is extremely important. In the design of the NHR, an intermediate loop is adopted to separate the heating grid from the radioactive primary loop. Moreover, the pressure of the intermediate loop is kept higher than that of the primary loop in all conditions, so that it ensures that leakages, if there are any, are always from the intermediate loop to the primary loop, never vice versa. In addition, pressure and radioactivity of the intermediate loop are monitored continuously. An isolating device is also installed in order to quickly isolate the intermediate loop from the primary loop in case of occurrence of a large leakage. These measures ensure that the contamination of the intermediate loop is very low. The pressure of the intermediate loop is also kept higher than that of the heating grid. This arrangement not only makes heating grid operation easier but also favors keeping the water quality of the intermediate loop.

The limits of radioactive concentrate are: 10 Bq/l for the intermediate loop; 0.37 Bq/l for the heating grid.

4. SUMMARY

Most of the design criteria for conventional NPPs are suitable for the NHR, but some of them have to be modified due to more stringent safety requirements and due to the unusual design approaches. A set of design criteria for the NHR has been drawn up for facilitating the development of NHRs in China. Meeting these criteria means that off-site emergency plans for protecting the population would not be necessary.

The evaluation of the design of the NHR-200 which meets the proposed design criteria shows that the design of NHR-200 has attractiveness both in safety characteristics and economy. The proposed design criteria have to be updated along with the accumulation of practical experience with NHRs.

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CHARACTERISTIC THERMAL-HYDRAULIC PROBLEMS IN NHRs: OVERVIEW OF EXPERIMENTAL INVESTIGATIONS AND COMPUTER CODES

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Abstract

The paper briefly reviews the specific thermal-hydraulic problems for AST-type NHRs, the experimental investigations that have been carried out in the RF, and the design procedures and computer codes used for AST-500 thermohydraulic characteristics and safety validation.

1. INTRODUCTION

In the course of the AST-500 design development a large scope of experimental investigations on reactor plant thermal-hydraulics has been carried out to select and verify the design decisions and the parameters, and to validate safety [1,2,3]. A complex of test facilities has been created, including electrically heated models of the reactor with the guard vessel, and simulators of various scale for in-depth study of reactor thermal-hydraulics, both under normal and emergency conditions. Table 1 gives the characteristics of the main experimental facilities.

The peculiarities of thermohydraulics in NHRs under normal operation and emergency conditions are associated with natural coolant circulation, reduced parameters of the plant and with specific features of the integral reactor, such as availability of built-in steam-gas pressurizer (SGP), in-reactor heat exchangers for the emergency residual heat removal, and a guard vessel (GV). Proceeding from these factors the main attention in performing the experimental work was paid to studying the following effects and phenomena:

- natural circulation flow stability;
- ultimate values for core heat loads;
- thermohydraulics of the built-in steam-gas pressurizer;
- behavior of non-condensable gases (gas transfer, gas distribution, dissolved gas in water);
- natural circulation modes under loss of coolant conditions;
- operation of the emergency heat removal HXs at steam condensation from a steam-gas mixture;
- reactor-guard vessel system behavior at LOCAs;
- passive safety systems initiation and functioning, and others.

Together with the tasks of studying the thermohydraulic processes running in the reactor plant under normal operation and emergency conditions, and validation of the operability and efficiency of the safety systems provided, obtaining the representative data for the verification of computer codes and calculation methods represents the most important objective for the experiments.

Table 1: Experimental facilities used for AST-500 thermal-hydraulics investigation

Test rig	Circul-ation	Max. power, MW	Max. pressure, MPa	Volume, m^3	Scaling Volume	Scaling Height
L-186 Studying of CHF	FC	2	18.0	19 rods 13.6 mm		1:1
37-rod bundle Partial core uncovery	NC	0.1	5.0	37 rods, 10mm, $T^{max}_{clad.} = 700°C$		1:1
L-800 Large-scale models of steam-gas pressurizer	FC	0.7	18	2	1:14	1:2
KMR-2 AST-500 reactor large scale model	NC	2.5	7.0	1.3	1:170	1:1
1385-MONOBLOCK Integral reactor model	NC	1.8	18	0.3	1:700	1:2

Abbreviations: FC - forced circulation, NC - natural convection.

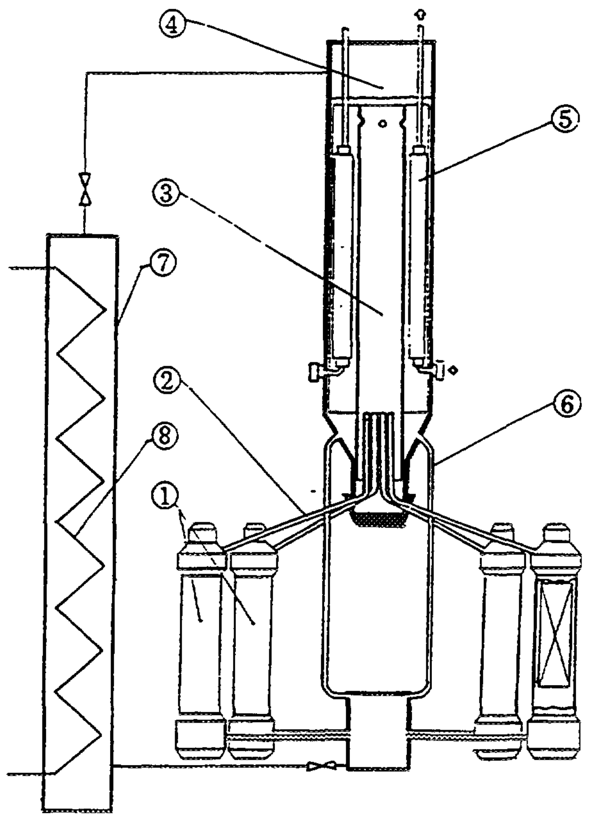
The paper reviews the experimental investigations that have been performed for the AST-500 reactor plant thermal-hydraulic design as concerns the aforementioned problems. The characteristics of the computer codes for the analysis of AST thermal-hydraulics which have been validated experimentally are presented as well. The investigation results on the AST static thermal-hydraulic characteristics are presented in [3].

2. NATURAL CIRCULATION FLOW STABILITY INVESTIGATION

Investigations of AST-500 primary circuit thermohydraulic characteristics, including the problem of natural circulation flow stability, were performed on the thermophysical test facilities 1385 and KMR-2 [9,10]. These facilities represent electrically heated models of the AST-500 integral reactor, different in scale, with coolant natural circulation and built-in steam-gas pressurizer, containing thus all main components of the AST circulation circuit.

The 1385 facility (Fig.1) includes a four-assembly heating zone. The modelling scale in respect to natural circulation circuit height is 1:2. Maximum power is 1.8 MW.

The KMR-2 facility (Fig.2) is a large-scale model of the AST-500 circuit, the height of which is close to the real one. The scaling factor for the volume and the flow areas in the circuit components was 1:170. Maximum power is 2.5 MW. The core simulator consist of two 37-rod heater assemblies with practically full-scale fuel rod simulators.



1.- Heating columns
2.- Chimney tubes
3.- Riser
4.- Steam-gas pressurizer
5.- Heat exchanger
6.- Downcomer
7.- Guard vessel simulator
8.- GV heat exchanger

Fig.1. 1385 test facility. Model of integral reactor

The influence of circuit pressure, coolant subcooling and steam content at the fuel assemblies outlet, coolant subcooling at the fuel assemblies inlet, inlet hydraulic resistance, fuel assemblies power non-identity upon parallel-channel and whole-circuit flowrate oscillations were studied. Flow instability zone boundaries versus coolant subcooling value at the outlet, or outlet steam quality, and coolant inlet temperature were obtained. Fig.3 shows the dependence of the whole-circuit flowrate oscillations amplitude versus relative enthalpy at the fuel assembly outlet.

Using the experimental data obtained, the development of computer codes for flow instability analysis based on linear and nonlinear models has been carried out.

Resulting from the calculations and experiments performed, the natural circulation stability boundaries for the AST-500 reactor were determined. Its operation mode parameters were selected assuring whole-circuit coolant circulation stability with a necessary margin. With such an approach the possible parallel-channel flow oscillations in individual fuel assemblies are limited in amplitude and do not reduce the core heat-engineering reliability and equipment operability.

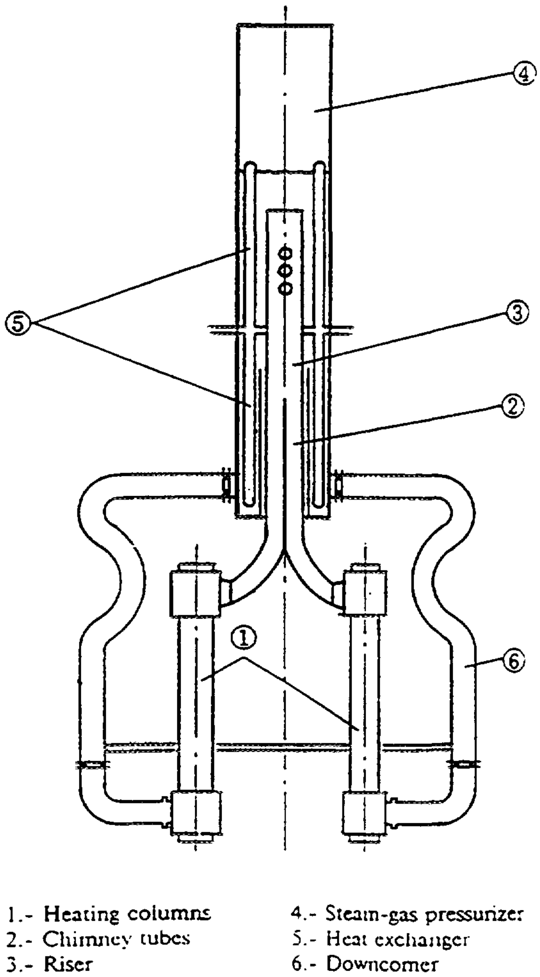


Fig.2. Natural circulation circuit of KMR-2 test facility

An additional potential for enhancement of parallel-channel flow stability is provided by the design margins available in the thermal-hydraulic characteristics of the primary circuit which allow to increase the coolant subcooling to saturation temperature at fuel assembly outlet.

3. BOILING CRISIS INVESTIGATION

To define the ultimate admissible heat loads for the AST-500, experimental investigations of boiling crisis were performed on core fuel assembly models under conditions corresponding to the AST parameter range [7]. The six experimental assemblies used for the investigation represented 7 and 19-rod bundles of electrically heated fuel rod simulators having the actual diameter, length and pitch.

The main part of the experimental data on the boiling crisis were obtained for the following parameter range:

pressure	1.0-6.0 MPa
mass flow velocity	0-1400 kg/(m s)
inlet temperature	130-220°C

The influence of radial and axial power distribution non-uniformity in a rod bundle on the ultimate heat loads for real peaking factors of Kr = 1.22 and Kz=1.8 was studied as well.

On the basis of the obtained data, a correlation for critical power calculations was developed specifically for the AST reactor operating conditions, describing the whole data base with a root mean square error less than 6%. The AST-500 reactor core heat-engineering reliability analysis performed with this correlation showed that the minimum critical power ratio for fuel assemblies at nominal parameters amounts to 2.5.

4. TWO-PHASE FLOW CHARACTERISTICS INVESTIGATION

A knowledge of the two-phase flow parameters in the riser section of the primary coolant flow path is necessary for natural circulation flowrate determination and for hydrodynamic and neutron-physical stability analyses etc.

Non-equilibrium and equilibrium two-phase flow characteristics were investigated using a 7-rod bundle model of the AST-500 fuel assembly, equipped with a simulator of the draught section (chimney) in form of a circular tube under conditions corresponding to the range of AST-500 operational parameters: P=1.5-2.5 MPa, w=320-790 kg/(m s), q=0.3-1 MW/m^2. Void fraction variation along the height of the draught section was determined in experiments using acoustic transducers.

Resulting from the investigations performed, the data on steam condensation lengths in non-equilibrium two-phase flow were obtained, and the influence of flow parameters on condensation length was studied. Dependences of void fraction in the draught section on the value of balance mass steam quality (x) at a fuel assembly model outlet were obtained for x ranging from - 0.01 to 0.05.

The influence of the scale factor on two-phase flow pattern in the riser was also studied. The void fraction profile spectra for steam bubbles velocities and sizes in the model of the reactor draught section of 450 mm diameter and 2 m height in a parameter range: P=0.6-1.1 MPa, w=200 kg(m s), x=-0.3-0.65% described in [8] were obtained experimentally. The statistically average size of the steam bubbles in the experiments was 6 mm. The experiments showed an appreciable non-uniformity of steam bubble distribution over the cross section and a complex pattern of flow. The average values of void fraction in a large diameter draught section were 10-20% lower than corresponding values for small-diameter tubes. This is explained by the non-onedimensional structure of two-phase flow.

5. INVESTIGATION OF STEAM-GAS PRESSURIZER CHARACTERISTICS

The presence of non-condensable gases in integral reactors with steam-gas pressurizer influences both the thermal-hydraulics of the primary circuit and the pressurizer characteristics under steady-state conditions and in transients, as well as in accident sequences. Reactor parameter variation within the operational range of power, the dynamics of the pressurizer, and the efficiency of heat removal from the reactor through in-reactor heat exchangers during LOCAs all depend on the gas quantity in the pressurizer.

During the development of the AST-500 design, calculational-theoretical analysis of this problem and related experimental investigations were performed. In the experiments the build-in steam-gas pressurizer characteristics were studied, as well as processes of gas transfer and gas distribution in the primary circuit. The experiments were carried out on a large-scale model of a pressurizer - the L-800 test rig (Fig.4) and on the reactor models 1385 and KMR-2 rigs (Fig.1,2). Fig.5 shows the obtained loading characteristic of the 1385 test rig pressurizer that represents a dependence of the pressure rise in the pressurizer on gas and gas concentrations in the circuit water versus the volume-average partial gas pressure in the pressurizer.

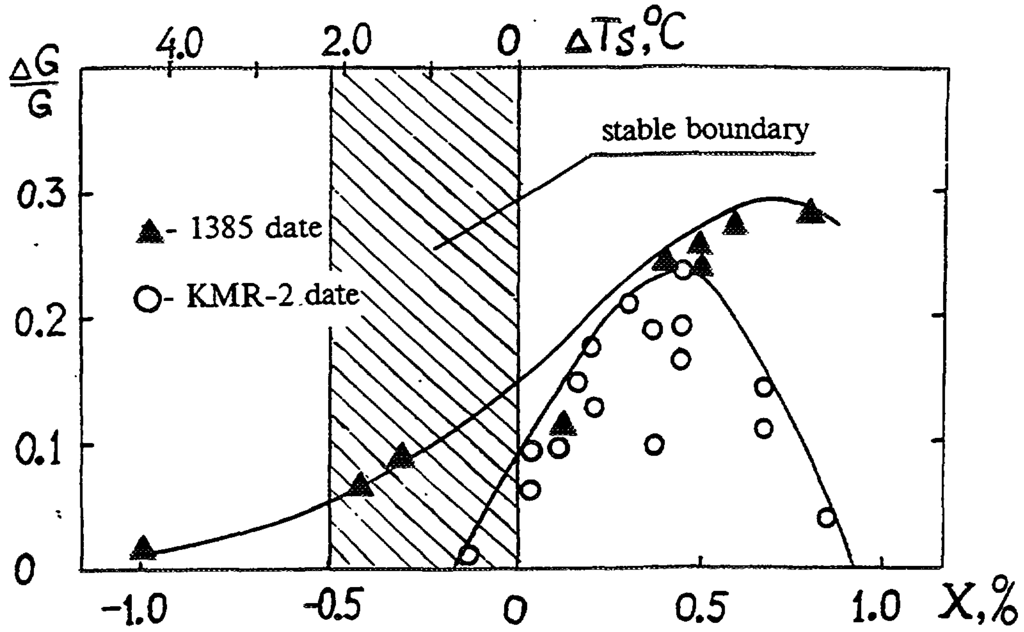


Fig.3. Effect of outlet relative enthalpy on ampl-itude of mass flow oscillation of natural circulation

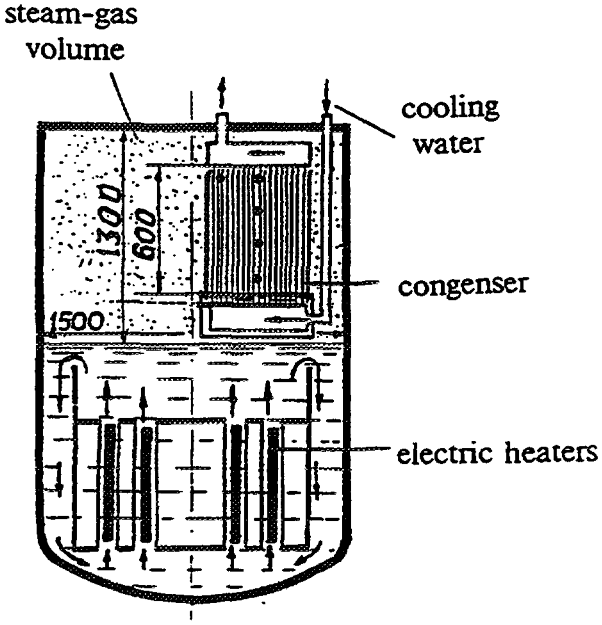


Fig.4. L-800 rig. Large scale model of steam-gas pressurizer with condenser located in steam-gas volume

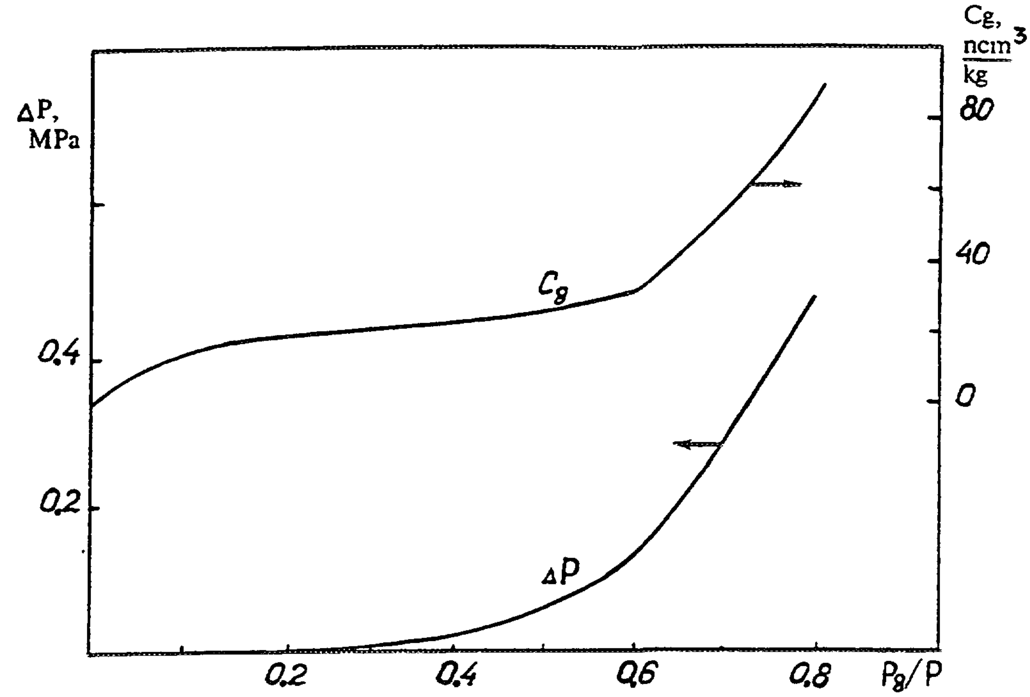


Fig.5 Change of pressurizer pressure and gas concentration in water depending on gas content in steam-gas volume

The procedure for the calculational analysis of the static characteristics of the steam-gas pressurizer and the gas distribution in the primary circuit of an integral reactor were developed and the computer code GARRIC was created. Calculation models of the code were verified based on the results of the experimental investigations.

6. RESIDUAL HEAT REMOVAL EFFICIENCY UNDER EMERGENCY CONDITIONS

Experimental investigations of the intensity of steam condensation from a steam-gas mixture were performed with tube bundle models for LOCA conditions accompanied by emergency residual heat removal heat-exchanger dry-out. The experiments were carried out on the L-800 test rig (Fig.4) using multi-row models of the AST-500 heat-exchanger tube bundle [6]. Investigations of gas influence on the intensity of steam condensation on the tube bundle of the 1385 test rig heat exchanger were performed in a wider range of parameters. Data obtained with the rig L-800 on the influence of gas content in the steam-gas plenum of the rig upon thermal power of the condenser are presented in Fig.6.

The procedure for calculational analysis of heat transfer intensity at steam condensation on a dried-out tube bundle of in-reactor ERHR heat exchangers was developed based on experimental results. The procedure is included in the UROVEN/MB-3 computer code which is used for the analysis of loss-of-coolant accidents in AST reactor plants.

7. RESIDUAL HEAT REMOVAL

The emergency residual heat removal channel with a steam condenser located on the reactor represents a simple and passive secondary circuit independent residual heat removal

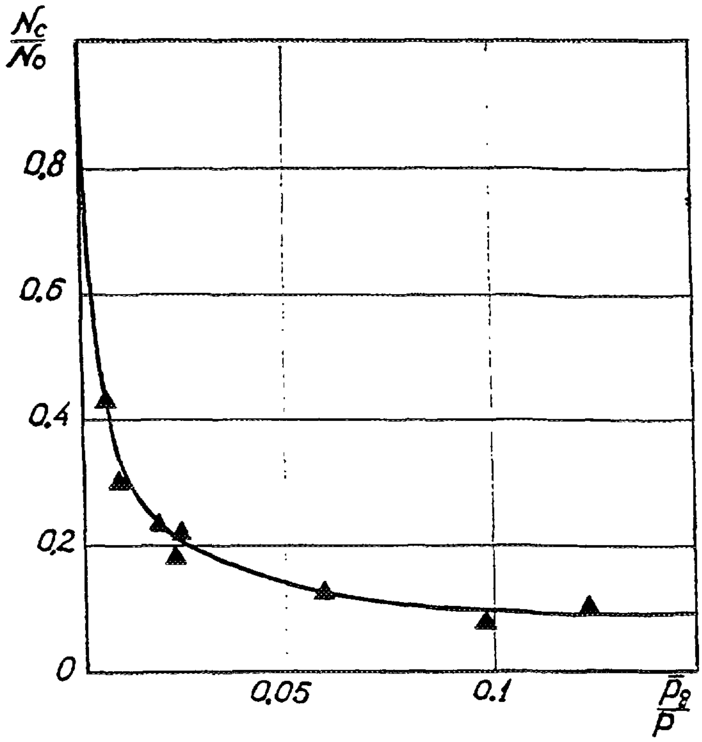


Fig.6. Effect of gas content in pressurizer on relative condenser thermal power

path for utilization in advanced NHRs. The condenser interior communicates with a pressurizer. At normal operation it is filled with a steam-gas mixture having a high content of gas which blocks heat removal from the reactor. The ERHR channel is actuated by gas blowdown from its tubing, followed by the condenser transition to a steam-condensing mode of operation with returning the condensate to the primary circuit.

The channel operation efficiency within the real range of working parameters and thermal loads is corroborated by the results of the experimental investigations carried out on a large-scale model of the condenser in the L-800 test rig [5]. In the course of the experiments, the modes of condenser actuation were investigated, and the influence of pressure, gas content and cooling water parameters on the heat removal intensity was studied. Optimal characteristics of the condenser were determined resulting from the provision of maximum heat removal capacity and the elimination of condensate entrainment into the dump line.

8. EXPERIMENTS WITH FUEL ASSEMBLY MODEL UNCOVERY

Within the framework of experimental validation of the safety required for the AST and other integral PWR-type reactors, investigations were performed on the dynamics of the thermohydraulic characteristics of 37-rod and 7-rod models of fuel assemblies under conditions with partial uncovery of heater rods in cases of small-break LOCAs. Experiments on coolant boil-off from the fuel assembly, on rod bundle dry-out and subsequent slow reflooding (velocity of 0.002-0.01 m/s), as well as experiments on the cooling of completely uncovered rod bundles by steam-gas mixture natural convection were carried out. The studied

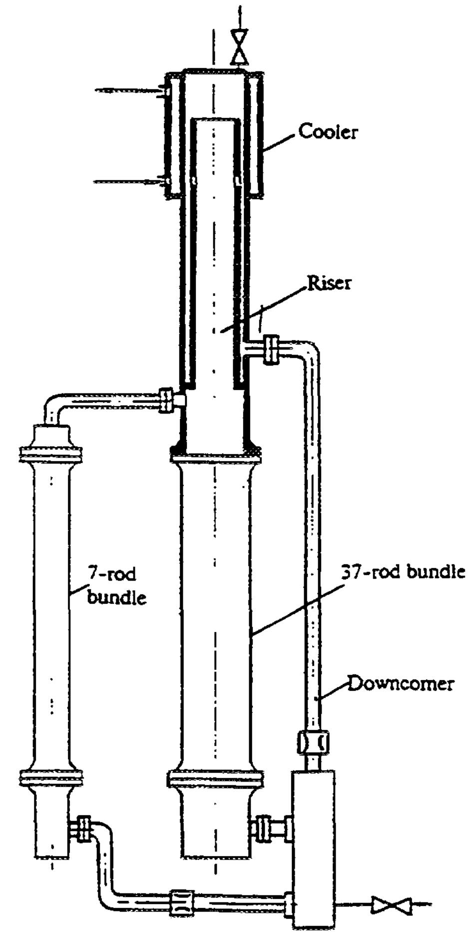


Fig.7. Primary circuit model with two test bundles for investigation of thermohydraulics during core uncovery

range of pressure variables was 0.85-5.0 MPa, that of power 15-40 kW, and that of heat loads non-uniformity over the assemblies 1.0-1.5. The maximum temperature of the fuel rod simulators was 750°C.

The test facility (Fig.7) represents a natural circulation loop consisting of two fuel assemblies (FA) models: a 37 rod bundle (d=10 mm, l=14 mm, H=3.5m) and a 7-rod one (d=14mm, l=18.7 mm, H=3 m) connected in parallel, and of a riser and a downcomer section. There is a cooler in the upper part of the model above the downcomer. The FA models include bundles of rods made of steel tubes which are electrically heated uniformily along the height, and spaced in a triangular lattice. The temperature of the heater rods, the temperatures both of the steam and the vessel metal, and the pressure difference along the height of the sections were measured.

Partial Uncovery of Rod Bundles

In the course of the experiments data on void fraction in rod bundles and on the quench front dynamics were obtained. The cooling of an uncovered part of the bundle by single-phase flow of superheated steam was also studied. The experimental data showed good agreement of a quench front coordinate with a two-phase swell level in the rod bundle under small-break LOCA conditions. The difference between rod temperature rise coordinate and the level position was 0.1-0.2 m which is within the limits of the level determination error in the experiments. This difference may be caused by heater rod cooling near the interphase surface due to entrained droplets. Correlations for the calculation of the intensity of heat transfer to superheated steam in a fuel rod bundle for the laminar and transition range of Re numbers were recommended based on the analysis of experimental data.

The effects of interaction between FA regarding parallel channel and loop oscillations at fuel assembly dry out and reflooding, including gravitation reflooding conditions have been investigated. The influence of heat load non-uniformity over the test assemblies upon the thermal-hydraulic characteristics of a system consisting of two FA models was studied as well.

Cooling of Completely Uncovered Rod Bundle

An investigation of cooling completely uncovered heater assemblies by natural convection of gas and steam-gas mixture was performed for the conditions of severe accidents. The experiments have confirmed that there exists a mechanism of heat transfer from a dried fuel rod bundle heated up to a high temperature (Tclad=600-700°C) by gas natural convection vortexes through a riser to a heat exchanger located in the upper part of the circuit. A considerable intensification of heat transfer at a pressure increase was noted. It should be expected that in the integral reactor with the operation of the built-in ERHR heat exchangers, the cooling of a dried core by natural convection will be more intensive due to the weak interaction of downstream and upstream flows because of the large cross section of the riser. This mechanism of fuel rod cooling should be taken into account in the analysis of accidents with core dry-out.

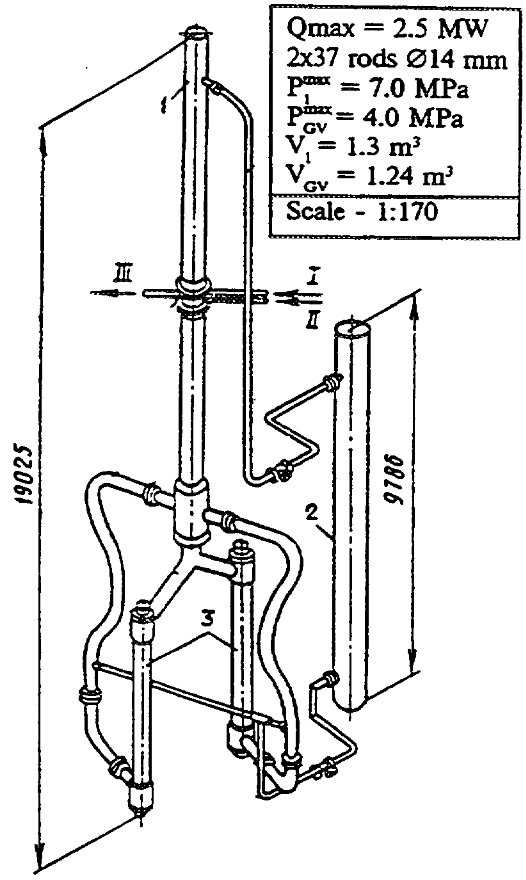
9. EXPERIMENTS AT LARGE-SCALE REACTOR MODELS

To validate the AST-500 safety, a complex of investigations on thermohydraulic processes at loss-of-coolant and loss-of-heat removal accidents was performed at the KMR-2 test facility (Fig.8). The facility includes a large-scale model of the AST-500 integral reactor and its guard vessel simulator serving for confinement of leaks at loss-of-coolant accidents.

Proceeding from the AST design features, the following processes important for a correct description of emergency conditions were studied in the experiments:

- dynamics of the built-in steam-gas pressurizer;
- flow circulation and heat transfer in the primary circuit under loss-of-coolant conditions;
- reactor-guard vessel system thermohydraulic behavior;
- heat-and mass transfer and gas distribution inside the guard vessel [2,3].

The test program included experiments with coolant outflow into the guard vessel model at different sizes and locations of openings imitating leaks in the primary circuit at ruptures both in the pressurizer area and in the reactor vessel bottom, as well as imitation of an unscrammed loss-of-heat removal accident. The range of break sizes studied in the facility was 4-30 mm which corresponds to 50-400 mm for the AST-500 primary circuit. Similar investigations were carried out on the smaller-scale model of an integral reactor in the 1385 test rig (Fig.1) for an expanded range of parameters.



1. Primary circuit model of integral reactor
2. Guard vessel simulator
3. Heating columns

Fig.8. KMR-2 Test facility

Experimental data on the reactor-guard vessel system behavior at primary circuit loss-of-integrity accidents were obtained that showed the efficiency of the AST-type guard vessel as a passive confinement system, providing for keeping the core under coolant level. Normal fuel rod cooling was shown under the "waterfall" and steam-condensate circulation condition, taking place after considerable coolant losses from the primary circuit. Coolant subcooling values and flow circulation mode variations at water outflow from the test rig are shown in Fig.9.

Issues such as non-condensable gas effects, heat-and mass transfer and gas distribution in the guard vessel model, and temperature stratification in a water volume of the guard vessel model were investigated. Data on heat transfer coefficients for steam condensation on the guard vessel walls were obtained for a wide range of steam-air mixture parameters (P=0.4-3.0 MPa; Pg/P=0.18-0.8). They showed a considerable rise of steam condensation intensity at a pressure increase. The experimental data are well described by the calculation model based on the heat and mass transfer analogy using correlations for natural-convective heat transfer on a vertical wall. The known Uchida correlations, used for the description of containment thermal-hydraulics in some Western codes (MARCH 3 and the others) does not take into account the influence of the parameters and underestimates the value of heat transfer coefficients at increased pressures of P > 0.3 MPa.

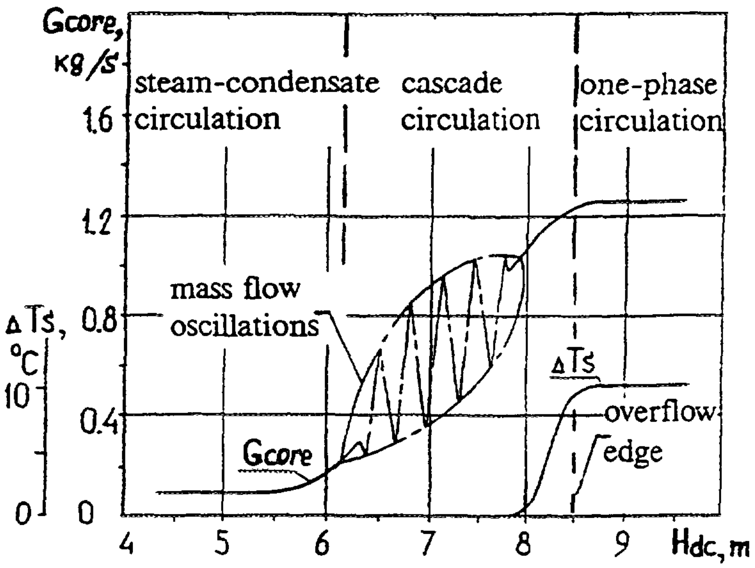


Fig 9. KMR-2. Natural circulation modes during small LOCA depending on water level in downcomer

10. CODES FOR THERMAL-HYDRAULICS ANALYSIS

The complex of codes used at OKBM for thermal-hydraulic calculations of AST-type reactor plants under stationary conditions, for the analysis of natural circulation stability and for transients and accidents is presented in Fig.10.

OKBM-developed Codes

The codes that have been developed at OKBM describe the details of thermal-hydraulic processes in the reactor during normal operation and in emergency situations. They can handle the distinctive design features of the integral reactor, such as built-in pressurizer with a large amount of non-condensable gas, in-reactor heat exchangers, guard vessel, and passive safety systems. The calculation models used in the codes rely upon the results of experimental investigations of thermal-hydraulics conforming to integral reactors of the AST type.

Empirical relationships are used for the calculation of heat and mass transfer, hydraulic resistance, steam slipping and leakage flow rates. The VERESK-M code for the calculation of stationary thermohydraulic characteristics describes non-equilibrium boiling in the core and in the riser section using the models given in [11,12]. In the codes DKAST and UROVEN/MB-3 for the analysis of thermal-hydraulics in transients, the equilibrium model of two-phase flow with steam slipping is used.

The procedures for calculating the steam condensation from a steam-gas mixture on a tube bundle of a heat exchanger are based on the results of experimental investigations on the in-reactor heat exchanger models. An empirical dependence is used for the calculation of heat transfer coefficients for steam condensation on the guard vessel walls. The procedure for the calculation of the steam-gas pressurizer characteristics and for the distribution of gas in the primary circuit have been developed based on the results of the experiments in the large-scale model of pressurizer (L-800 test rig) and in integral reactor models.

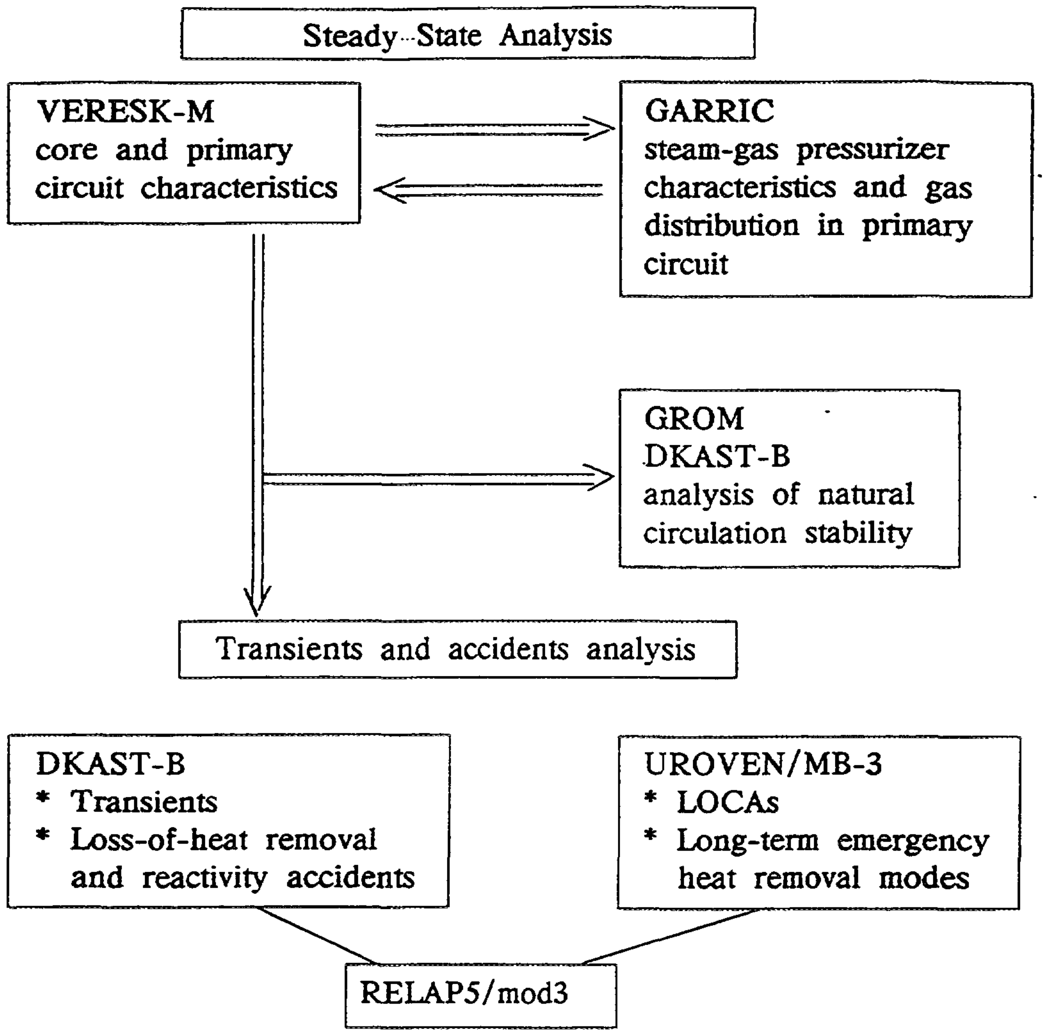


Fig.10 Code package used for thermohydraulic analysis of AST reactor plants

The reliability of calculational prediction of thermal-hydraulic characteristics of the AST under normal and emergency conditions has been confirmed by verification of the codes performed on the basis of the results of experimental investigations of separate effects and integral phenomena, and on integral data obtained from different scale models of the integral reactor in the KMR-2 and 1385 test facilities.

Use of RELAP5/mod3 code

The RELAP5/mod3 code is used in OKBM for alternative analysis of the thermal-hydraulics of AST-type reactors under emergency conditions. A verification study is under way to estimate the code applicability to integral reactors. Calculation analysis of experimental modes realized on the integral reactor models (KMR-2 and 1385 test rigs) has shown a satisfactory accuracy of the plant behavior representation in the RELAP5/mod3 code. The class of transients and accidents corresponding to the range of the RELAP5/mod3 code applicability for AST-type integral reactors has been determined based on the results of verification studies.

11. CONCLUSIONS

Comprehensive experimental investigations of the AST reactor thermal-hydraulic problems for normal and emergency conditions have been performed. These investigations included the study of primary circuit natural circulation characteristics, natural circulation stability issues, boiling crisis, two-phase flow characteristics, steam-gas pressurizer and gas distribution characteristics, intensity of steam condensation from steam-gas mixtures, as well as emergency processes studies using different-scale models of the reactor, etc.

The calculation models and computer codes for the analysis of AST-reactor plant thermal-hydraulics under normal operation and emergency conditions have been developed and validated experimentally. The code verification confirmed the reliability of calculational prediction of the AST reactor thermal-hydraulics [13].

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THERMAL-HYDRAULIC DESIGN OF THE 200 MW NHR

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Abstract
Abstract

The main problems regarding the AST-500 NHR thermal-hydraulics are considered. Basic thermal data of the reactor plant are given and peculiarities of coolant parameters at natural convection in the primary circuit are discussed. The in-reactor instrumentation system is briefly described, as well as the results of natural-convective flow characteristics investigations using reactor test models.

1. INTRODUCTION

In the development of a nuclear reactor with natural convection of the coolant, the issues of primary circuit hydrodynamics and thermal-physics play an exceptionally important role. Its thermal and hydraulic characteristics determine the possibility to provide the basic design parameters of the reactor, the operating conditions of the core, its thermotechnical reliability, and the reactor behaviour under transient and emergency conditions. Intensification of coolant natural convection by allowing boiling in the core requires a comprehensive analysis of two-phase coolant flow hydrodynamics in the reactor.

A complex of calculations and experimental investigations carried out on the AST-500 thermal-hydraulics [1-4] allowed to reveal the most essential problems that should be solved, particularly:

- loop-level and interchannel instability of natural circulation flow and their mutual influence;
- interrelations between hydrodynamic and neutron-physical processes;
- water chemistry and gas control mode in conformance with the primary circuit of an integral PWR.

A large volume of experimental data has been obtained at thermal- physical test rigs, and the design analysis results have shown that for the AST-500 reactor boiling conditions are possible in the range of stable coolant flow circulation at a certain combination of working parameters. However, for an in-depth study of the neutron-physical stability problem in the reactor, additional investigations would be needed. However, if non-boiling working conditions are adopted these problems would be eliminated.

Therefore, a non-boiling mode of operation was adopted for the AST-500 together with steam-gas pressurizing for primary circuit pressure control. The value of average subcooling with respect to the saturation point of the primary coolant was chosen proceeding from the requirements to provide stable flow circulation in the circuit in general, and to exclude the accumulation of radiolitic gases in the reactor up to the explosion-hazardous concentration.

2. BASIC THERMAL CHARACTERISTICS AND COOLANT NATURAL CONVECTION PECULIARITIES

The main parameters for the working range of the AST-500 reactor thermal power are given below:

Reactor thermal power	50-500 MW
Primary coolant flow rate	680-1550 kg/s
Reactor coolant temperature	
- core outlet	172-208°C
- core inlet	155-134°C
Reactor pressure	1.04-1.96 MPa
Secondary coolant temperature	
- reactor outlet	159°C
- reactor inlet	152-87°C

The reactor coolant flow rate and temperature values provide for a reliable removal of heat from the reactor core and for its subsequent transfer to the secondary coolant via the in-reactor heat exchangers. The reactor coolant flow rate increases when core power rises which is especially important in emergency situations.

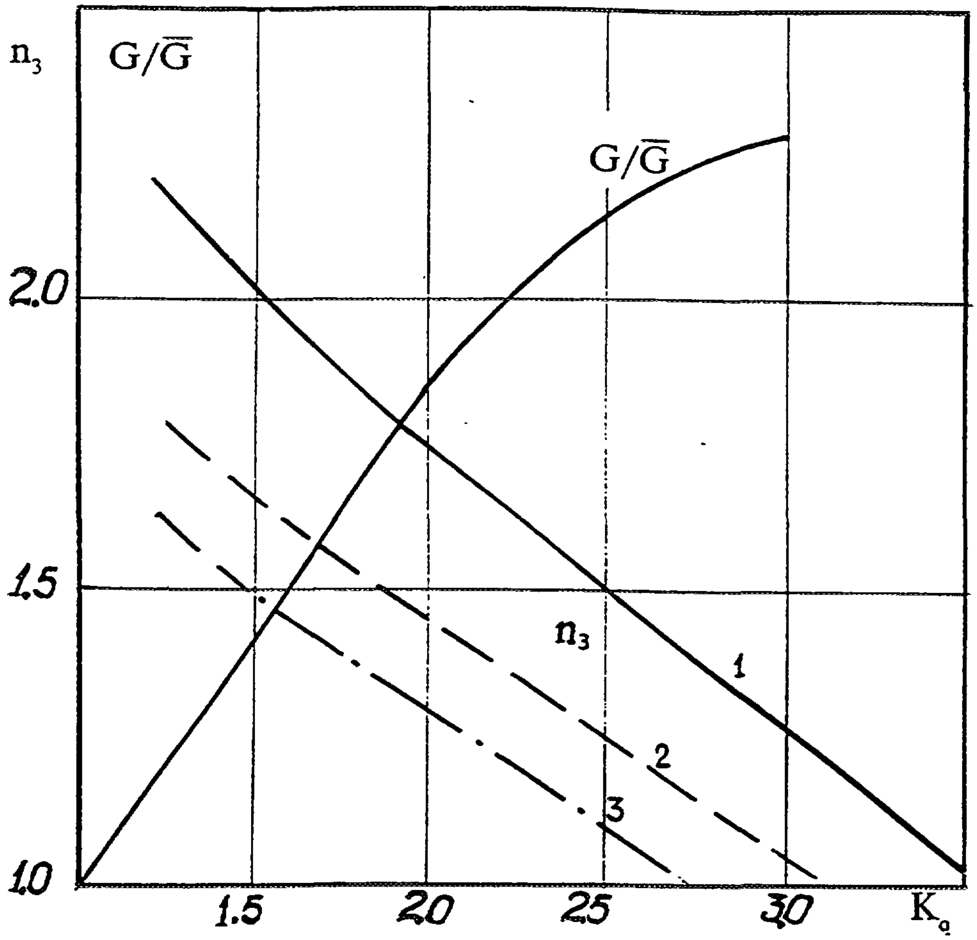
To shape the reactor coolant flow distribution through the core fuel assemblies, a set of individual draught tubes is used above the core structure. The draught tubes, made as extensions of the fuel assemblies, allow to provide a hydraulic self-profiling of the coolant flow through the core by increase of the individual hydrostatic head fraction in the pressure difference for the whole system of parallel channels. When core power rises the corresponding flow rates are established in fuel assemblies.

This feature of coolant natural convection allows to tolerate significant local rises of heat release in fuel assemblies thanks to the coolant flow rate self-shaping. The main indicator of the core thermotechnical reliability is the critical power factor. The necessity of considering this factor is conditioned by the principal possibility of a core power rise up to the reactor protection setpoint at unfavourable superpositions of deviations of the parameters defining the character of heat removal in the fuel assemblies. Reactor power, thermal power distribution over the core fuel assemblies, pressure value, core inlet temperature, and coolant flow distribution through the core fuel assemblies are the essential parameters determining the margin of the fuel assemblies with regard to critical power.

Account of possible deviations of the parameters from their nominal values reduces the core thermotechnical reliability. For a conservative evaluation, the deviations were treated by the procedure of considering an ultimately unfavourable (from the viewpoint of boiling crisis appearance) superposition of the deviation of parameters.

Variations of the DNBR-factor (n_3) and of the relative flow rate through a fuel assembly (G/G_0) versus relative power of the core fuel assembly (Kq) at rated parameters of the reactor (curve 1), accounting for parameter deviation (curve 2) and accounting for parameter deviation along with a reactor power increase up to the emergency protection setpoint (curve 3), are given in Fig.1 as an example.

So, a local increase of fuel assembly power up to the values of 2.5-3.0 does not result in a deterioration of the core thermotechnical reliability.



1. at nominal parameters;
2. at the account of parameters deviation;
3. at the account of parameters deviation and power increase up to emergency protection set point

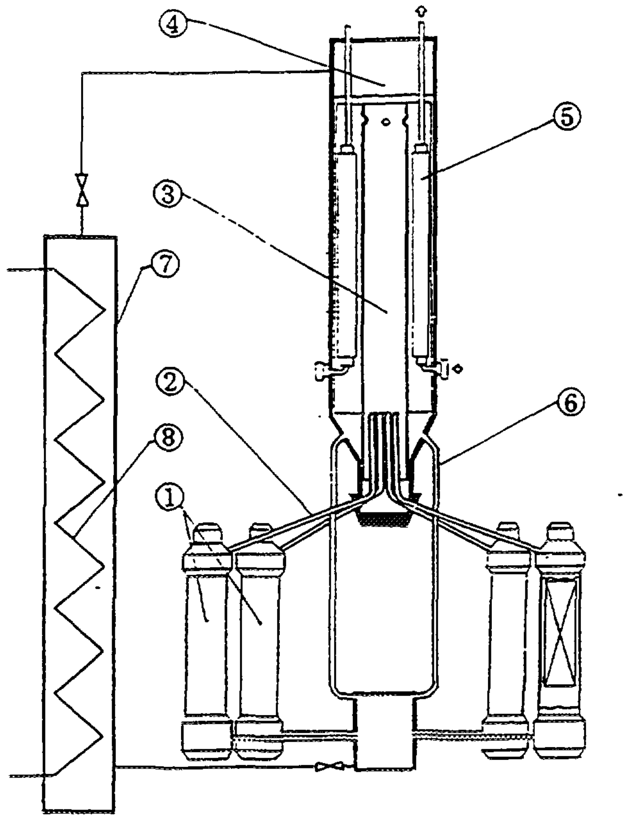
Fig.1. **Change critical power ratio and flow-rate through fuel assembly depending on relative power of fuel assembly**

At a reactor operation with one section of the heat exchanger isolated when a secondary coolant pump is stopped in one of the loops, the primary coolant circulation through an inoperative HX's section is virtually absent. It is associated with the diminishing of the hydrostatic fraction of the total pressure drop on the heat exchanger at loss of heat removal in it.

This feature, which is also associated with the reactor operation at natural convection of the primary coolant, allows to provide a uniform temperature of coolant at the core inlet in operating modes with heat removal skewing over the secondary circuit loops.

3. IN-REACTOR INSTRUMENTATION SYSTEM

Along with the control of the overall thermotechnical parameters of the reactor such as power, pressure, coolant level in the reactor, core inlet and outlet temperatures, coolant temperature near its surface, an in-reactor instrumentation system is provided for the AST-



1.- Heating columns
2.- Chimney tubes
3.- Riser
4.- Steam-gas pressurizer
5.- Heat exchanger
6.- Downcomer
7.- Guard vessel simulator
8.- GV heat exchanger

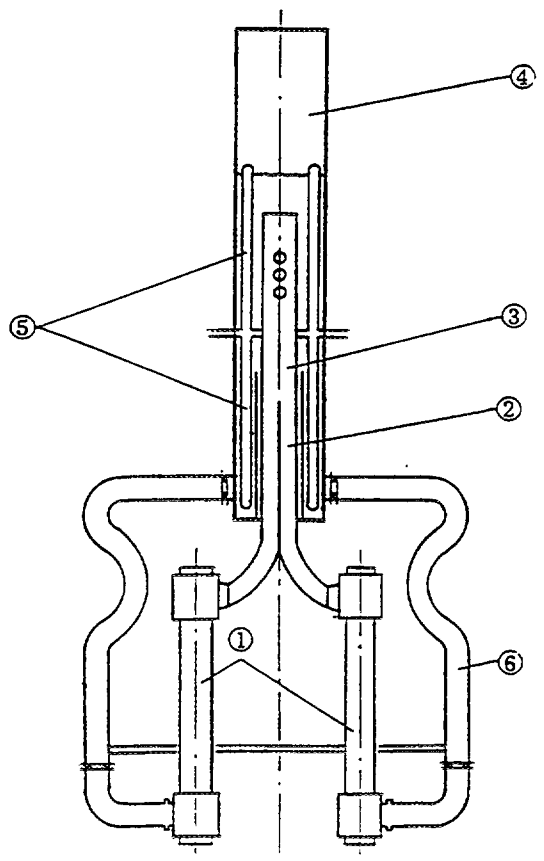
Fig.2. 1385 test facility. Model of integral reactor

500. For this aim, a zirconium tube (13.6x0.9 mm) is installed in the fuel assemblies instead of one of the fuel rods for placing an instrumentation probe comprising thermal transducers and neutron flux detectors.

The in-reactor instrumentation system gives on-line information about the neutron flux distribution over the core volume, coolant flow rates and temperatures in individual fuel assemblies. The number of instrumentation probes is chosen to reproduce a full picture of the neutron flux distribution in the core. The neutron flux detectors in each probe are positioned at several points along the core height. The coolant temperature is monitored by thermocouples located in the probes at the core inlet and outlet. Besides, each probe includes special flow meters which are prior calibrated in flow rigs.

4. PRIMARY CIRCUIT THERMOHYDRAULICS INVESTIGATION ON TEST FACILITIES

Comprehensive investigations of the thermohydraulic characteristics of a natural convection circuit have been carried out at the thermophysical test rigs 1385 and at KMR, both representing integral reactor models [1-4]. The rigs are different in a scale of electrically heated models of the AST-500 reactor. They include all main components of the



1.- Heating columns
2.- Chimney tubes
3.- Riser
4 - Steam-gas pressurizer
5 - Heat exchanger
6 - Downcomer

Fig.3. Natural circulation circuit of KMR-2 test facility

primary circuit: a core simulator which is composed of several heater rod assemblies with individual draught tubes, a common riser section, built-in pressurizer, overflow section, built-in heat exchanger and down comer section.

The 1385 test rig (Fig.2) comprises four heater rod assemblies of a total power up to 1.8 MW. The scale of the reactor circuit simulation in respect to height is 1:2.

The KMR test rig (Fig.3) represents a large-scale model of the primary circuit with height dimensions close to the actual AST-500 ones. The scale of the primary circuit simulation in respect to volume and flow path sections of the circuit's items is 1:170. The maximum power of the rig is 2.5 MW. The core simulator consists of two electrically heated 37-rod assemblies with rod dimensions similar to the real ones.

The distribution of hydraulic resistance along the primary circuit and the relation of the heights and flow path sections of different circuit's items were simulated in the rigs. The investigations were carried out in a range of parameters overlapping the range of AST-500 operating parameter variation.

Natural convection flow depending on thermal power, coolant pressure, core outlet enthalpy, and power distribution over the heater rod assemblies were studied in the experiments. In addition, gas transport and gas distribution processes in the primary circuit were investigated. The characteristics of the built-in pressurizer were studied as well. The main tasks of the experiments were to define the area of stable flow circulation, and to study the peculiarities of flow distribution in the circuit's items, including modes with significant skewing of power distribution over the fuel assemblies and of the heat removal over the HX sections.

Some results of comparing experimental data with calculation analysis results regarding the flow rate characteristics of the test rigs are given in Figs. 4 and 5. It might be concluded that both kinds of results are in a good agreement.

Along with the experiments on the reactor models, hydraulic tests were carried out on full-scale models of other items of the primary circuit under isothermal conditions (fuel assemblies, HX sections, etc).

The hydraulic characteristics obtained in the experiments were used in the AST-500 reactor plant thermal-hydraulic design. The described combination of comprehensive experimental investigations of the reactor thermal-hydraulics and the analytical procedures based on experimentally-verified calculation models allowed to validate reliably the AST-500 characteristics under non-boiling operating conditions.

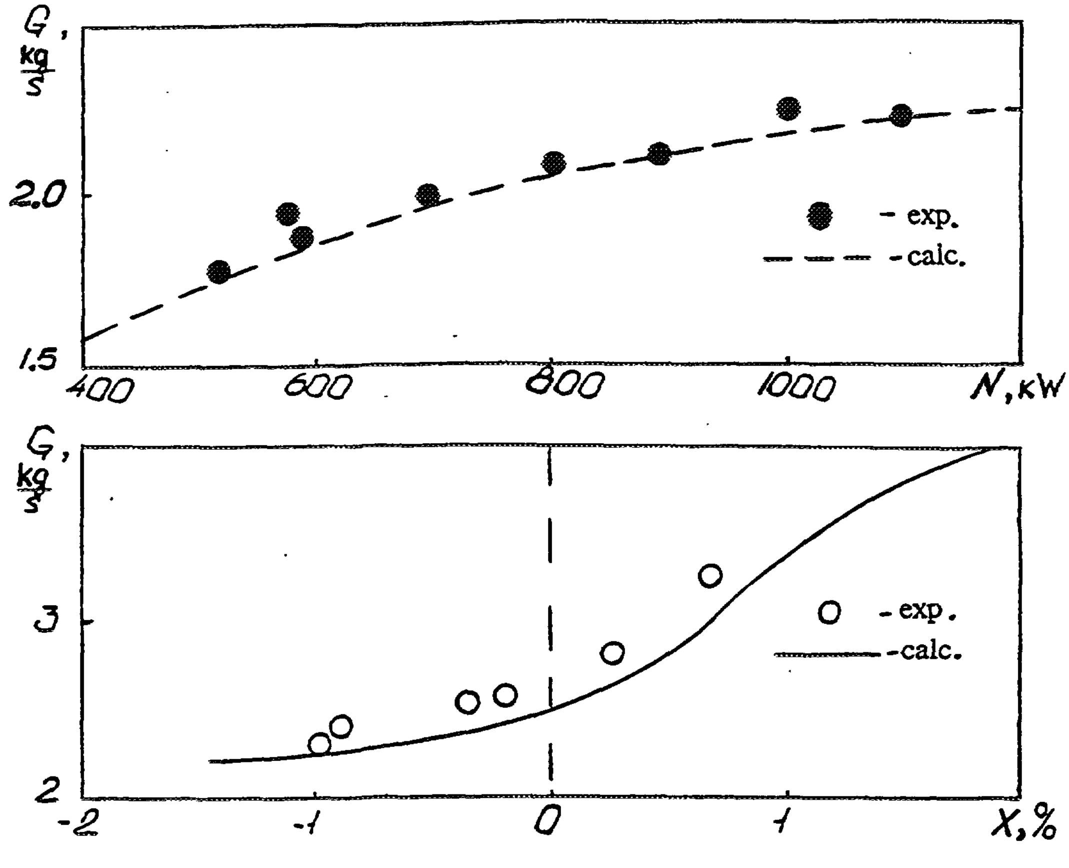


Fig.4. NC flow rate characteristic of 1385 test facility. Comparison with calculations. P ≈ 2.0 MPa

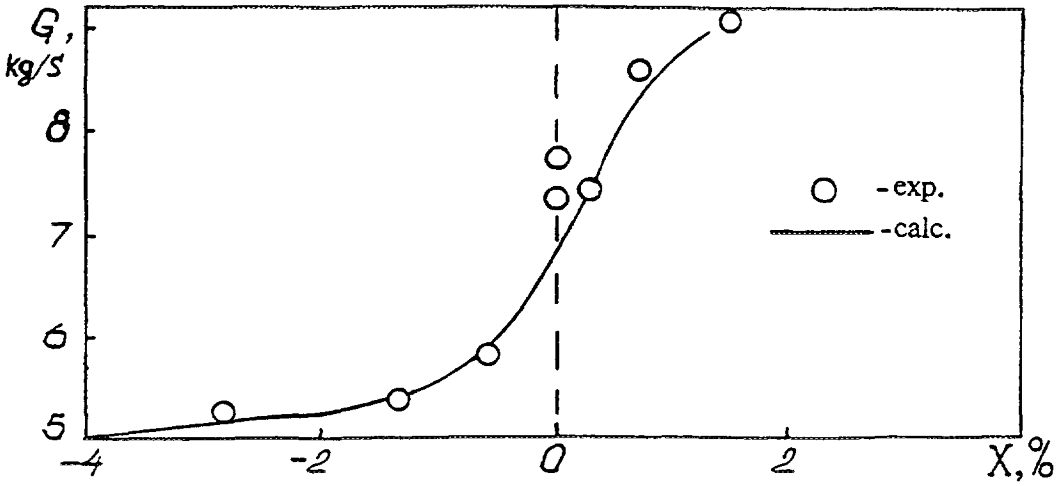


Fig.5. **NC flow rate characteristic of KMR-2 test facility. Comparison with calculations. P ≃ 2.0 MPa, N ≃ 1600 kW**

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THERMAL-HYDRAULIC DESIGN OF THE 200 MW NHR

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Abstract

The thermal hydraulic design of the 200-MW Nuclear Heating Reactor (NHR), design criteria, design methods, important characteristics and some development results are presented in this paper.

1. DESIGN TASK AND IMPORTANT CHARACTERISTICS

The 200-MW NHR is a demonstration nuclear heating reactor. It operates at low temperature and low pressure conditions in an integrated primary circuit with natural circulation and a self-pressurizer filled with nitrogen gas. Its longitudinal section is shown in Fig. 1.

The basic task of the thermal hydraulic design of the 200-MW NHR is to provide an adequate heat transfer ability that is matched with the heat-generating ability in the core, to provide a set of reasonable parameters for the intermediate loop, to enable the 200-MW NHR to have good economic benefits under the condition that the integrity of the three levels of radioactivity barriers for preventing the release of radioactive products is ensured, and that the requirements of various operation safety are satisfied.

The 200 MW NHR will be used for domestic heating for cities, so adequate safety is highly required. Based on the design and operation experience with the 5-MW Experimental Heating Reactor (NHR-5), the design of the 200-MW NHR is underway. The thermal hydraulic design has different characteristics in comparison with conventional PWR nuclear power plants as follows:

- integrated arrangement of the primary circuit, natural circulation operating pattern,
- a self-pressurizer in the upper dome of the reactor vessel with nitrogen gas,
- fuel bundle with box, flow distribution by throttling at the core inlet,
- lower temperature, lower pressure, lower reactor thermal parameters (volume power density, linear power density).

2. DESIGN CRITERIA

There are no official design standards for NHRs in China up to now. Under the agreement of the National Nuclear Safety Administration of China, the following criteria are used in the design of 200-MW NHR [1].

General design criteria:

- To assure fuel elements not to be damaged in normal operation, anticipated operating occurrences and infrequent faults.
- Fuel elements may be damaged in limited accidents, but the effective radioactive

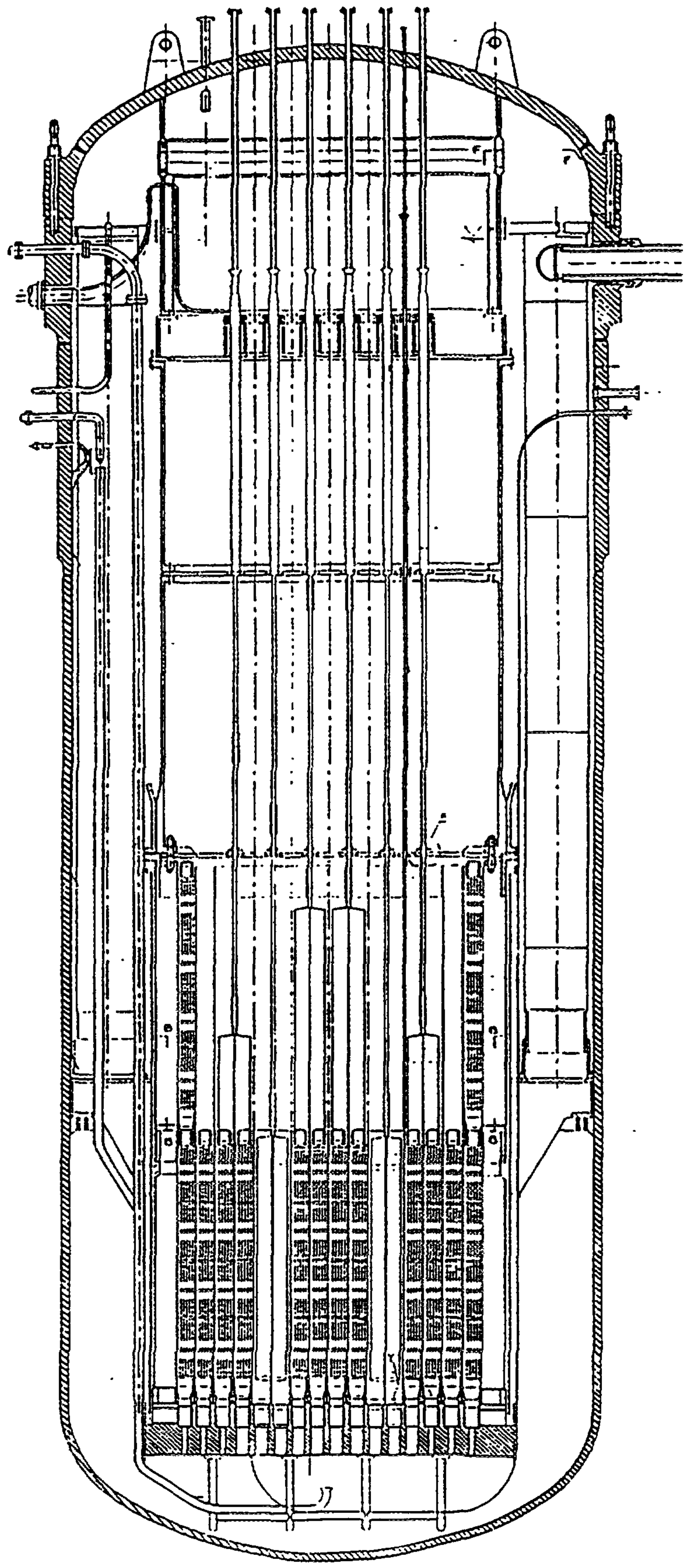

Fig.1 Longitudinal section of NHR-200

dose-equivalent accepted by any individual should be lower than 5 mSv, and the thyroid gland dose-equivalent should be lower than 50 mSv.

- The reactor core is not to melt in limiting beyond design accidents; the effective radiation dose-equivalent received by any individual should be lower than 5 mSv and the thyroid gland dose-equivalent should be lower than 50 mSv.

Design criteria used in the design:

- Departure from nucleate boiling ratio (DNBR)

  The advanced Barnett formula is used to analyze DNBR. The minimum DNBR (MDNBR) should be larger than 1.35 in normal operation, anticipated operating occurrences and infrequent faults.

  According to the analysis of DNB experiments of twenty-rod bundles and simulation rod bundle experiments for the NHR-5, satisfying a 95% confidence and a 95% probability level, the limit value of MDNBR for the 200 MW NHR should be 1.15. Considering fuel rod bending and a suitable DNB margin, the final DNBR limit value of 1.35 is suitable.

- Fuel temperature:

  In operation (see paper "Safety Objectives and Design Criteria for the NHR-200") Categories I, II and III, the maximum temperature of the fuel pellets should not be larger than 2590°C.

- Hydraulic instability:

  In operating Categories I, II and III, there should be no hydraulic instability in the 200-MW NHR primary circuit.

- Core covered with water:

  The 200-MW NHR is operated under lower pressure; the surface temperature of the fuel rods is not a limiting factor as long as the core is covered with water in any condition.

- Other limits:

  There are still other limits associated with other systems or components that should be satisfied.

3. THERMAL DESIGN EXPLANATION

3.1. Computer codes

The RETRAN-02, COBRAIIIC/MIT-2 and the STEADY-LTHR codes are used in the thermal hydraulic design of the 200-MW NHR.

The RETRAN-02 code [2] is used to analyze steady and dynamic processes of 200 MW NHR systems. It can provide the distribution of mass flow, pressures and temperatures, etc., in the primary circuit.

The COBRAIIIC/MIT-2 code [4] is used for the detailed analysis of subchannels and rod bundles in the core. It can provide detailed distributions of mass flow, temperatures, pressures, and DNBR in subchannels and rods, respectively.

The STEADY-LTHR code [5] is used for the analysis of the steady parameters in the primary and the intermediate loop, and in the thermal network. It can also be used to analyze the mass flow distribution in side-by-side channels.

3.2. DNBR analysis

The AD-Barnett formula is used for the DNBR analysis in the thermal-hydraulic design of the 200-MW NHR. The DNB formula is concluded on the basis of DNB experiments of rod bundles [3]. Its suitable range and the design parameters of the 200-MW NHR are shown in Table 1.

TABLE 1: EXPERIMENT DATA SCOPE AND 200 MW NHR PARAMETER

Parameter	Unit	Data scope	200-MW parameter
Rod diameter	mm	10-14.3	10
Heating section length	mm	836-4440	1900
Pressure	MPa	1.03-5.0	2.5
Mass flow flux	kg/s-m^2	34-2288	494
Inlet subcooling	kJ/kg	13.9-798	350

The calculated results of the AD-Barnett formula are compared with 743 experimental data in Fig. 2. 95% of the data are inside a band of 15% difference, the spread of all data differs not more than 20%. The Root-Mean-Square (RMS) is 6.01%. The simulating experiments concerning DNB experiment for NHR-5 were conducted at the Nuclear Power Institute of China.

Experiment parameters:

Heating rod diameter:	10 mm
Rod lattice pitch:	13.3 mm
Rod cluster:	square 3x3 or 4x4
Heating section length:	800 mm
System pressure:	1.2-1.8 MPa
Mass flux:	500-1400 kg/s-m2
Critical quality:	-0.05 $\pm$ 0.05

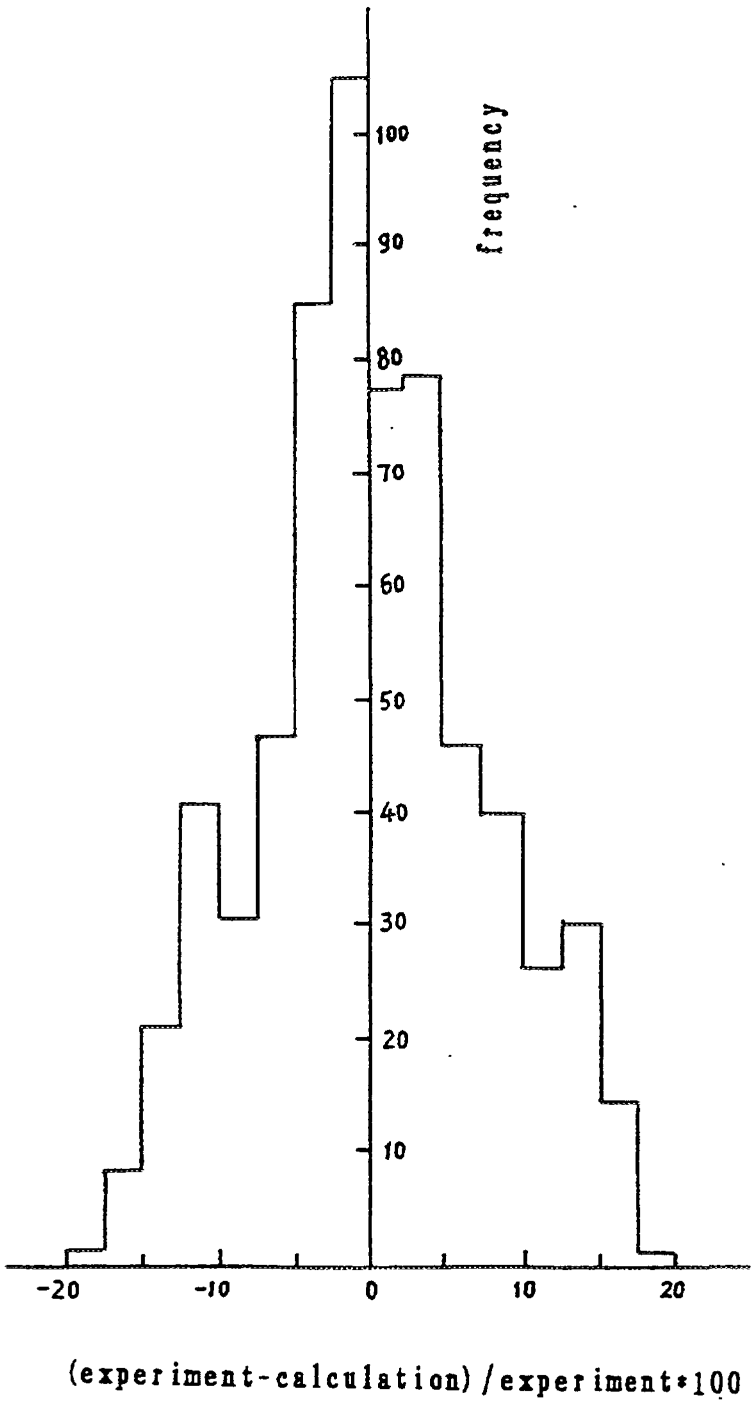

Fig.2 comparison between AD–Barnett and 743 experiments

The results calculated with the AD-Barnett formula are compared with 98 experimental data. The comparison of analysis and experiments shows for:

55.1% of the data	a relative difference < 5%
82.7% of the data	a relative difference < 10%
95.9% of the data	a relative difference < 15%
100% of the data	relative difference < 20%

The result is the same as for the 743 experimental data [3].

3.3. Nuclear heat flux factor and engineering hot spot factor

The axial nuclear heat flux distribution at the beginning of life (BOL) of the 200-MW NHR is shown in Fig. 3. The radial power distribution of the rod clusters in the core is shown in Fig. 4. The maximum local nuclear heat flux factor for the fuel rods in rod bundles is 1.32. At BOL the overall hot spot factor is 4.39. The overall hot spot factor at mid-life and end-life will be lower than that at BOL.

The effects of the difference between real parameters and the rated values such as the fuel pellet diameter, pellet density, fuel enrichment, and the heat flux will be included by the Engineering Hot Spot Factor (EHSF). The EHSF is 1.04 in the 200-MW NHR. Considering other effects, a total EHSF of 1.10 is used in the 200-MW NHR design.

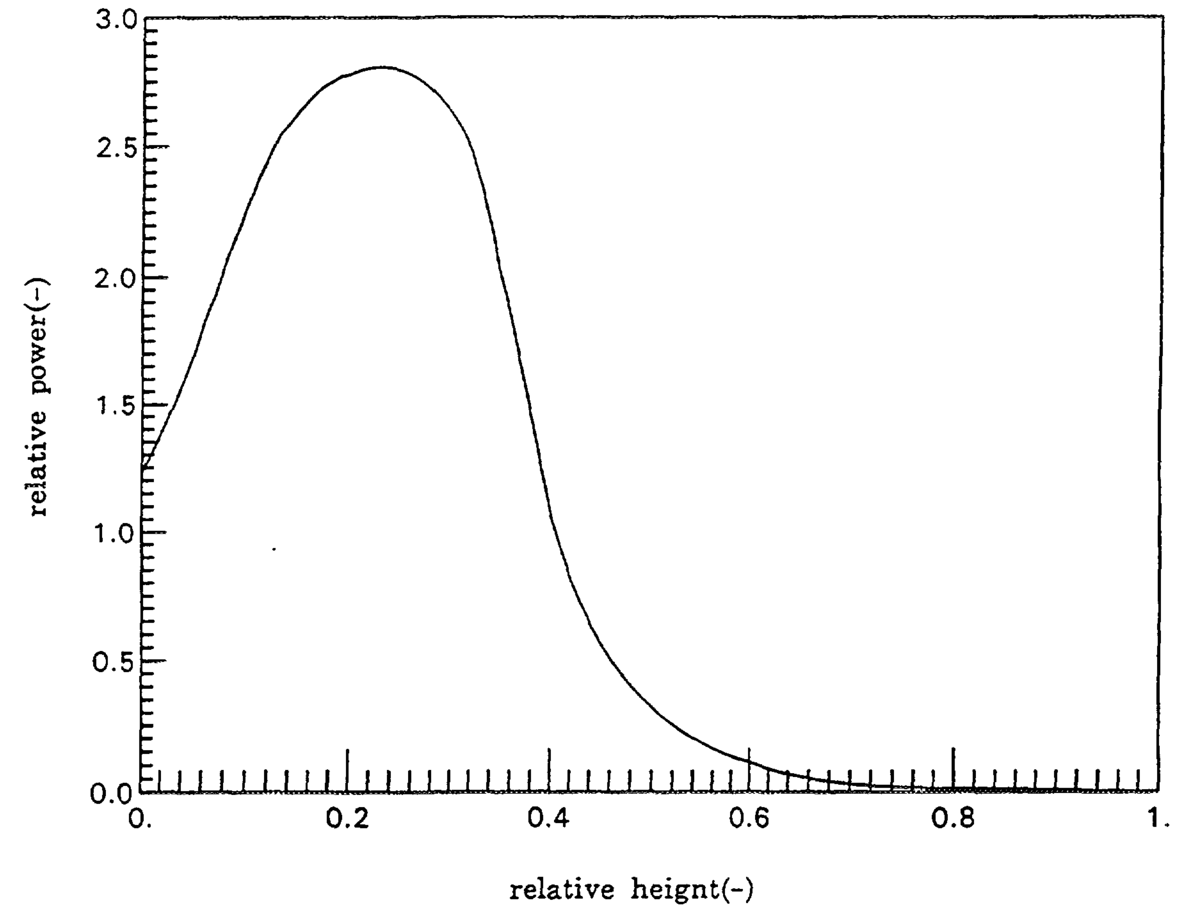

Fig.3 Axial power distribution at BOL

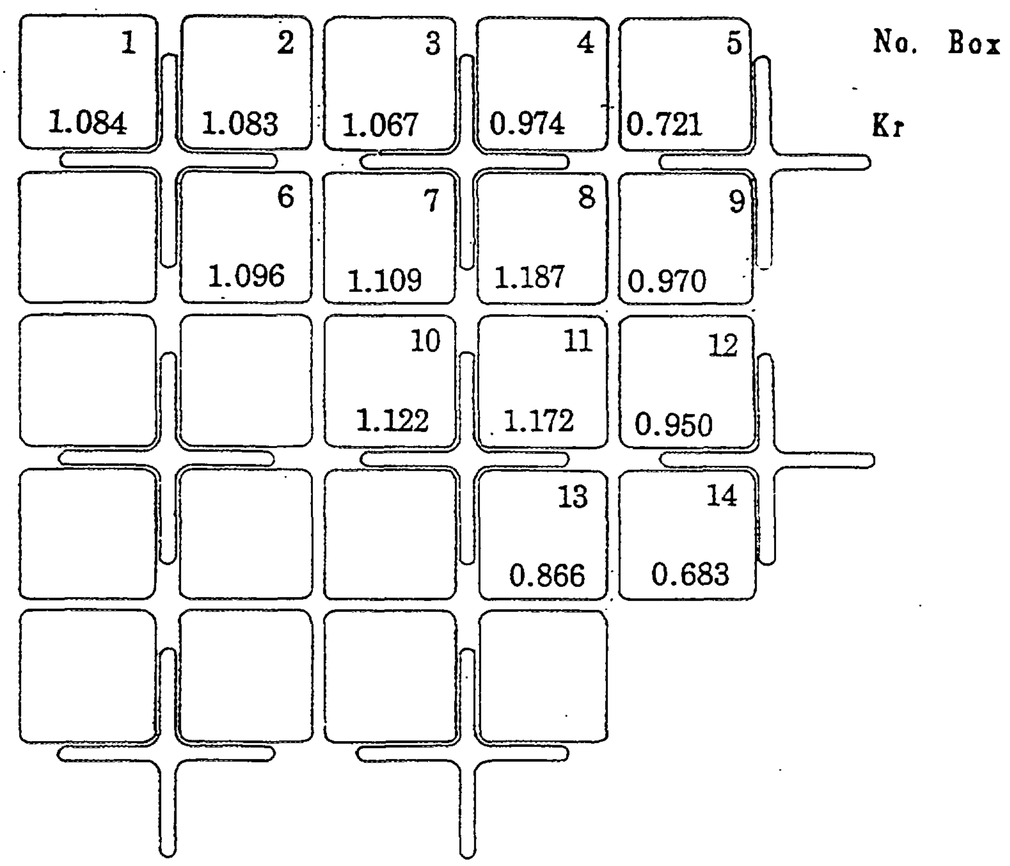

Fig.4 Radial power distribution of rod bundles at BOL

3.4. Temperature analysis in fuel rod

The following UO_2 thermal conductivity formula is used in calculating the temperature of fuel rods.

$$K=38.24/(t+402.55)+4.788e^{-13}(t+273.15)^3$$

with:

t: fuel temperature, °C,
K: thermal conductivity, W/cm-°C

It is assumed that the equivalent heat transfer coefficient between fuel pellet and cladding is 5678 W/m^2-°C.

3.5. Thermal parameters of the 200-MW NHR

Some thermal parameters of the 200-MW NHR at BOL and at rated operating condition are shown in Table 2.

TABLE 2: SOME THERMAL PARAMETERS

Parameter	Unit	NHR-5	200-MW NHR
Reactor power	MW	5	200
Primary system pressure	MPa	1.47	2.5
N_2-pressure	MPa	0.32	0.59
Total mass flow	t/h	106.6	2527
Effective core flow	t/h	100	2376
Bypass/total flow	--	0.06	0.06
Average velocity in core	m/s	0.24	0.57
Average mass flux in core	kg/s-m^2	202	494
Core inlet temperature	°C	147	145
Riser outlet temperature	°C	186	210
Average thermal flux	w/cm^2	17.8	24.6
Maximum thermal flux	w/cm^2	78.1	107.9
Average liner power density	w/cm	56.1	77.2
Maximum liner power density	w/cm	246	338.7
Core power density	kW/l	24.0	36.2
MDNBR	--	2.39*	2.65
Maximum fuel temperature	°C	1125	1366
DNB formula	--	AD-Barnett	AD-Barnett

* Including a safety factor of 1.3.

3.6. Comparison with NHR-5

The NHR-5 has been successfully operated for 5 heating seasons since 1989. A set of experiments associated with operation and safety were conducted. The results of the experiments show that the NHR-5 has a good inherent safety. The NHR-5 design was very successful.

The design and operational experience of the NHR-5 are fully taken into account in the design of the 200-MW NHR. A comparison of some important parameters between the NHR-5 and the 200-MW NHR is shown in Table 2.

After a full consideration of safety issues, higher primary loop pressure, higher core heat flux, higher core power density than NHR-5 were selected for the design of the 200-MW NHR. This is helpful to advance the economics of the 200-MW NHR.

3.7. Hydraulic instability

In some conditions a hydraulic instability may take place in the steam-water two-phase flow under low pressure in a natural circulation loop. This type of instability normally is a density wave oscillation or dynamic instability. To avoid this type of instability, nitrogen gas is filled in the upper dome of the reactor vessel. Under rated operating conditions, the total system pressure in the primary loop is 2.5 MPa. The nitrogen partial pressure is 0.59 MPa and the steam partial pressure is 1.91 MPa. The riser outlet temperature is 210°C. The saturation temperature is 224°C at 2.5 MPa pressure. The subcooling temperature of 14°C between riser outlet temperature and saturation temperature is maintained during operation by regulating the intermediate loop parameters. The subcooling temperature of 14°C at riser outlet will make the following possible:

- there is no boiling in the core at rated operating conditions, and at anticipated operating deviations,
- the 200-MW NHR will not go into an instability region of steam water two phase flow in accident conditions.

3.8. Flow distribution in the core

Rod clusters within a fuel box are used in the 200-MW NHR. Through flow distributors in the fuel clusters, a limited natural circulation flow in the primary loop is effectively used. The total mass flow is 702 kg/s in the primary loop at rated operating conditions. The mass flow through the gap between cluster boxes and other bypass gaps is considered ineffective. It is about 6% of the total flow. Total effective flow through the fuel boxes is 660 kg/s.

According to the radial power distribution an appropriate flow distribution is achieved by installing throttles at the inlet of the cluster boxes. By distributing flow, the clusters with larger heating power have larger mass flow. The difference of radial power distribution at BOL, MOL and EOL are considered in flow adjustment to get the best use of flow in any period of the life.

The mass flow distribution at BOL is shown in Fig. 5. The outlet water temperature distribution at BOL, MOL and EOL are shown in Fig. 6.

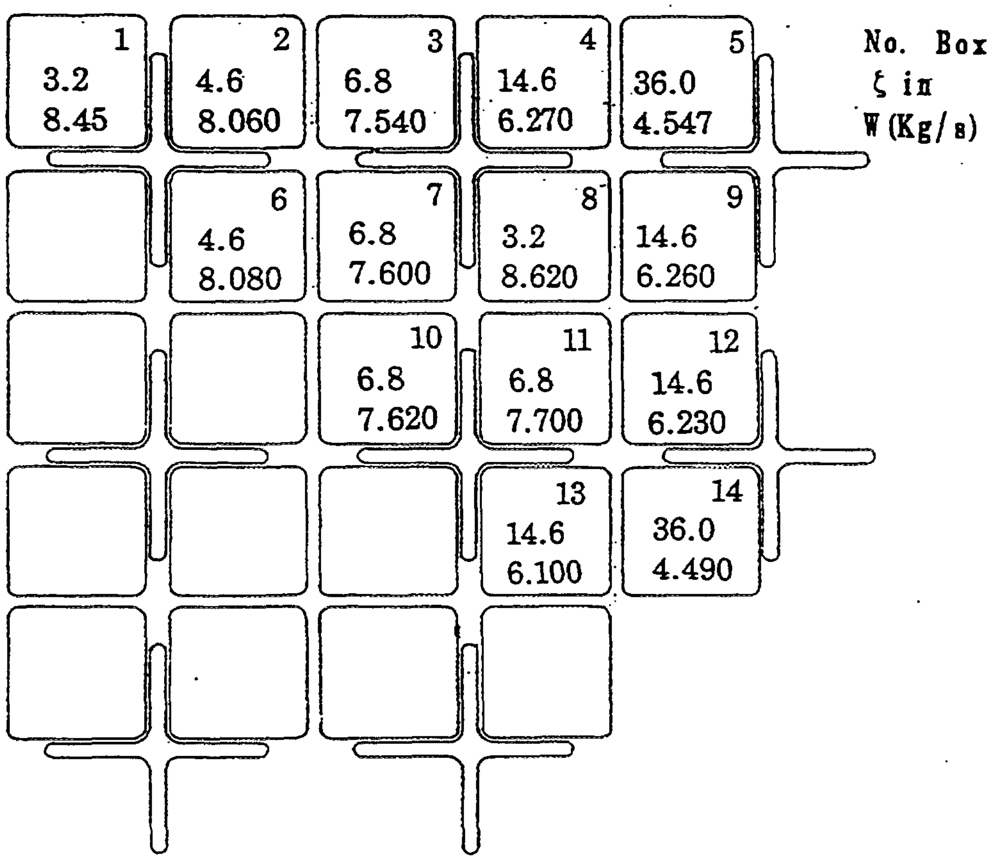

Fig.5 Core flow distribution at BOL

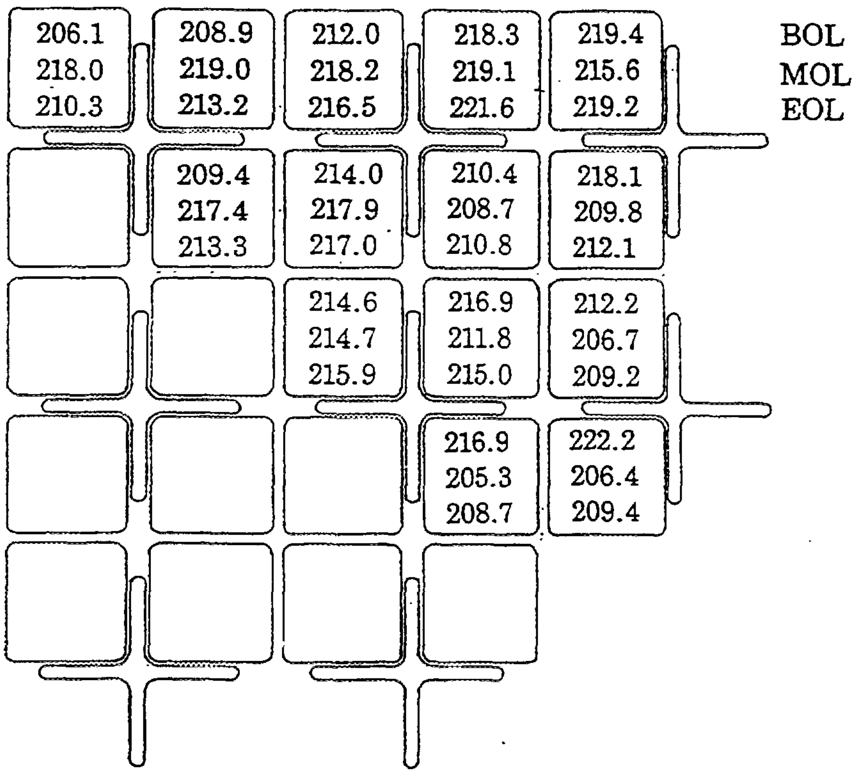

Fig.6 Outlet temperature distribution at BOL, MOL and EOL

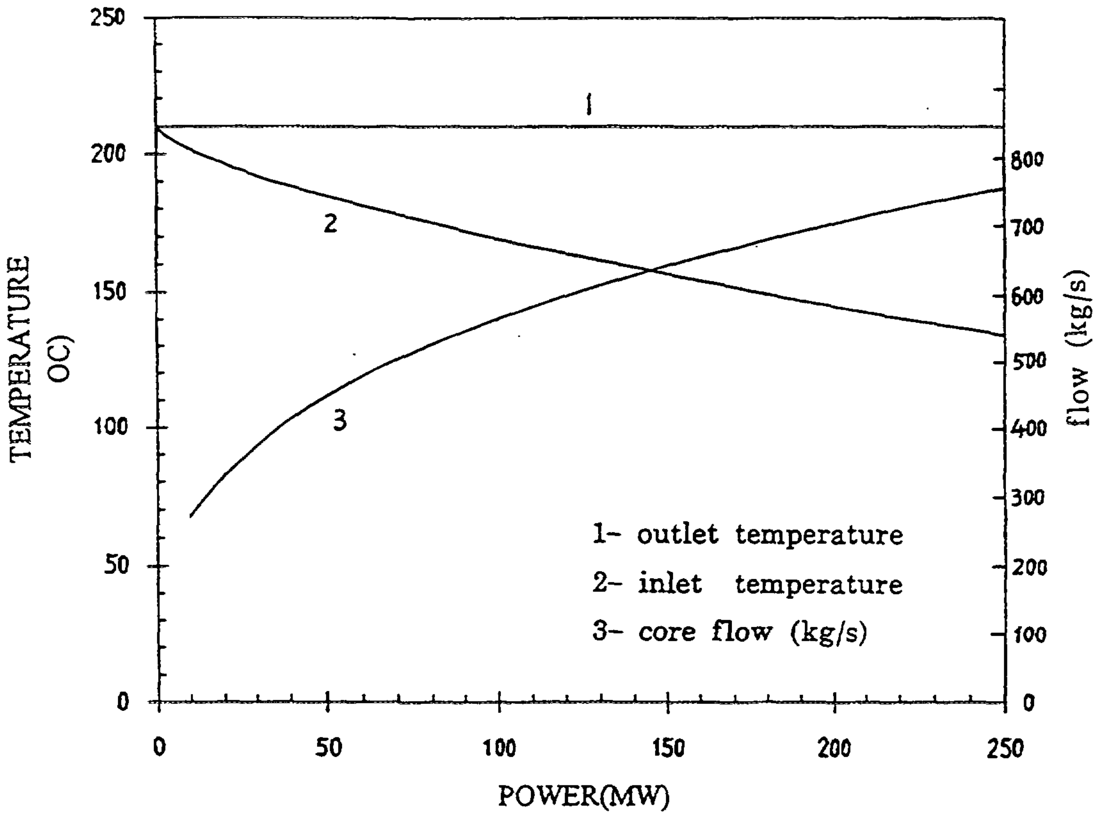

Fig.7 Natural circuilation character:
flow and inlet temperature chage with power

3.9. Natural circulation character of primary loop

Natural circulation operation is used in the integrated primary loop of the 200-MW NHR. The flow circulation is somewhat different from PWRs in which the flow circulation is driven by pumps in the primary loop. The natural circulation flow in the primary loop, and the core inlet temperature in the 200-MW NHR will change with different operating power. By maintaining constant outlet temperature of the riser and constant pressure in the upper dome of the reactor vessel, the natural circulation flow increases and core inlet temperature decreases with reactor power rising. That is shown in Fig. 7.

3.10. The thermal response in accidents

The most dangerous accident for the 200-MW NHR regarding MDNBR and fuel maximum temperature is an unexpected control rod drop. When the induced negative reactivity by dropping one control rod is unable to scram the 200-MW NHR, assuming that the protection of the fast decrease of the neutron flux is not available, and the intermediate circuit and the thermal network circuit continue in normal operation, the accident will cause the moderator temperature in the core and the fuel temperature to decrease. The induced positive reactivity feedback by the decrease of moderator and fuel temperature will be compensated the induced negative reactivity by dropping the control rods. Finally the reactor power will increase and return to about the initial power level. The change of the reactor power by dropping a 0.5$ control rod under the described severe assumptions is shown in Fig. 8.

Because of the drop of a control rod into the core, a radial power distortion is induced. The neutron flux in 1/8 quadrant that is symmetric to the quadrant of the dropped control rod will rise. The MDNBR will decrease and the maximum fuel temperature will increase.

The radial power worsening coefficient for fuel elements is defined as the ratio of the radial power factor of a cluster after dropping a control rod to that before dropping:

$$K_{rw} = K_r'(\text{after dropping}) \;/\; K_r\ (\text{before dropping})$$

Kr' is the radial power factor of a cluster after dropping. Kr is the radial power factor of a cluster before dropping.

The results of a two-dimension neutron physics analysis with the SIMULATE code show that the maximum radial power worsening coefficient is 1.34 when dropping one control rod into core. The location of the maximum radial power worsening coefficient is the location of the maximum radial power factor (1.187) before dropping the rod. The subchannel analysis results show that the MDNBR decreases from 2.65 at normal operating condition to about 2.0 after dropping a control rod. The maximum fuel temperature increases from about 1360°C increasing to about 2000°C.

4. CONCLUSIONS

The analysis presented in above sections show that the design criteria for the 200-MW NHR are satisfied. Although the power density is increased in the 200-MW NHR over that in the NHR-5, safety is still maintained under normal operating and under accidents conditions.

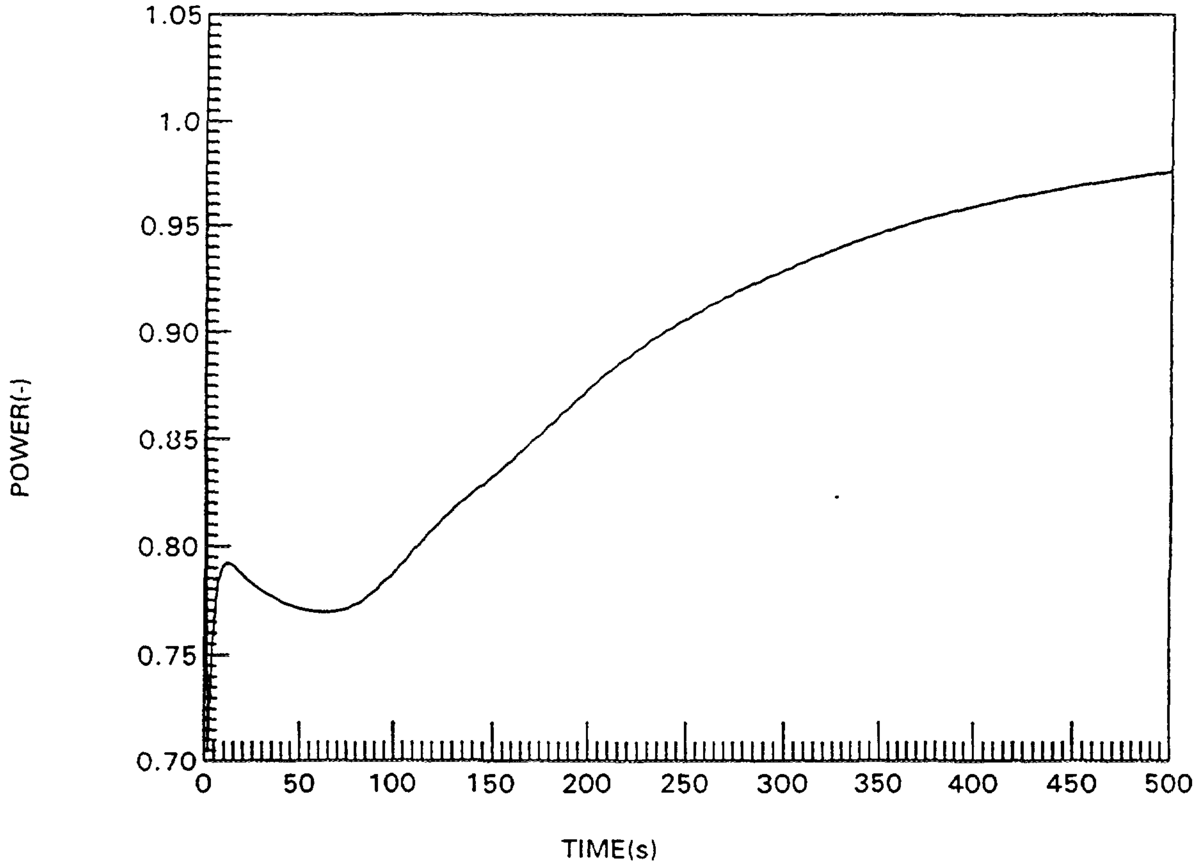

Fig.8 0.5 $ control rod drop accident:
change of reactor power against time.

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INVESTIGATIONS ON HYDRODYNAMIC STABILITY OF TWO PHASE FLOW IN A LOW PRESSURE NATURAL CIRCULATION SYSTEM

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Abstract

Appropriately scaled "Loop Stability" tests and "Channel Stability" tests were performed with single heated channel system and two parallel channel system separately at the Institute of Nuclear Energy Technology (INET) of the Tsinghua University in China. A broad range of several operational parameters such as heating power, system pressure, test inlet subcooling and resistance coefficient were investigated. It was found that under certain geometric conditions and operating parameters a self-sustaining, low frequency, even amplitude mass flow oscillation may be excited at very low steam qualities and subcooling conditions. Stability maps under different conditions have been provided to assist the design of the NHR.

1. INTRODUCTION

In nuclear reactors with water as moderator and coolant, the instability of thermal-hydraulic parameters affects the stability of the nuclear reaction. It is even more important that these systems actually operate under stable thermal-hydrodynamic conditions because the coupling between the thermal-hydraulic and the neutronic parameters may amplify any flow-induced instabilities under certain circumstances. In the recent 20 years, a great number of publications on theoretical and experimental research on this subject can be found [1, 3].

Along with the utilization of so-called quasi-boiling systems operating at low pressure, as in some heating reactor designs, low quality two-phase flow instabilities have been brought to more attention and have become an important area of recent two-phase flow research.

A nuclear heating reactor design recently developed by the Institute of Nuclear Energy Technology (INET) comprises a primary system operating at low pressure, at low steam quality and with natural circulation flow. The steam quality at the exit of the reactor core is less than 1%, the steam void fraction-due to the low system pressure-is relatively high. A riser above the fuel assemblies accommodates the steam-water mixture leaving the reactor core and provides the driving force for the natural circulation flow.

For this type of a natural circulation system, a random perturbation of the mass flow at the core inlet leads to two main feedback chains of the thermal-hydraulic parameters: (i) inlet mass flow rate, steam quality, void fraction, driving head for the natural circulation, mass flow rate; (ii) inlet mass flow rate, steam quality, void fraction, flow resistance in the two-phase region, mass flow rate. At any link of the above chains, a time delay and a spatial distribution of the thermal-hydraulic parameters are involved. Therefore, a perturbation of the inlet mass flow rate may lead to oscillations of mass flow and other parameters, such as temperature, pressure drop, void fraction and water level. The phenomenon is known as density wave oscillation. In order to investigate this phenomenon a series of tests was performed under a research program in INET. These tests included the investigation of the steady state characteristics and the stability behavior of the test loop under single channel and two parallel channel operation, uniform axial power distribution and various subcoolings and system pressures.

2. TEST SYSTEM

The test system consists of a primary and a secondary loop. The principal flow diagram and the arrangement of the test loop are shown in Fig.1 [5]. The primary loop comprises two parallel vertical test channels, risers and steam separators, one heat exchanger, steam condenser and downcomer, throttle valves and connection tubes. The design parameters of the primary loop are listed in Table 1. According to these parameters, the available exit steam quality is less than 5%. The length of the heated test section (including inlet plenum) is 1.2m. The longest riser height is 3m. The total height of the test system is about 7m.

Desalinated water is used as working fluid. It enters the heated test section with the required subcooling, leaves its exit as a steam-water mixture and flows through the riser into the steam separator. The steam coming from the separator gets into a condenser, from where the corresponding condensate flows back into the liquid phase section of the separator. The saturated water flows from the separator through the heat exchanger -where it reaches the set subcooling-through the throttle valve which is used for setting the resistance coefficient, through the main flow meter and the preheater, then flows back into the inlet of the two test sections and closes the natural circulation loop.

The system pressure and the inlet subcooling in the primary loop are adjusted by controlling the secondary mass flow through the heat exchanger and the condenser separately.

TABLE 1: PARAMETERS OF THE TEST SYSTEM

System pressure	20 bar
Water temperature	200°C
Power supply	200 kw, 0-72 V(AC), 2500 A
Inlet subcooling	5-45 K
System size	approx. 7 m
Resistance coefficient at the test section inlets	10-200

One test section consists of 16 electrically heated rods with uniform power distribution fabricated at PSI's laboratories [4]. The heated rods are arranged in a 4x4 cluster with a pitch of 13.3mm. The diameter of the rod is 10mm. Three venturi tubes are used to measure the channel mass flow rates and the total mass flow rate in the loop respectively. The pressure drops and the system pressure are detected by capacitance-type transducers. The temperature of the fluid at the inlet and outlet of the test sections, in the risers and in the steam volumes are measured with sheathed copper-constantan thermocouples.

3. MEASUREMENT OF THE STABILITY BOUNDARY

In this paper and in reference [4], power-subcooling conditions leading to a self-sustaining mass flow oscillation with a relative amplitude of 5% were defined as the stability

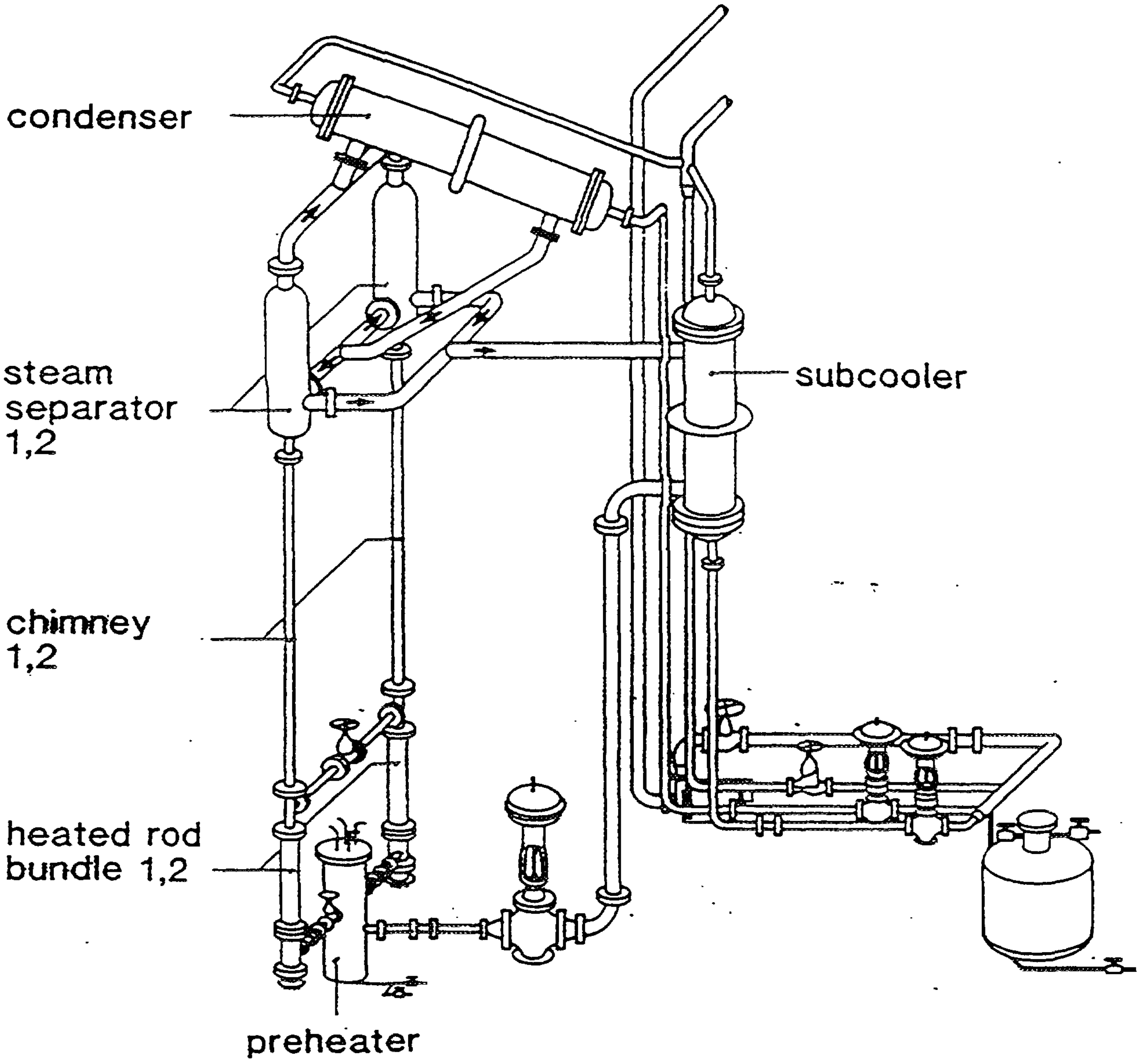


Fig. 1 Thermohydraulic stability test loop

boundary of the system. The relative amplitude of the mass flow rate Arel is given as the ratio of the peak-to-peak amplitude and the time-averaged mass flow rate

$$Arel = \Delta m \ / \ m = \Delta m \ / \ (1/\tau \int_0^\tau m dt)$$

where Δm is the difference between the local maximum and minimum of the mass flow rate oscillation (see Fig.2). m is the measured mass flow rate.

4. EXPERIMENTAL RESULTS

Single Channel Stability:

The effect of inlet subcooling on stability. Under low heating powers and high inlet subcoolings, there is only single phase flow in the system. In this case, it is obviously stable. But for a given channel power and system pressure, if the inlet subcooling is being reduced, the system suddenly falls into an unstable condition, and an even amplitude, low frequency, self-sustaining mass flow oscillation is being excited (a typical mass flow oscillation is shown in Fig.2).

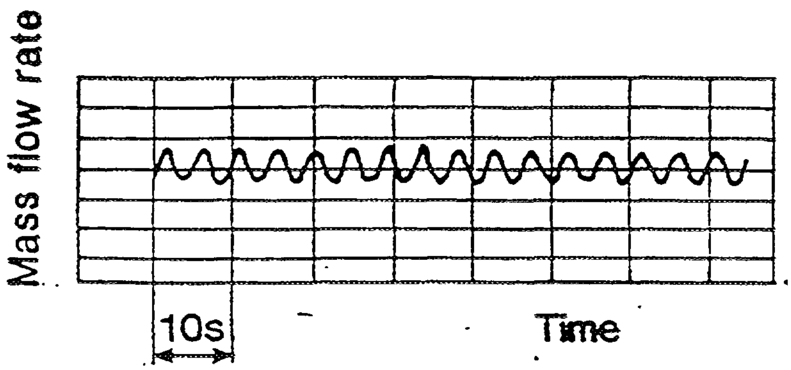
With a further reduction of the inlet subcooling, the relative amplitude of the mass flow oscillation steeply rises (the system becomes more unstable), reaches a maximum, and then sharply declines (the system becomes stable again). The dependency of the relative amplitude of the mass flow rate on the inlet subcooling for a given power is shown in Fig.3. There is an unstable region between two stable regions. According to the definition of the threshold of instability mentioned above, the unstable region has two boundaries: one at low, the other one at high subcooling rates.

Based on Fig.3, the subcooling-power stability map can be derived. Bearing in mind that the main control parameters of a heating reactor with natural circulation are power and subcooling, it is useful to provide the subcooling-power stability map shown in Fig.4. The stability map described via inlet subcooling and exit steam quality is shown in Fig.5.

The effect of system pressure on stability. The effect of the pressure on the system stability is shown by the stability map in Fig.6. The higher the system pressure the more stable the system at a given power. Under lower system pressure, the density difference between steam and water is more pronounced. So, the same steam quality perturbation gives rise to a bigger perturbation of void fraction and thus of the driving head for the natural circulation. Therefore, a system operation under higher system pressure is more stable.

The effect of inlet resistance coefficient on stability. The inlet resistance coefficient was changed by adjusting the opening of the throttle valve. For the resistance coefficients of Kin=60 and Kin=25, the oscillation of the mass flow rate at different power levels was recorded. The corresponding stability boundaries are shown in Fig.7. The bigger the inlet resistance coefficient the more stable the system.

Low quality instability. The dependency of the relative amplitude of the mass flow rate on the exit steam quality under different operating conditions is shown in Fig.8. This figure demonstrates that all unstable conditions with relative amplitudes greater than 5% lie in the low quality region with x less than 1.0%. There, according to the selection of the



Single channel
test conditions:

$p = 15bar$

$K_{in} = 25$

$K_L = 8$

$R = 3m$

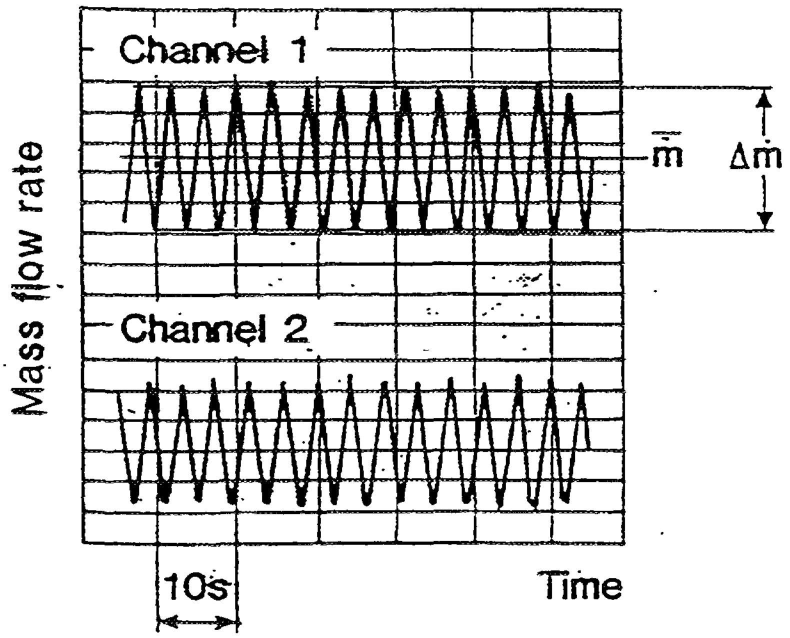
$\Delta T_S = 15K$

$\dot{Q} = 64kW$

No. B71807

$\Delta \dot{m}$ Peak–to–peak mass flow oscillation amplitude

$\bar{\dot{m}}$ Time–averaged mass flow rate



Parallel channel
test conditions:

$p = 15bar$

$K_{in} = 27$

$K_L = 8$

$R = 3m$

$\Delta T_S = 16K$

$\dot{Q}_N = \dot{Q}_S = 69kW$

No. B82909

Fig. 2 Mass flow oscillation

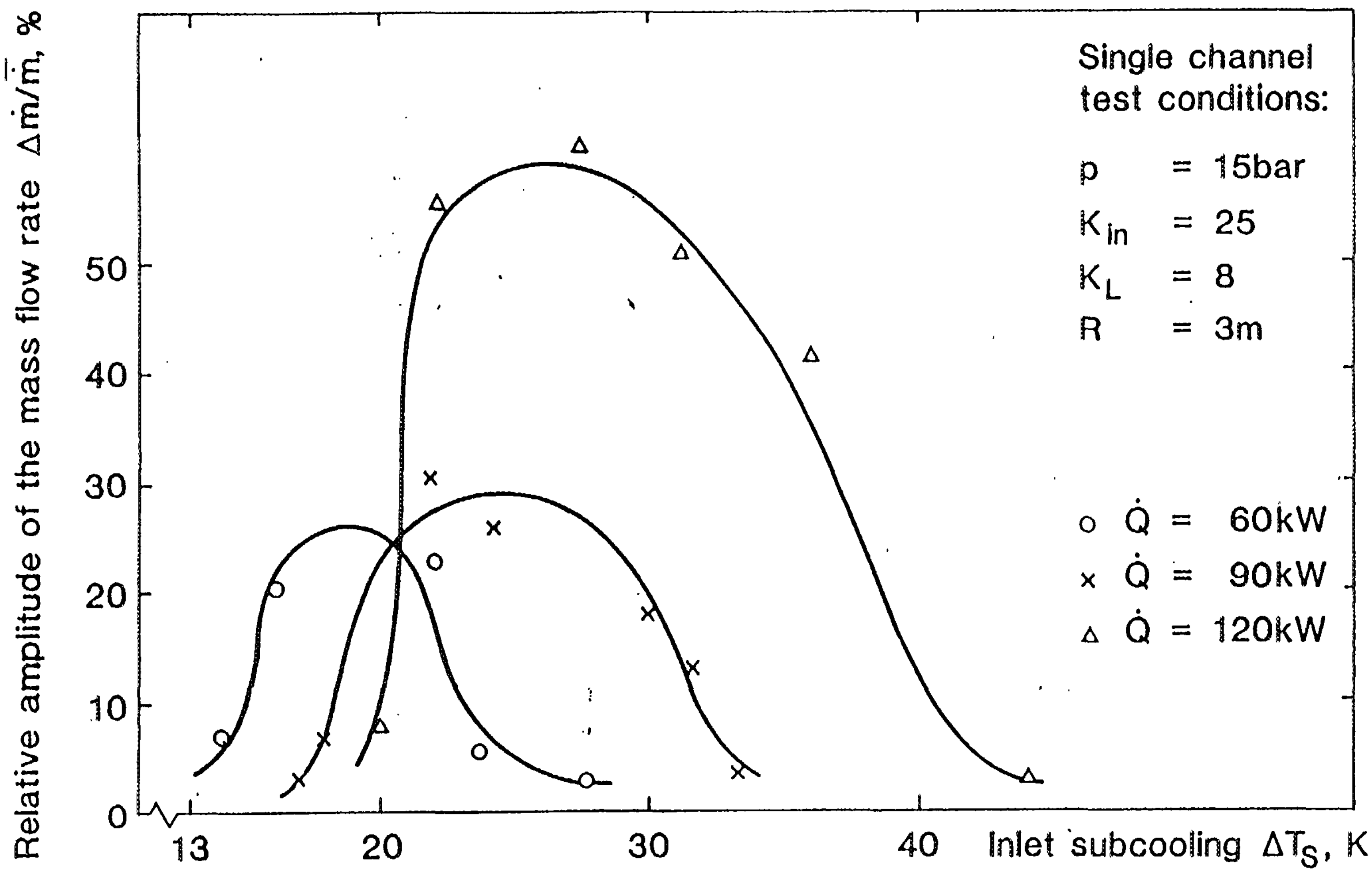


Fig. 3 Effect of the inlet subcooling on stability

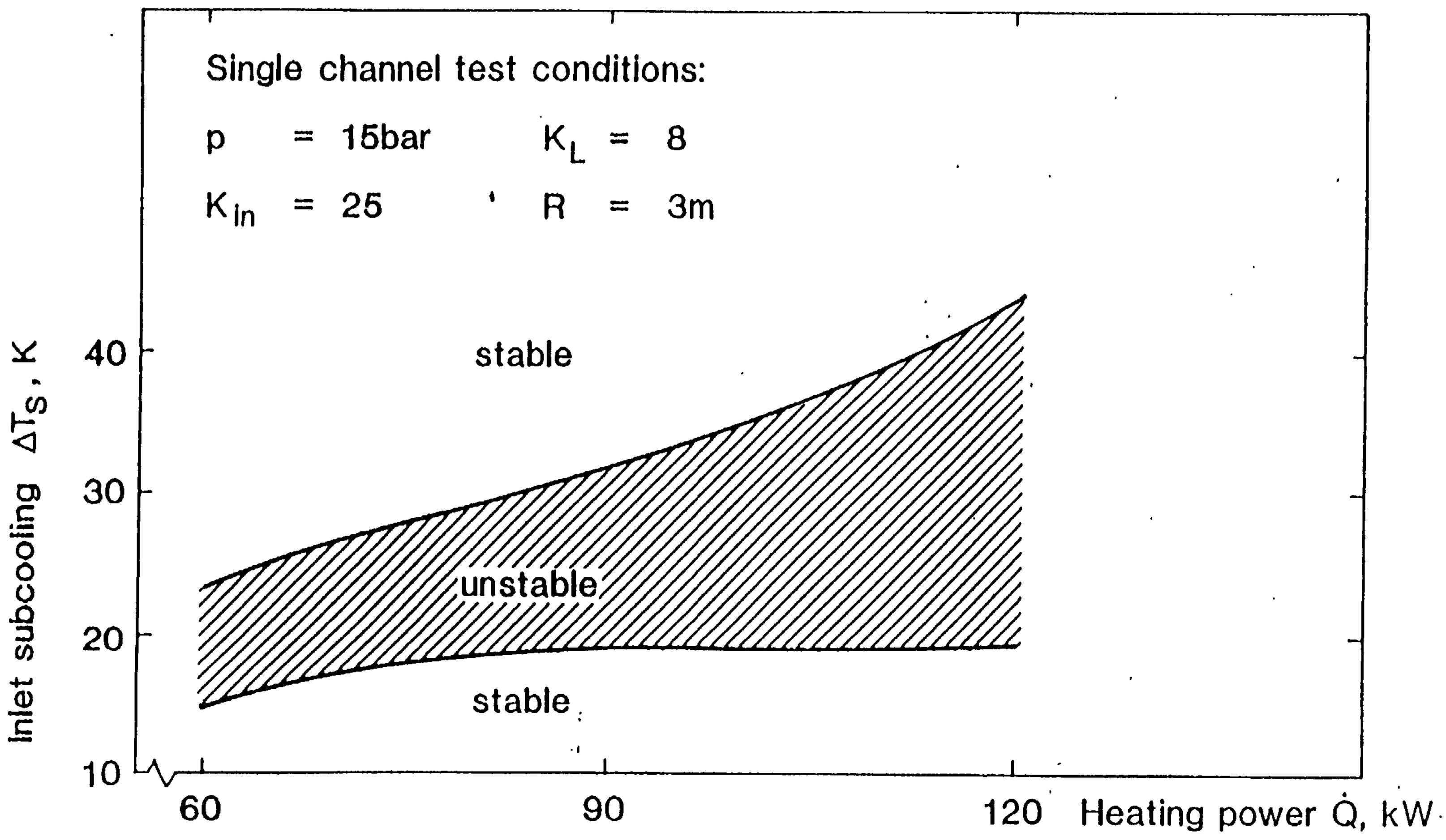


Fig. 4 Stability map of single channel tests

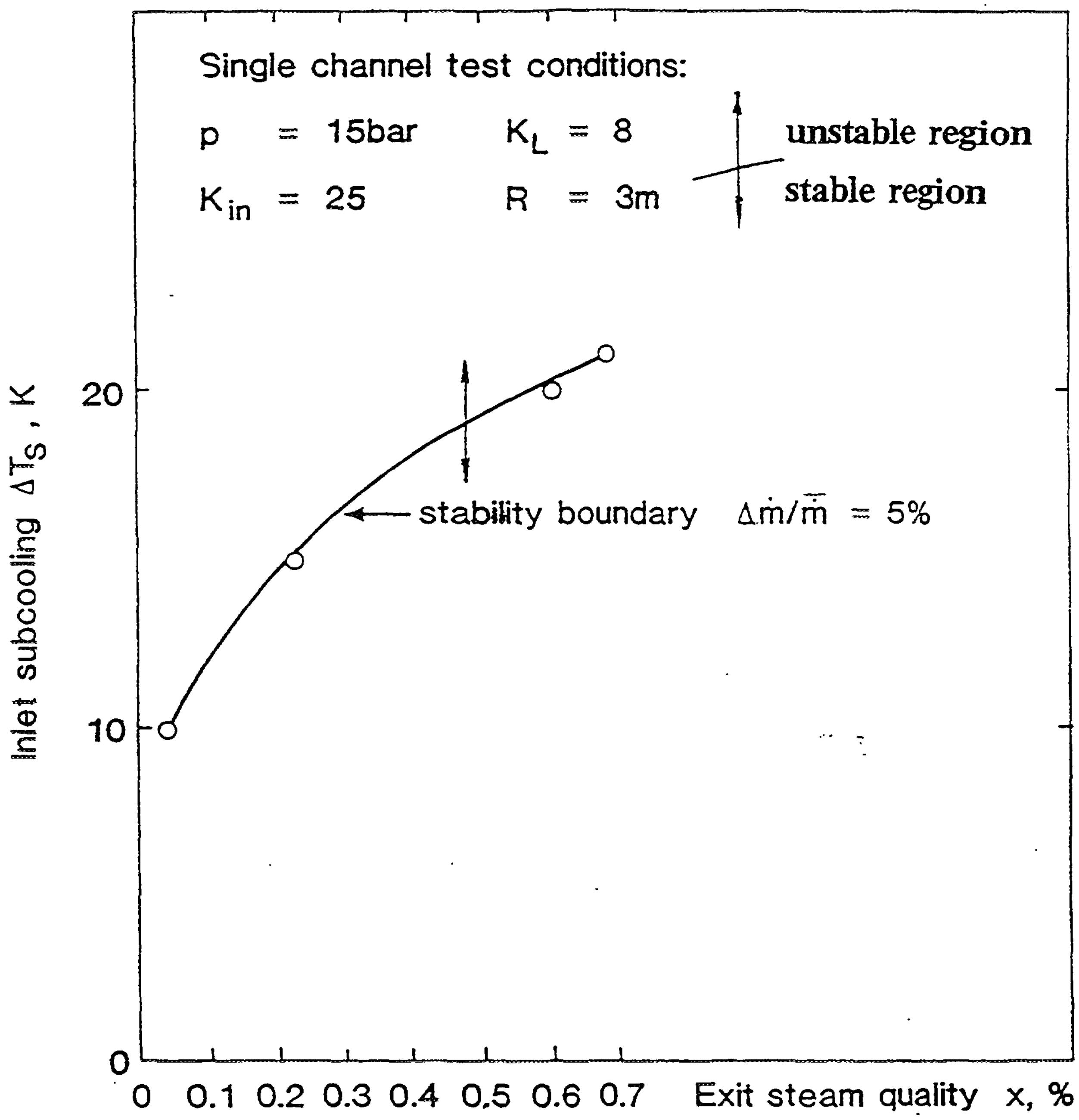


Fig. 5 Stability boundary of single channel tests

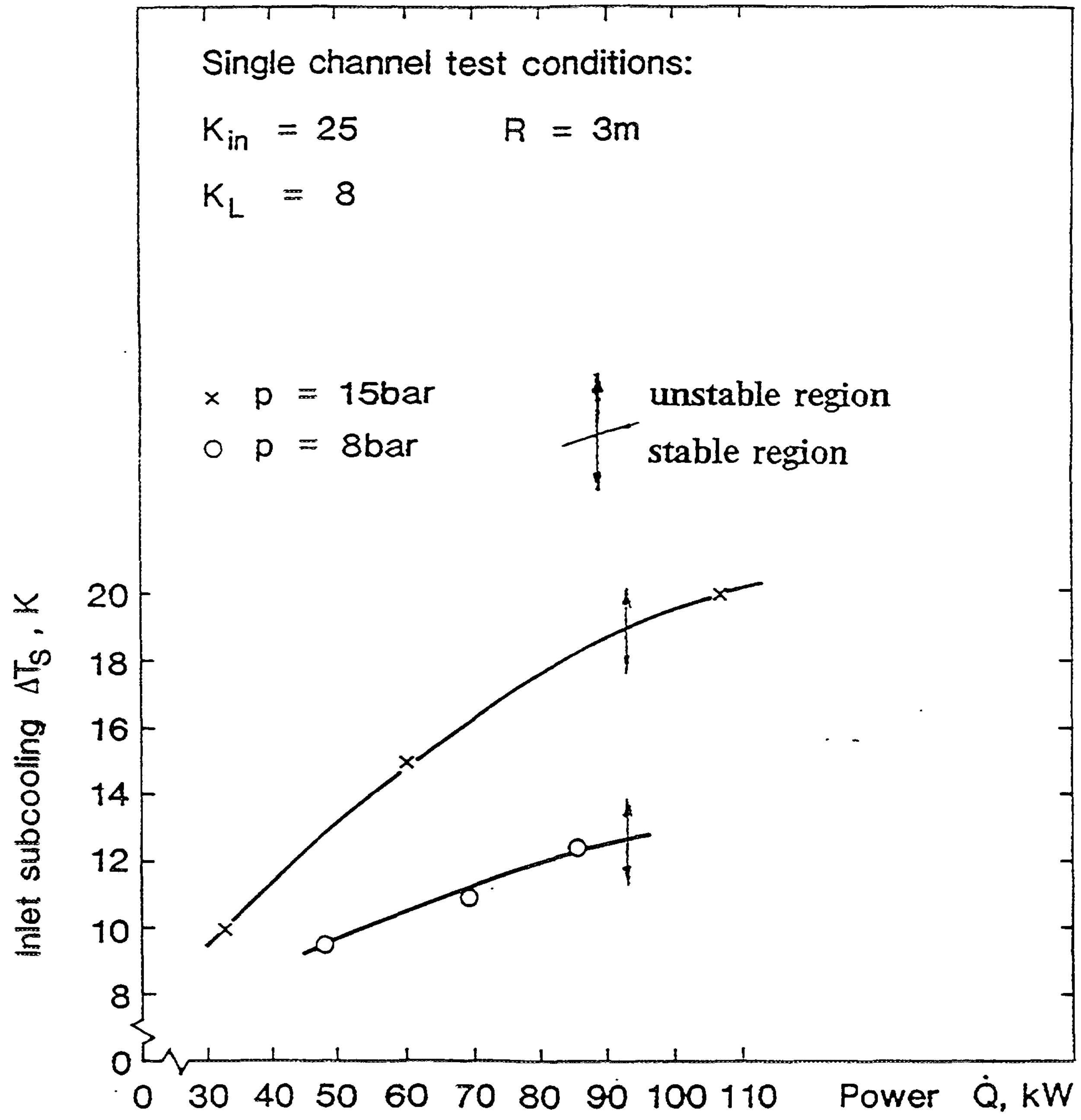


Fig. 6 Effect of system pressure on stability

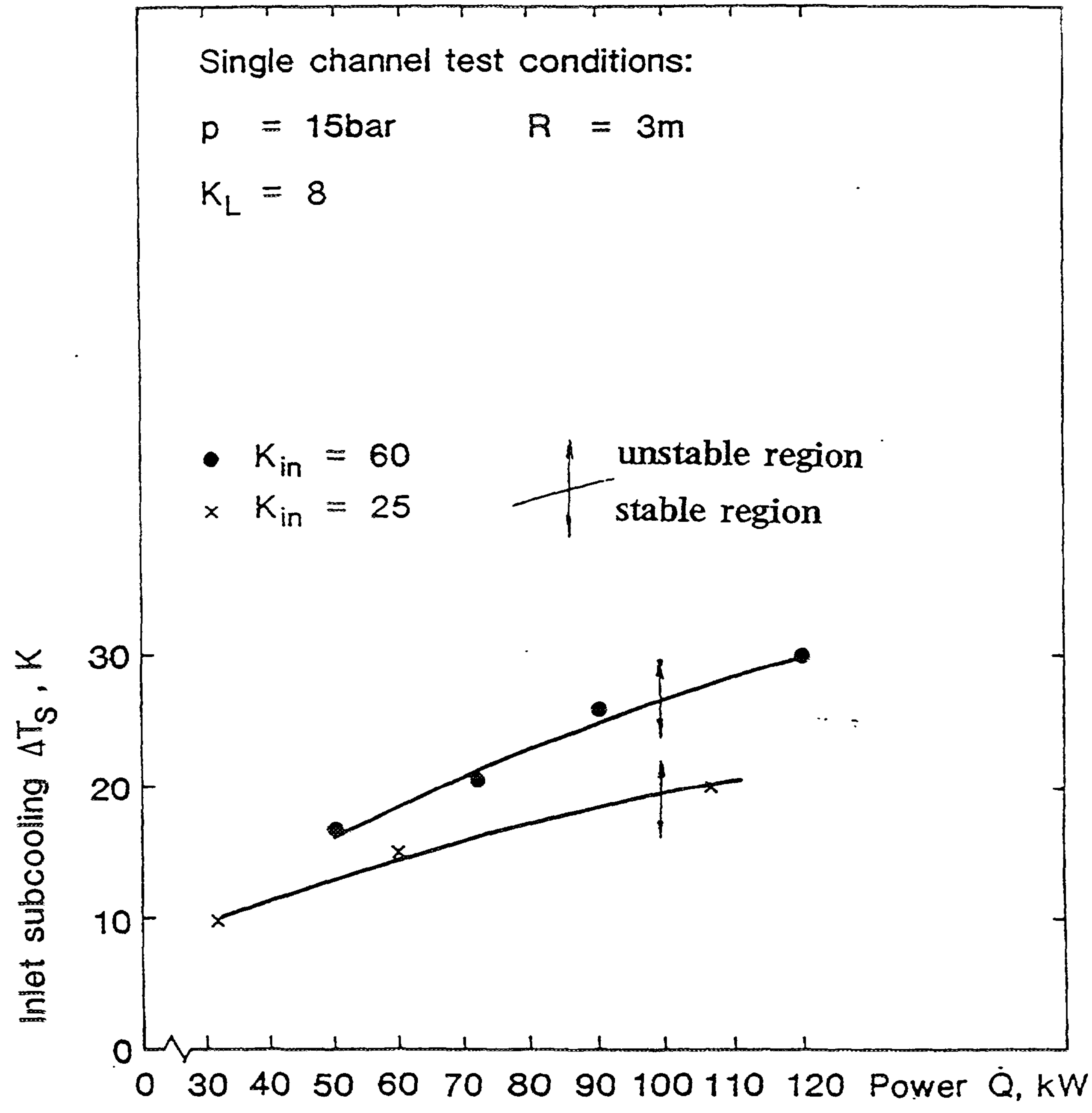


Fig. 7 Effect of the inlet resistance coefficient on the single channel stability

subcooling rate the system is stable or unstable (see Fig.5). When the exit steam quality gets greater than 1.0%, the system becomes stable.

The experimental results confirm that the exit steam quality is the dominant factor for the stability of systems operating under low pressure, low steam quality and natural circulation.

Stability of the Two Parallel Channel System:

Mass flow oscillation between the parallel channels. With identical inlet resistance coefficients in the two parallel channels and the cross connection valve closed, the heating power in the two channels was symmetrically adjusted. When the system got into an unstable condition, the oscillations of the mass flow rates in the two channels and in the loop were recorded. It was found that the mass flow oscillations in the two channels were out of phase by 180° (see Fig.2) while the main mass flow in the loop showed only very weak fluctuations. The stability boundary of the two parallel channels is shown in Fig.9 and compared with the one of the single channel. Under the same operating conditions, the single channel system is more stable. For the single channel system, the loop resistance (including subcooler and downcomer) acts as a restrictive factor on the feedback of the mass flow perturbation, while in the case of the parallel channel system without the cross connection the loop resistance does not contribute to the dampening of the mass flow oscillation between the two channels.

The effect of asymmetric heating on the stability of the two parallel channels. The heat generation of each fuel element assembly in a reactor is different depending on the radial power distribution in the core. In order to investigate the influence of the power distribution on the system stability, the following experiments were performed:

(a) Asymmetric heating with equal inlet resistance coefficient. In the experiment, keeping the heating power ratio between the two channels constant, the dependency of the relative amplitude of the mass flow rate on the inlet subcooling is measured. The results are shown in Fig.10 and Fig.11 respectively. Comparing the results of symmetric and asymmetric heating (Fig.11) it can be seen that in the condition of asymmetric heating, the system stability is mainly dependent on the stability of the channel with smaller heating power. The stability of this channel is better than that of the symmetrically heated parallel channel with the same heating power.

(b) Asymmetric heating with equal exit steam quality. In the experiment, the ratio of the heating powers and the ratio of the inlet resistance coefficients are set by the following formula:

$$Q_s/Q_n=(Kin_n/Kin_s)^{1/2}, \text{ where } Kin_n=35,\ Kin_s=15.$$

The corresponding stability map is shown in Fig.13. It is demonstrated that the stability boundary of the two parallel channels with equal exit steam quality is well-matched.

Flow Instability of natural Circulation under Lower System Pressure:

The y-T_{sin} map under lower system pressure may be divided into 6 divisions according to the oscillation mode of the mass flow rate recorded, shown in Fig.14.

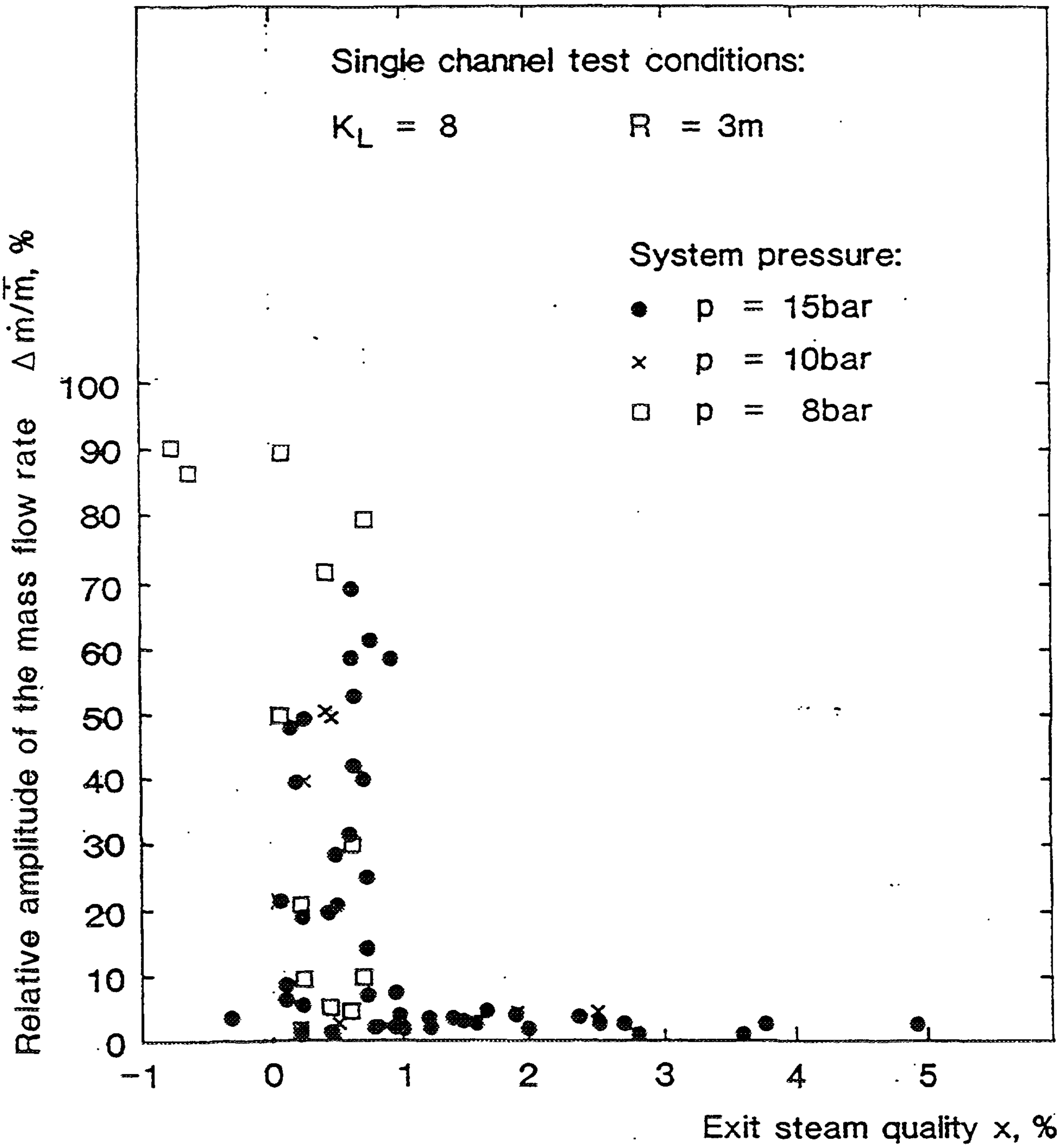


Fig. 8 Effect of the exit steam quality on system stability

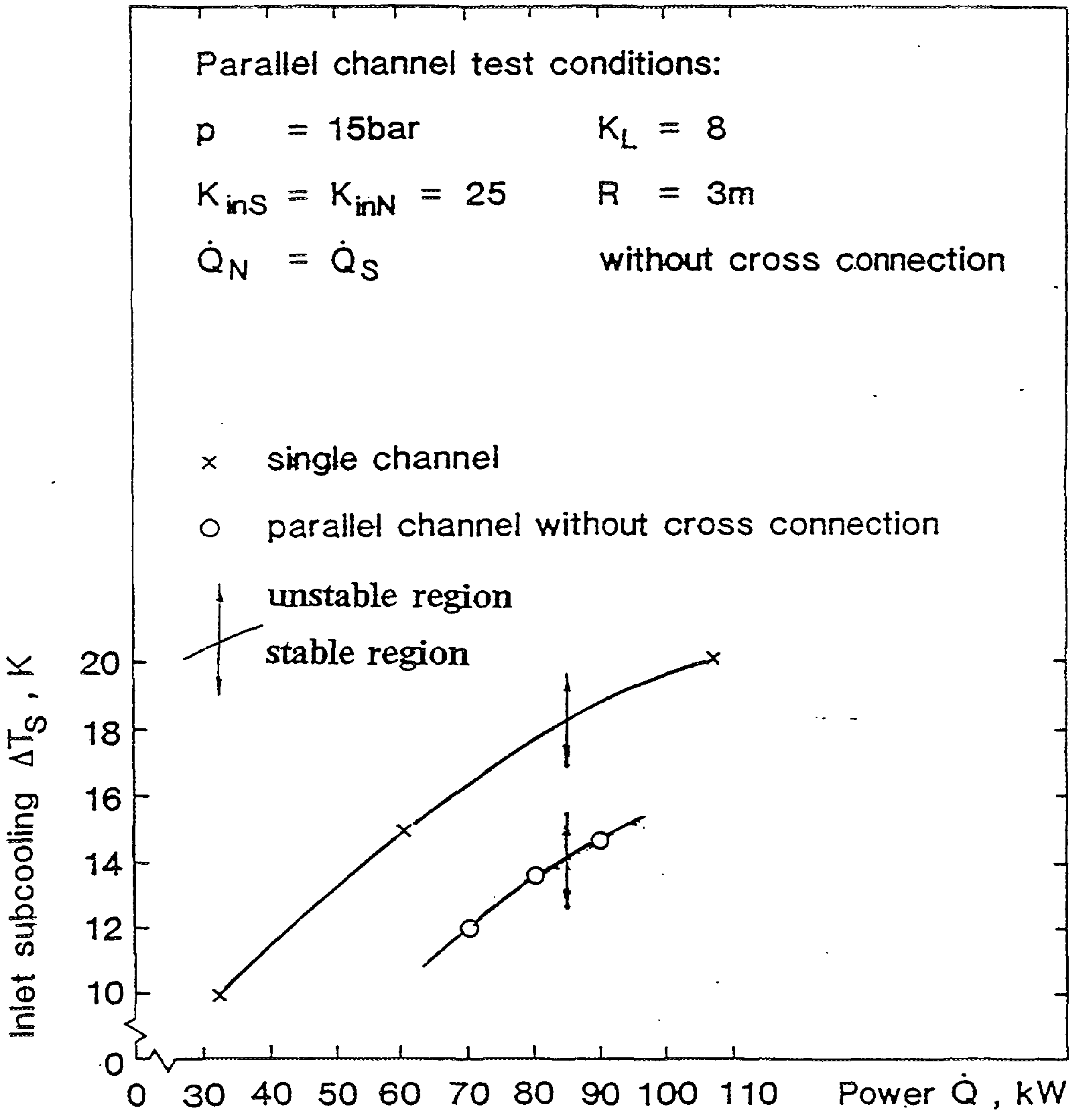


Fig. 9 Stability map of the parallel channel system without cross connection

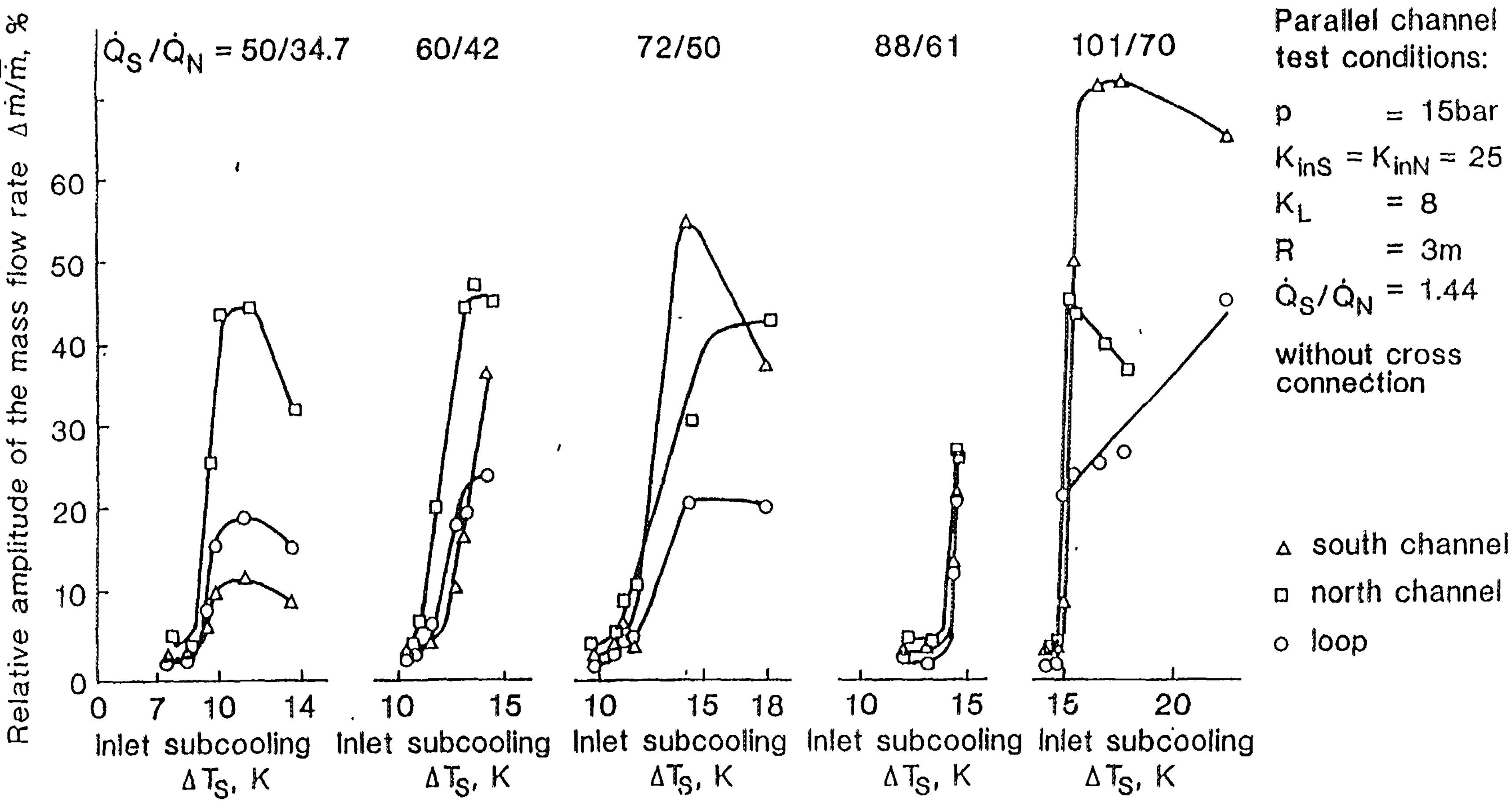


Fig. 10 Effect of asymmetric heating on the stability of the parallel channel system

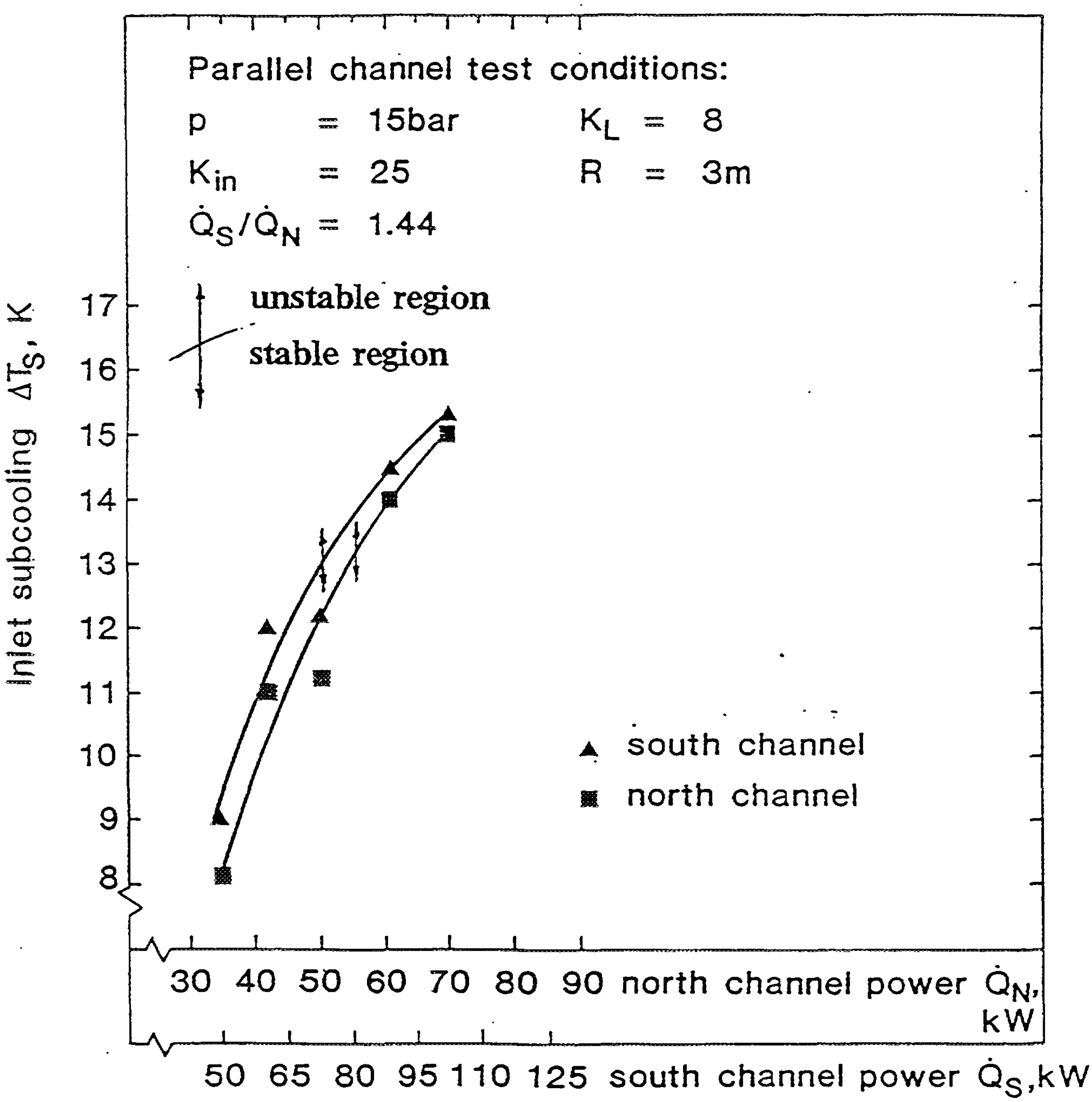


Fig. 11 Stability map of the parallel channel system with asymmetric heating

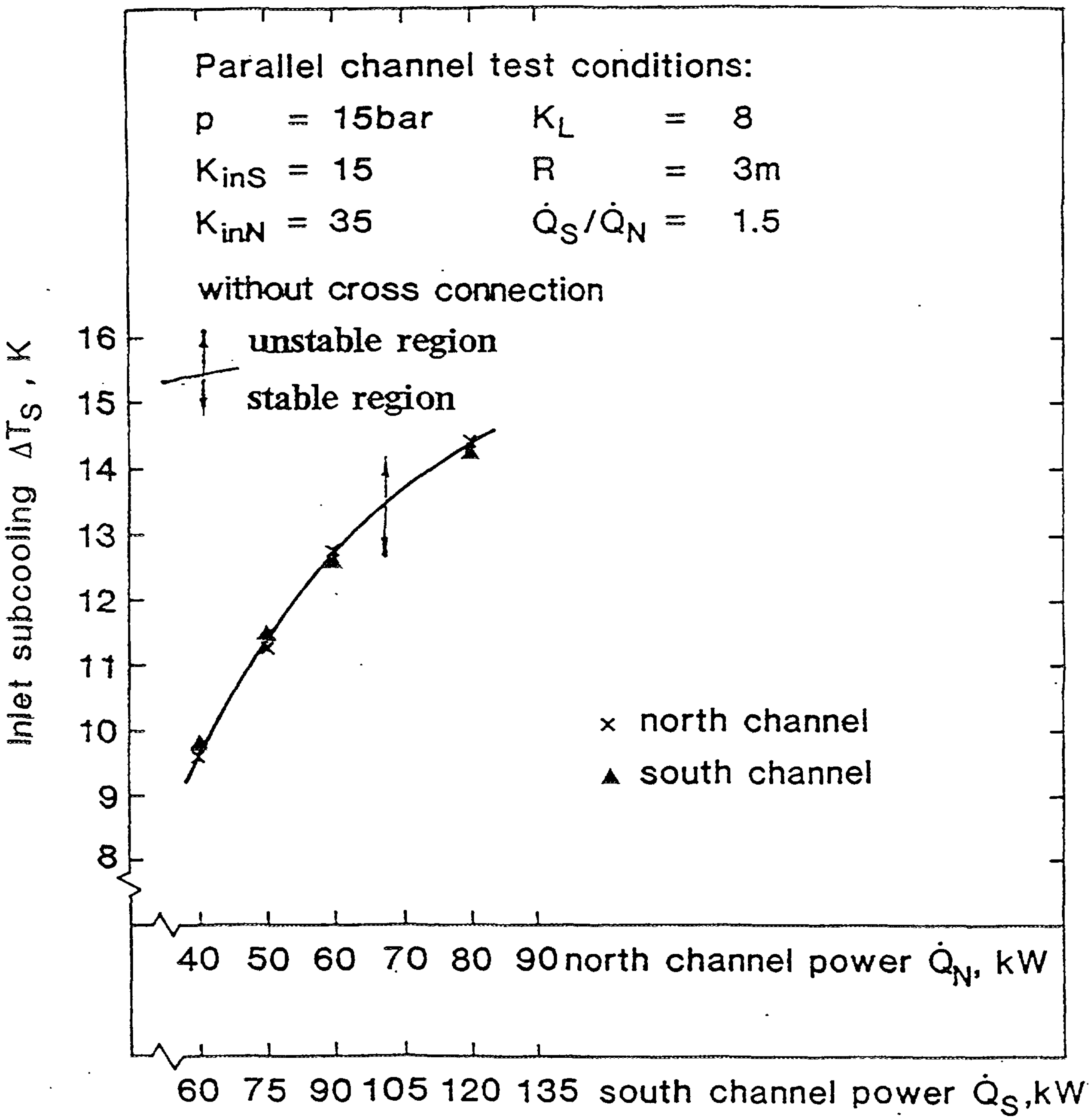


Fig. 12 Stability map of the parallel channel system with equal exit steam quality

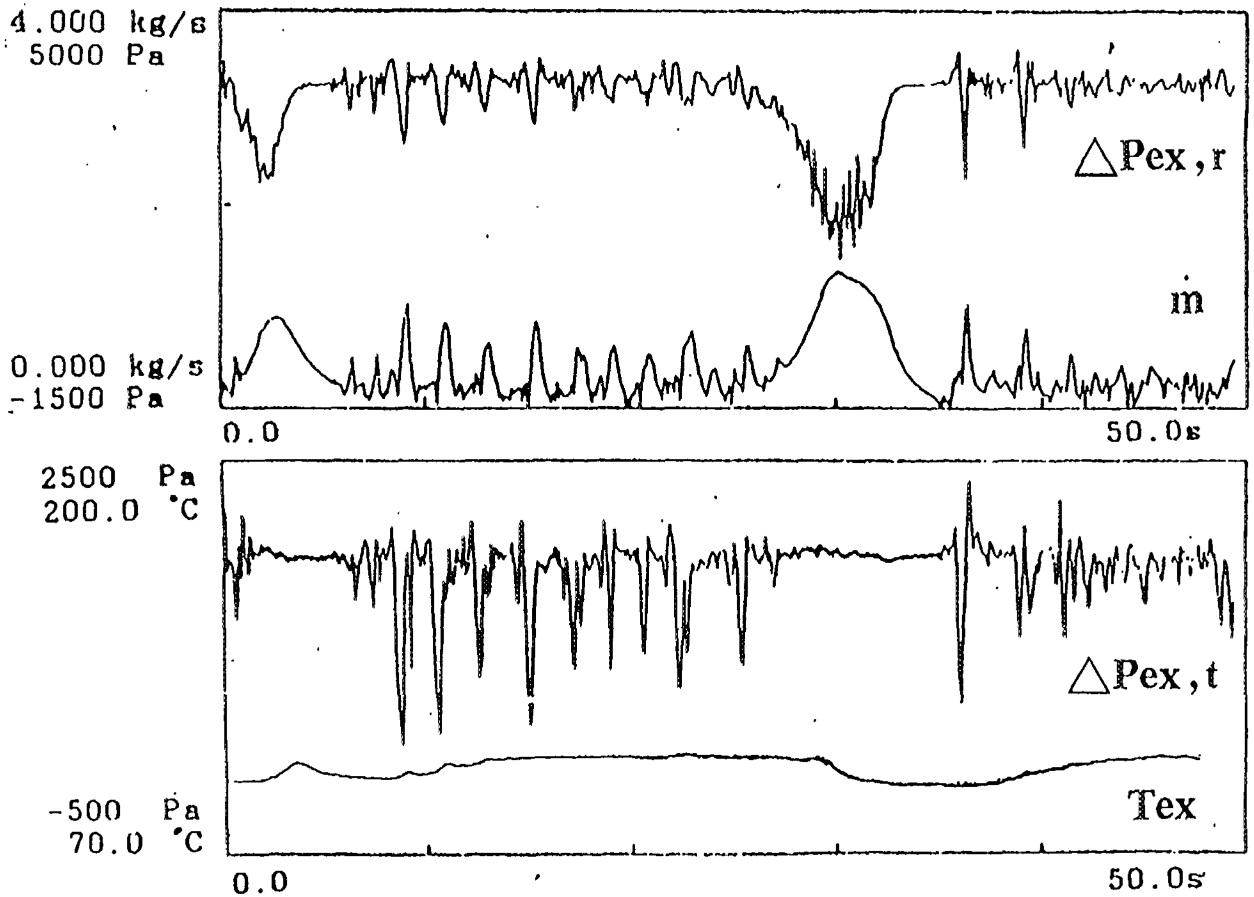


Fig. 13 Unstable flashing under atmosphere condition

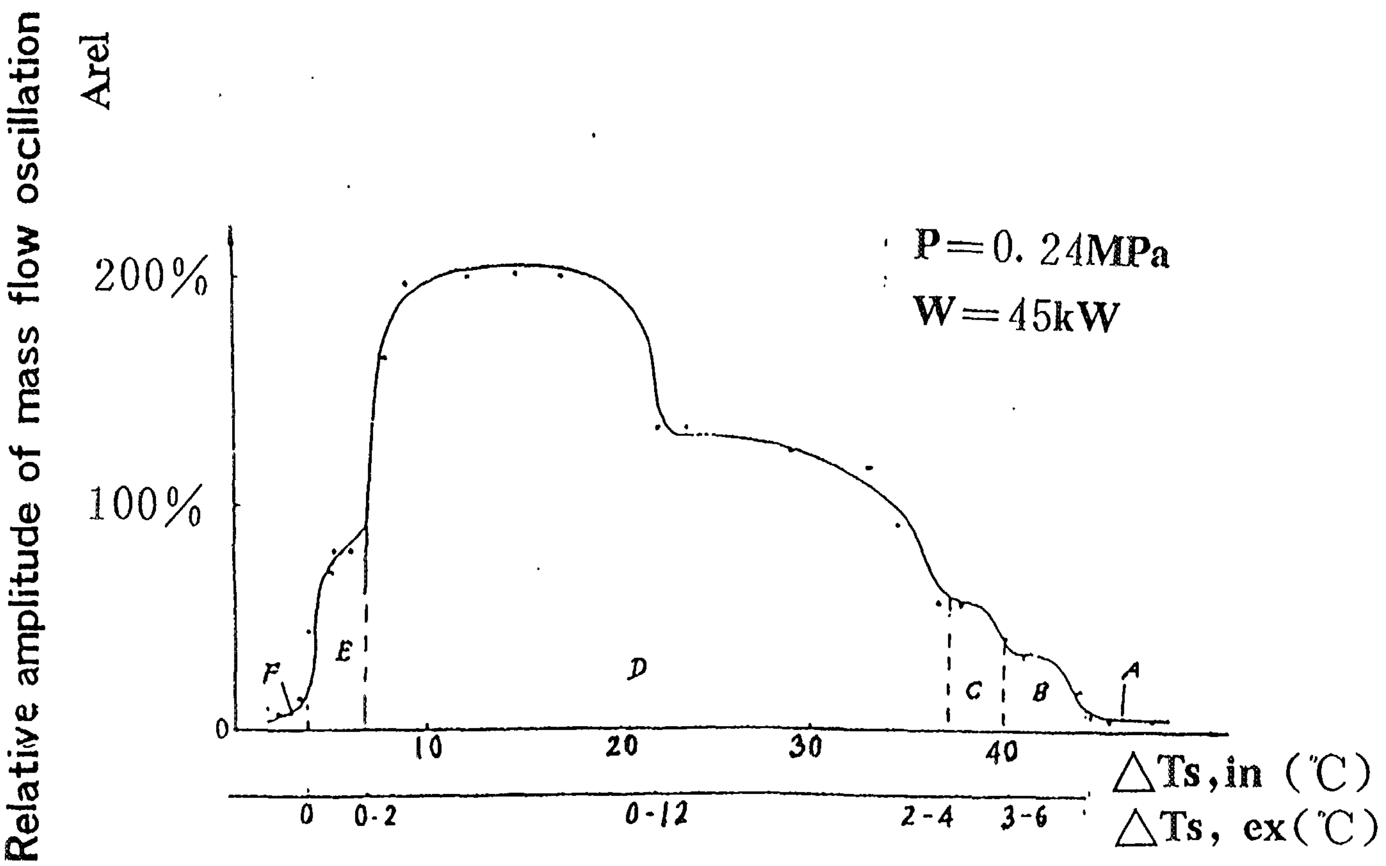


Fig. 14 Relative amplitude of mass flow oscillation for low system pressures

Division B: a division with quasi-stable flow and weaker subcooled boiling. The main stream is subcooled water with a mild fluctuation due to the appearance and condensation of steam bubbles near the surface of heated rods.

Division C: a division with higher frequency oscillation of mass flow. The main stream at the exit of the test section is subcooled. Subcooled boiling occurred at the surface of heated rods. The bubbles breaking away from the surface were condensed in a short range. The alternation of void fraction of subcooled boiling led to a flow oscillation with "higher frequency", see Fig.15.

Division D: It is a division with flow oscillation stimulated by unstable flashing in the riser. The phenomena and its mechanism may be described as one in Division C mentioned in [6].

Division E: Flow oscillation stimulated by low quality bulk boiling at the exit part of the test section was found in this division. An even amplitude and low frequency, self sustaining mass flow oscillation was recorded. The mixture of steam and water with different density flowing alternately along the riser formed a series of density wave transferring up along the riser. During the whole period the flow pattern was bubble flow.

Division F: The mass flow rate is stable with very small random fluctuation induced by stable bulk boiling in the division.

5. CONCLUSIONS

Under certain geometric conditions and operation parameters, a self-sustaining, low frequency, even-amplitude mass flow oscillation may be excited at very low steam qualities in a low pressure system with natural circulation.

The steam quality at the outlet of the heated section is considered to be a dominant factor for the hydrodynamic stability of a low quality two-phase natural circulation system at low pressure.

The experimental results have provided a set of useful reference data for the design and operation of the INET-5MW experimental heating reactor.

Further investigations of the effect of the riser height and axial power distribution on the system stability are planned and will be performed in the near future.

Intensive thermodynamic unequilibrium existing in a natural circulation system with subcooled boiling may lead to local cyclical accumulation of energy in the riser and form unstable flashing in the riser.

Flow instability induced by subcooled boiling in natural circulation system under low pressure may show in several different mode of flow oscillation including intensive random pulse-type oscillation, unstable flashing, higher frequency oscillation and density wave oscillation. Some modes of flow oscillation are restrained under higher system pressure.

Stable flow with subcooled boiling exists in a wide range of inlet subcooling under higher system pressure.

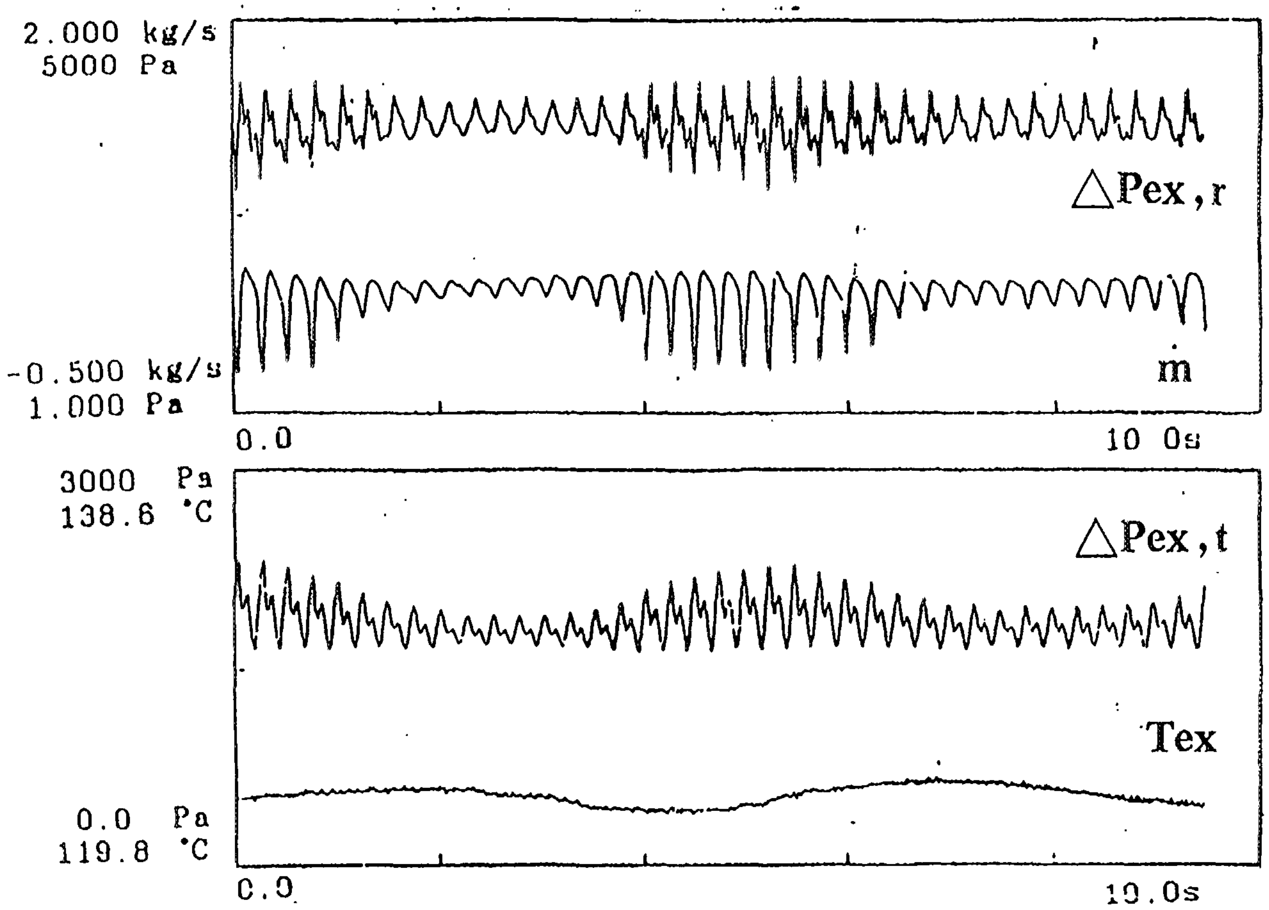


Fig. 15 Subcooled boiling—induced higher frequency oscillation

NOMENCLATURE

Arel	Relative amplitude of the mass flow rate oscillation, %
Kin	Resistance coefficient at the test section inlet, dimensionless
Kin_n	Resistance coefficient at the inlet of the north test section, dimensionless
Kin_s	Resistance coefficient at the inlet of the south test section, dimensionless
K_L	Loop resistance coefficient, dimensionless
Q	Heating power, kW
Q_n	Heating power of the north test section, kW
Q_s	Heating power of the south test section, kW
R	Height of the riser, m
m	Time-averaged mass flow rate, kg/s
p	System pressure, bar
x	Steam quality at the test section outlet, %
Ts	Inlet subcooling, K
Δm	Peak-to-peak amplitude of the mass flow rate oscillation, kg/s
T	Oscillation period, s

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THEORETICAL STUDY ON THE FIRST KIND OF DENSITY WAVE INSTABILITIES

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Abstract

The present paper summarizes the theoretical studies carried out by INET (Institute of Nuclear Energy Technology) of Tsinghua University on the first kind of density wave instabilities (DWIs) of natural circulation systems. The analysis methods of DWI and mathematical models of drift flux are presented. Based on the general excess entropy production criterion of non-equilibrium thermodynamics, an energy principle of DWI is established.

1. INTRODUCTION

Over the past thirty years two-phase DWI has been one of the important problems in the heat transfer field. It usually exists in many fields such as in boilers, nuclear reactors and steam generators. It may negatively affect mechanical vibration, thermal fatigue and heat transfer process [1], etc. Most of previous work was devoted to DWI with relatively high exit quality [1]. This kind of instabilities, caused by the resistance of two-phase flow systems, is referred to as the second type or resistance-type instability, and can be found in steam generators and BWR systems. Another kind of instabilities, so called the first type or gravity-type instability [2] caused by the void and flow oscillations in a low pressure and low quality natural circulation system, was investigated in Germany, Russia, Switzerland and China. With the development of NHRs, the first kind of DWI seems more and more important [3]. Since 1980 the Institute of Nuclear Energy Technology (INET) has been systematically carrying out theoretical and experimental studies on the first kind of DWIs. This paper summarizes the theoretical work as follows:

- Development and modification of analysis methods;
- Systematic study of parameter effects, comparisons of theoretical and experimental results, and determination of instability boundaries;
- Study of nonlinear oscillations;
- Study of DWIs during transients;
- Study of physical mechanisms of DWIs based on the energy principle of non-equilibrium thermodynamics.

2. STUDY OF THEORETICAL MODELS AND ANALYSIS METHODS

Theoretical analysis of DWIs starts from describing the conservation equations of the two-phase flow system [4], then the system response to an external disturbance is obtained by using numerical methods, and the threshold of occurrence of unstable flow is determined. The factors that induce and influence unstable flow are known through theoretical analysis. Analytical results are synthesized with those from a dimensionless criteria and verified by test data from a test loop. In this way a quantitative criterion correlation of a stable boundary is obtained.

The mathematical models of two-phase flow of a natural circulation system include mass, momentum and energy conservation equations. For fuel elements in the reactor core, the heat conduction model of the heated fuel elements, and the neutron kinetic model which reflects nuclear power and void feedback, still need to be considered. Subcooled boiling and two-phase drift flux models are considered. The point neutron kinetic model with section weight is applied in the analysis.

Time-domain and frequency-domain methods are the two different approaches which are usually utilized to analyze DWIs. In the time-domain mode, the change pattern of variables is obtained through solving the equations described above. When an external disturbance is given, a decay ratio is defined to be used in describing divergence or convergence of the variables against time. If the oscillation amplitude of the variables declines against time, the decay ratio DR is less than one and the system is said to be stable, otherwise the system is said to be unstable.

In the process of analysis, drift flux models and nonlinear methods are utilized to study the flow stability. There is only little similar research work in recent years in the world. Because the first type density wave oscillation appears at the border region between low pressure single-phase and two-phase flow [5], several subcooled boiling models and drift flux correlations have been adopted [6] so that the first kind of DWIs can be analyzed in a low pressure and low quality system with natural circulation.

Subcooled boiling models employed include:

- Lellouche & Zolotar model
- Saha model
- Levy model
- Griffith model
- Bowring model
- Saturated boiling model

Drift flux models utilized include:

- Levy model
- Lahey model
- Dix model
- EPRI model

In the frequency-domain mode, the two-phase flow conservation equations are linearized and subjected to a Laplace-transform with the assumption of a minor disturbance; then the open-loop transfer function of the system is evaluated. If the root trace of the characteristic equations contains the point (-1,0) in the Nyquist chart on the complex plane, it means that the solution σ of the equation has a positive real part and the system is

$$Y = (z,t) = X(z)\, e^{\sigma t}$$

divergent. The frequency-domain method is accurate when it is used to predict the stability boundary. Although it is only suitable for a linear system, it is extensively applied in the world today to study stability because it needs less computing time. A typical frequency-domain analysis program is out of the NUFREQ series codes. A natural circulation flow module of drift flow is developed and added into the NUFREQ [6] code in the process of the study of this topic.

The applicability and reliability of the analysis models of RETRAN-02 and NUFREQ have been verified by comparison of theoretical results with those from a stability test loop (Fig.1) and NHR-5 (Fig.2). The thresholds of instability determined by NUFREQ, RETRAN-02 and experiments of the test loop are drawn out in Fig.3. NUFREQ results agree well with the experimental data, their difference is no more than 2°C. The comparison of RETRAN-02 results and the test loop indicates that calculation values and test results are almost identical in the operation range of the test loop, their difference is less than 2°C [7].

2. STUDY OF DWI THRESHOLD AND ITS EFFECT FACTORS

Time-domain, frequency-domain and criterion methods were used to analyze the instability behavior of the test loop. A dimensionless stability map was obtained (Fig.3). The results from the test and those from the three analytical methods are in good agreement with each other.

The effects of operating pressure, inlet resistance, riser height, two phase models, parallel channel and neutron feedback on DWI threshold were investigated theoretically [8] and experimentally (Fig. 4,5,6,7,8 and 9). It is obvious that increasing pressure or inlet resistance or reducing riser height is advantageous to the system stability. In Fig.7 the impact of subcooled boiling models on the stability of the 200 MW Nuclear Heating Reactor (NHR-200) system was studied. The results from the homogenous boiling model agrees well with those from Levy, Bowring, Griffith, Saha subcooled boiling models and differs from Lelloche & Zolotar slightly in decay ratio and stability threshold, but largely in phase and frequency. Fig.8 shows that the stability performance of a parallel channel system differs from that of a single one. As indicated, instability of a parallel channel system can be avoided by reducing the riser height [1]. The effects of coupled void-power feedback on stability is dependent on the phase difference of the heat transfer in the fuel rods.

3. STUDY OF NONLINEAR OSCILLATIONS OF DWI

In a two phase unstable region, a small perturbation may excite divergent oscillations which can be classified as two kinds: infinitely increasing and finitely increasing oscillations.

It is found that in low pressure, low quality and natural circulation systems the increase of the oscillation amplitude or quality makes the dependency between void fraction and quality greatly weakened and void becomes insensitive to a disturbance of the enthalpy wave of the system. The nonlinear feature restrains the connection of the void and the flow. Because of the nonlinear feature between void fraction and quality as well as between flow rate and resistance, the system expands toward stable side and the divergent oscillation transforms into a finite amplitude oscillation.

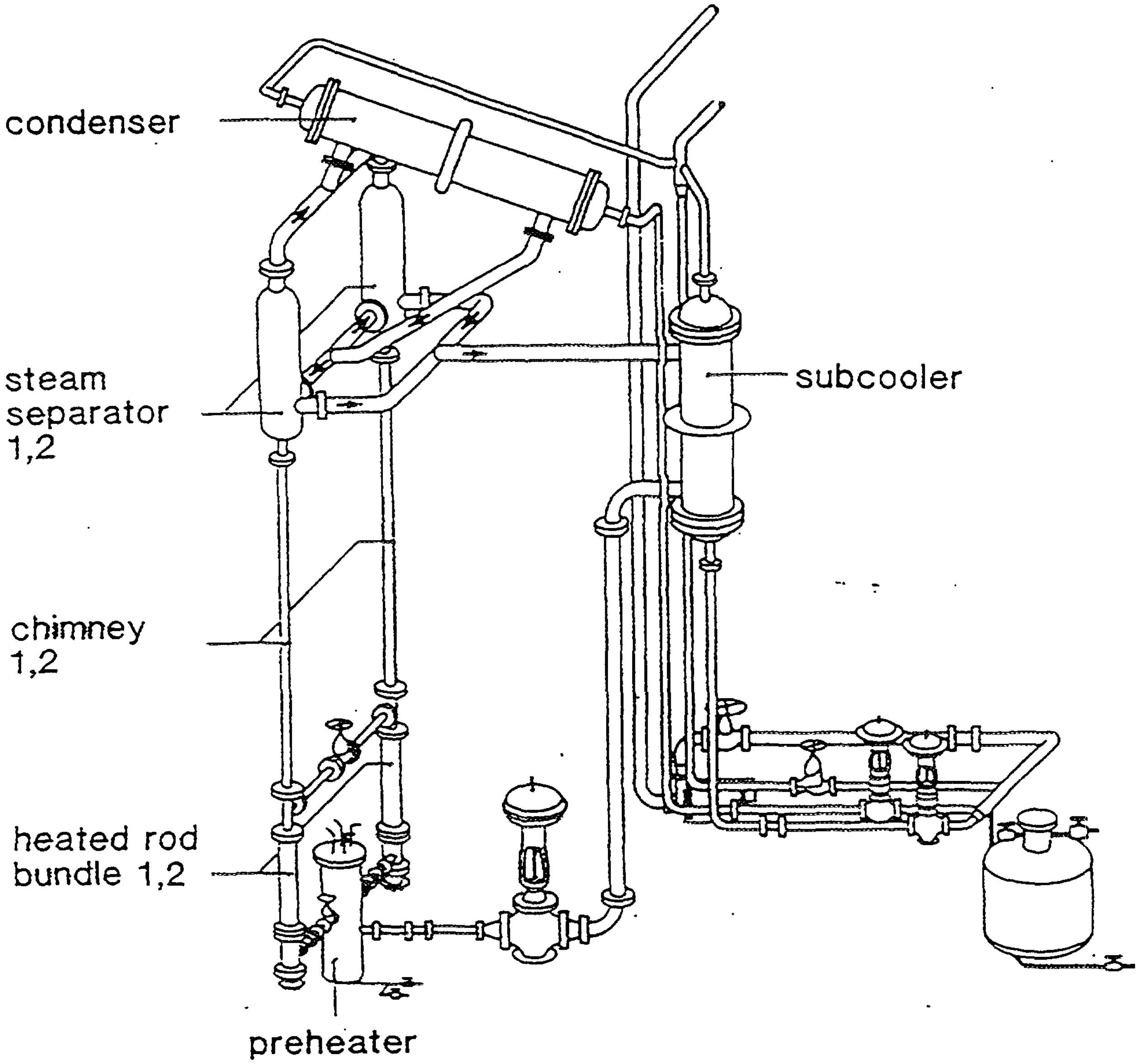
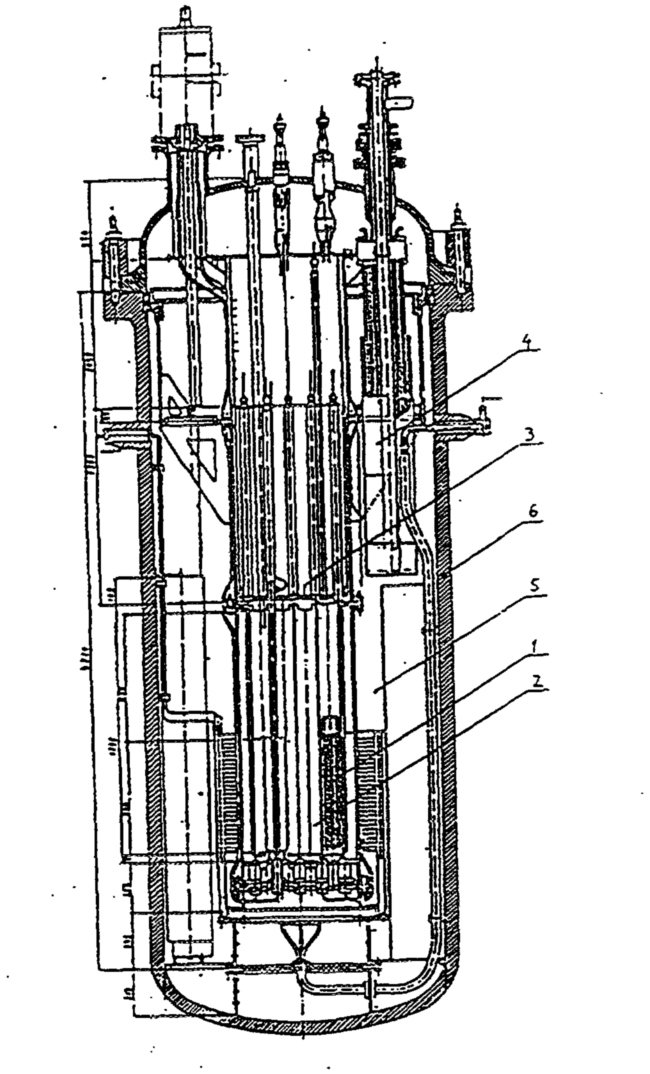


Fig. 1 Thermohydraulic stability test loop



1. fuel assembly　2. core　3 chimney
4. heat exchanger　5. downcomer　6. vessel

Fig.2 The NHR-5

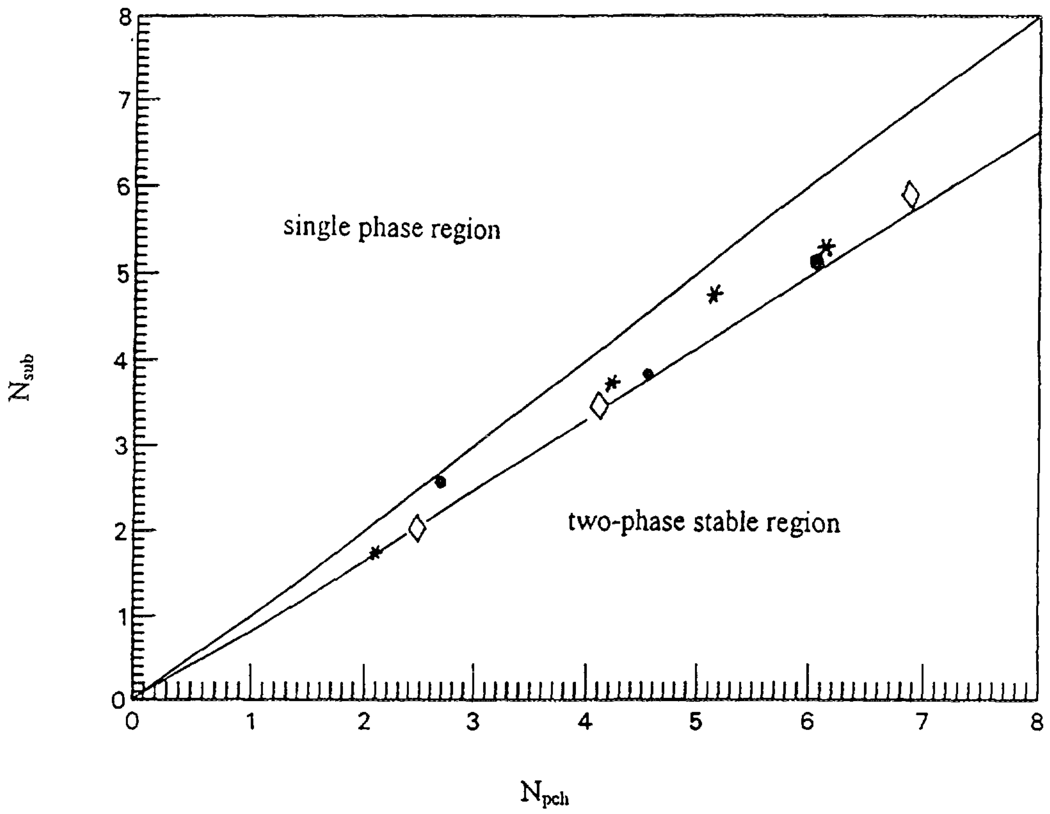


Fig.3 Dimensional stability map

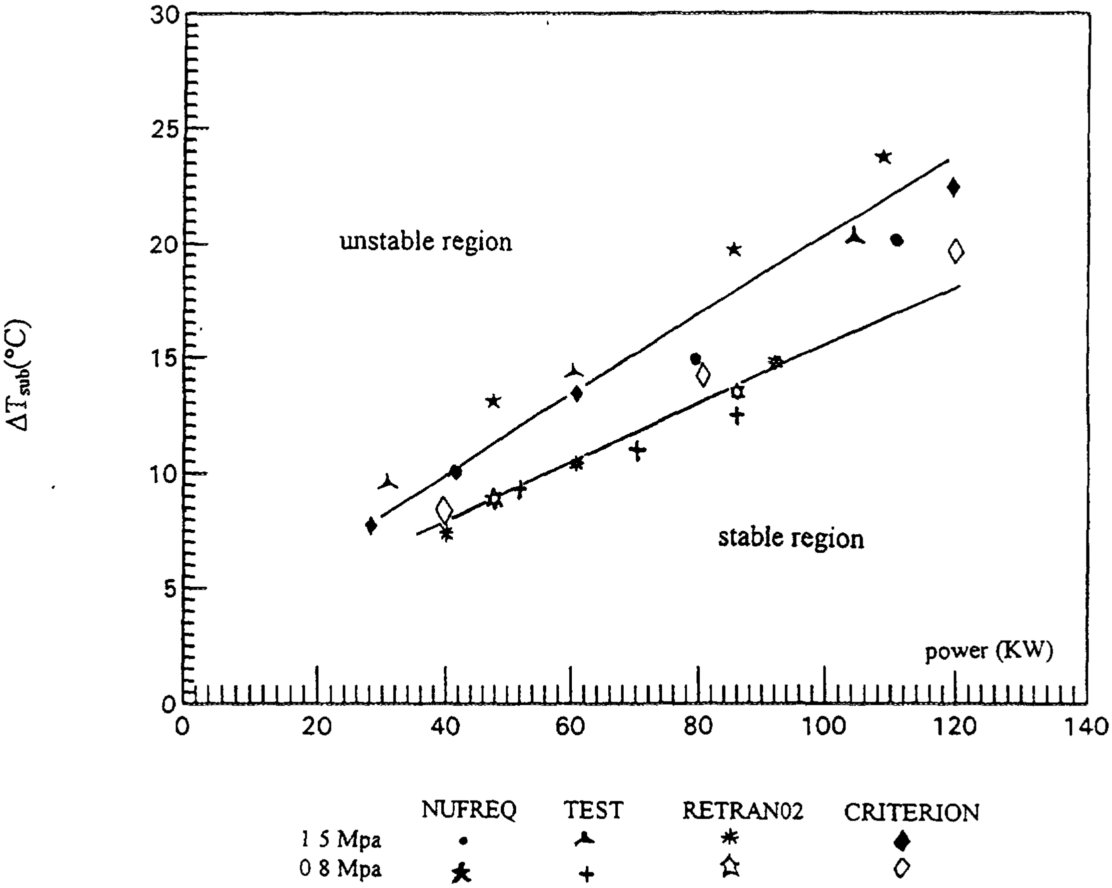
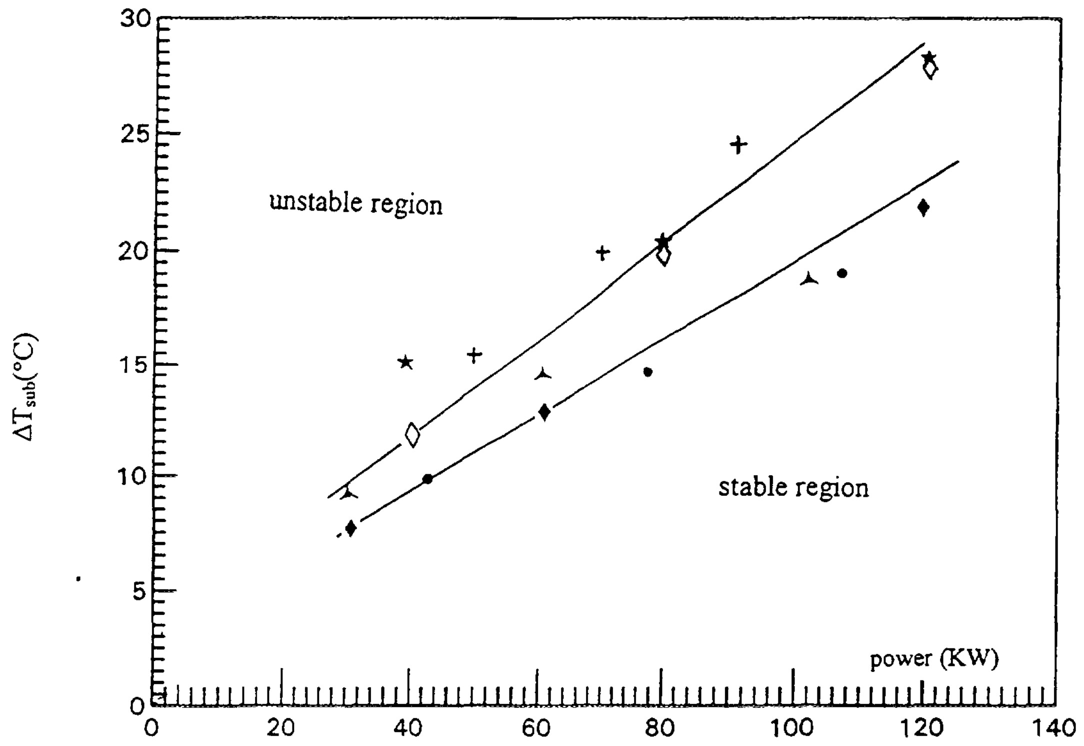
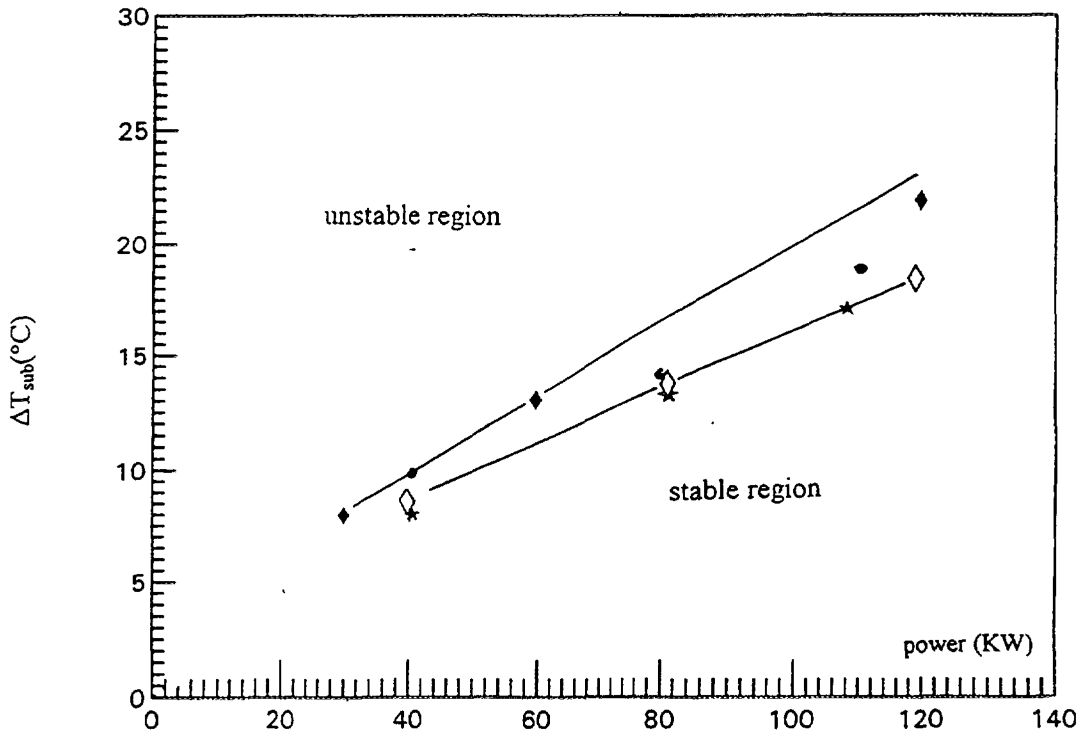


Fig 4 The effect of pressure on stability boundary



	NUFREQ	TEST	CRITERION
Kin = 25	•	⅄	◆
Kin = 60	★	+	◇

Fig.5 The effect of inlet resistance on stability boundary



	NUFREQ	CRITERION
H = 3 m	•	◆
H = 4 m	★	◇

Fig 6 The effect of riser height on stability boundary

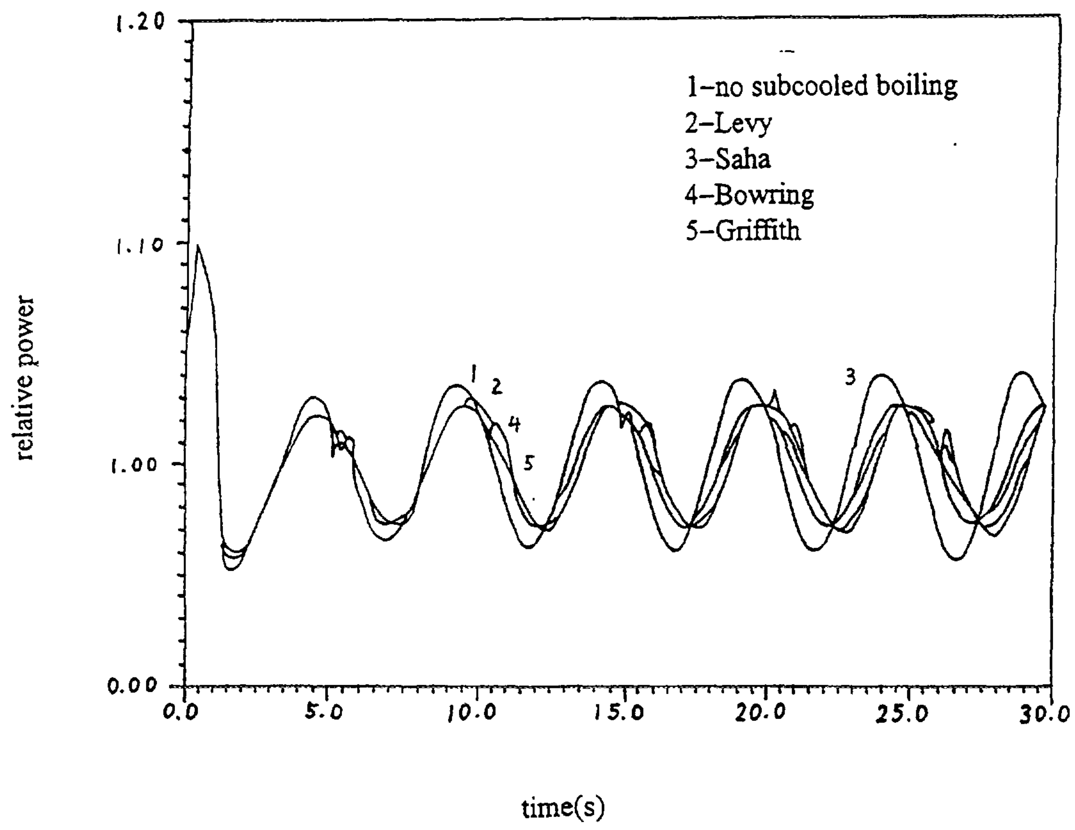


Fig.7a The effect on subcooled boiling models on stability boundary

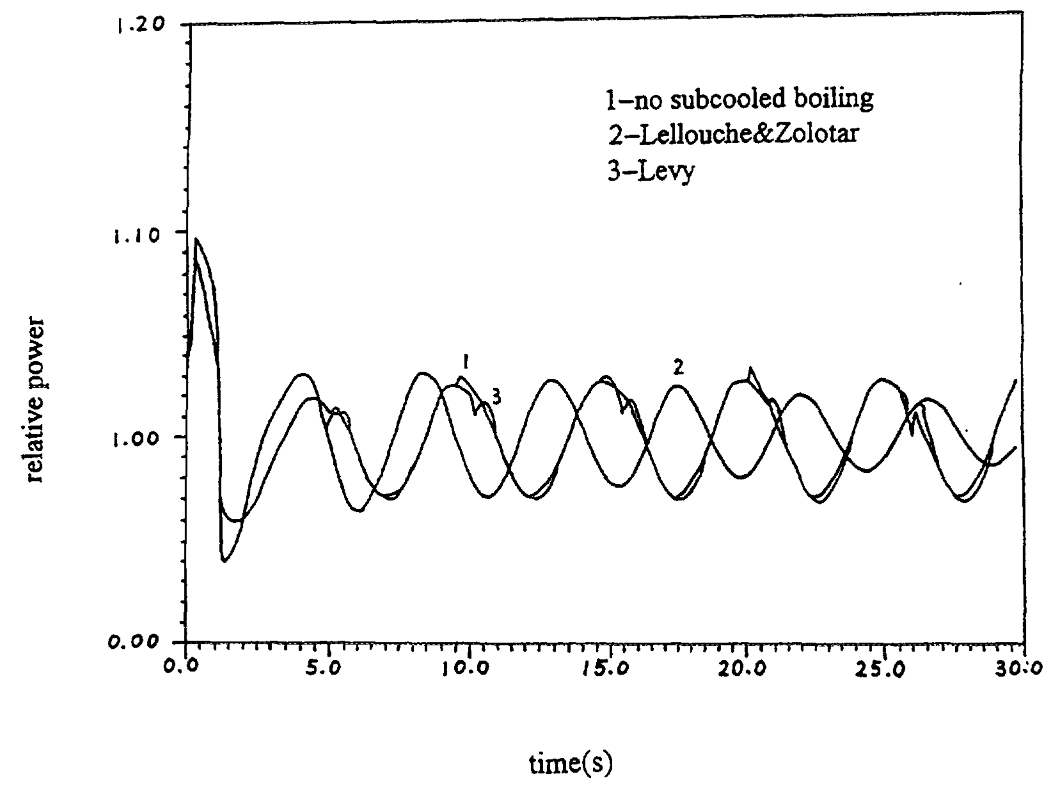


Fig.7b The effect on subcooled boiling models on stability boundary

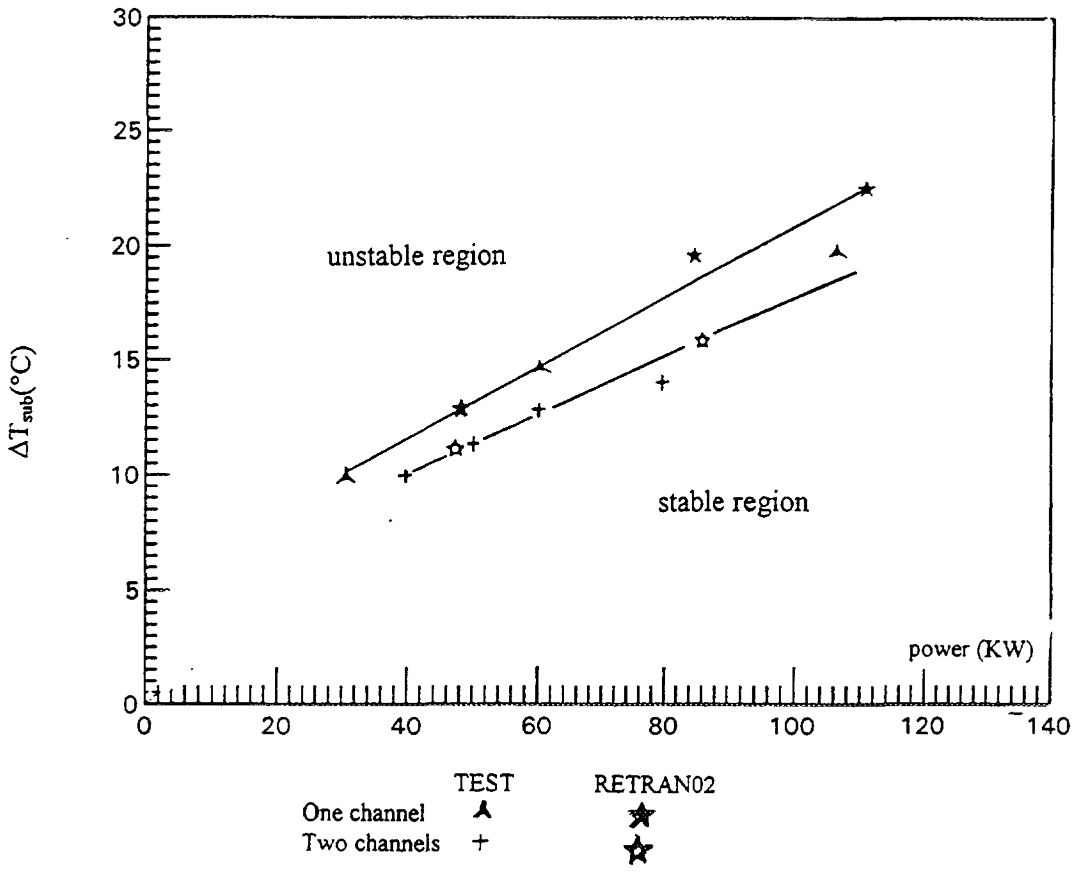


Fig.8 The effect of parallel channels on stability boundary

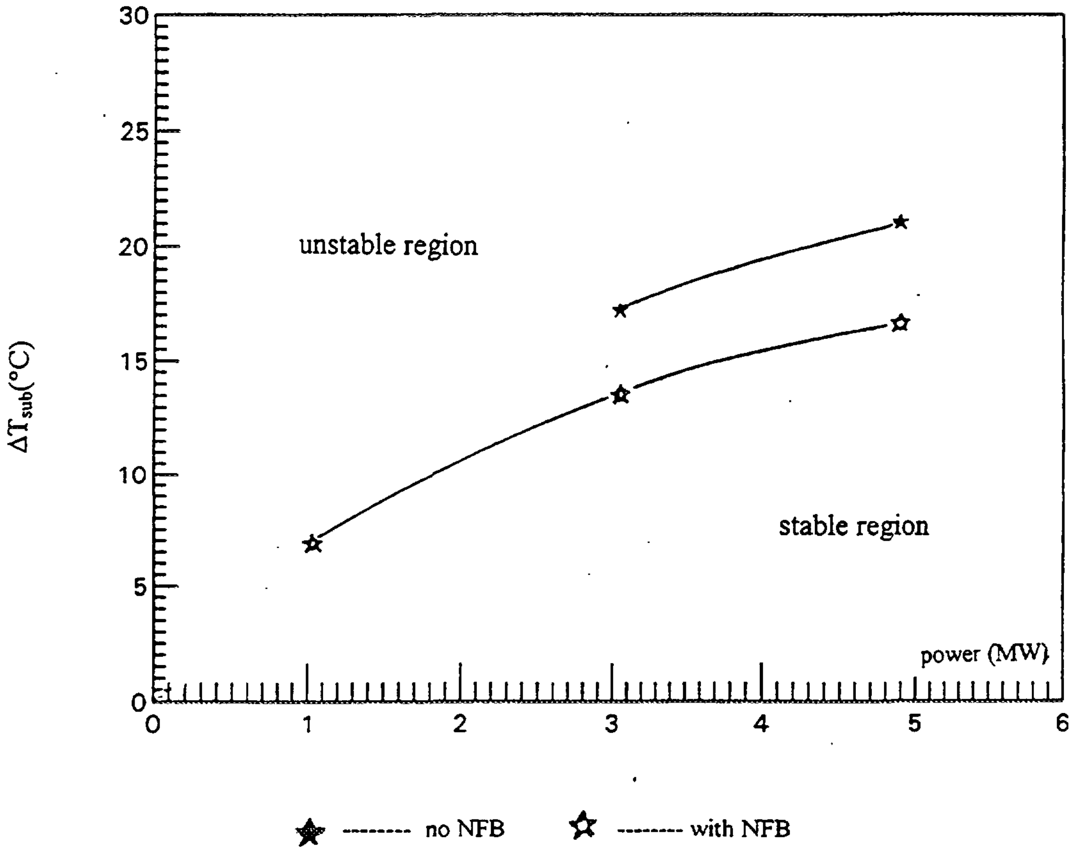


Fig.9 The effect on neutron feedback

The regularity of two-phase flow density wave oscillation has been brought to light by means of studying the nonlinear oscillations of DWI of NHR-5 and some conclusions are drawn [9]:

- In the unstable region a limit amplitude perturbation may make the two-phase flow system oscillation divergently but it tends to finite oscillation with the increase of the amplitude, and the farther the system deviates from the stable boundary, the larger the limit amplitude is.

- The frequency, which is about 0.5 Hz, is independent of the amplitude, and the period of oscillation is approximately two times that the fluid passes through the reactor core.

- The finite oscillation have two distinct kinds, equi-amplitude and vari-amplitude oscillations (Fig.10 and Fig.11). The former occurs in the vicinity of the stable boundary. This kind of oscillation is also called the limit circle oscillation. The latter, density wave coupled with compressible wave oscillation, has much larger amplitudes and greater danger to the system. It is the latter kind of DWI in some serious condition that shows an intermittent form.

- Vari-amplitude oscillations are largely affected by the size and mass inertia of the upper compressible space. A situation of increasing the water mass of the riser indicates that the stability of the system is enhanced.

4. STUDY OF DENSITY WAVE INSTABILITY DURING TRANSIENTS

DWI may occur when a stable operated system enters the instability region during transient processes. It is very complex due to the strong coupling between neutrons and thermal hydraulics. Study of this problem would be advantageous to deepening our understanding of phenomena and mechanisms of DWI, and to improving reactor safety.

The nuclear heating reactor, as a natural circulation system, can be operated under either single phase or slightly boiling modes. The transient analyses showed that DWI may take place at both conditions during loss of heat sink accident, but their characteristics and effect factors and different. It is found that the oscillation can be avoided by improving the system structure and operating parameters.

4.1. DWI during transients from single phase flow condition

DWI may occur when a stably operating system enters the unusable region during transient processes. Study of this problem would be advantageous to deepening our understanding of phenomena and mechanisms of DWI, and improving reactor safety. Transient analysis shows that DWI could take place at both single phase and slightly boiling conditions during a loss of heat sink accident [10].

Under single phase condition, the system operates in steady state at first. After a loss of external power accident occurs, the primary circuit loses its heat sink, making the heat accumulate in the heat exchanger and the flow rate decreases. On the other hand, because of the decrease of the flow rate, the fluid in the reactor core becomes warmer and warmer, as a result of negative temperature reactivity feedback, the reactor power declines.

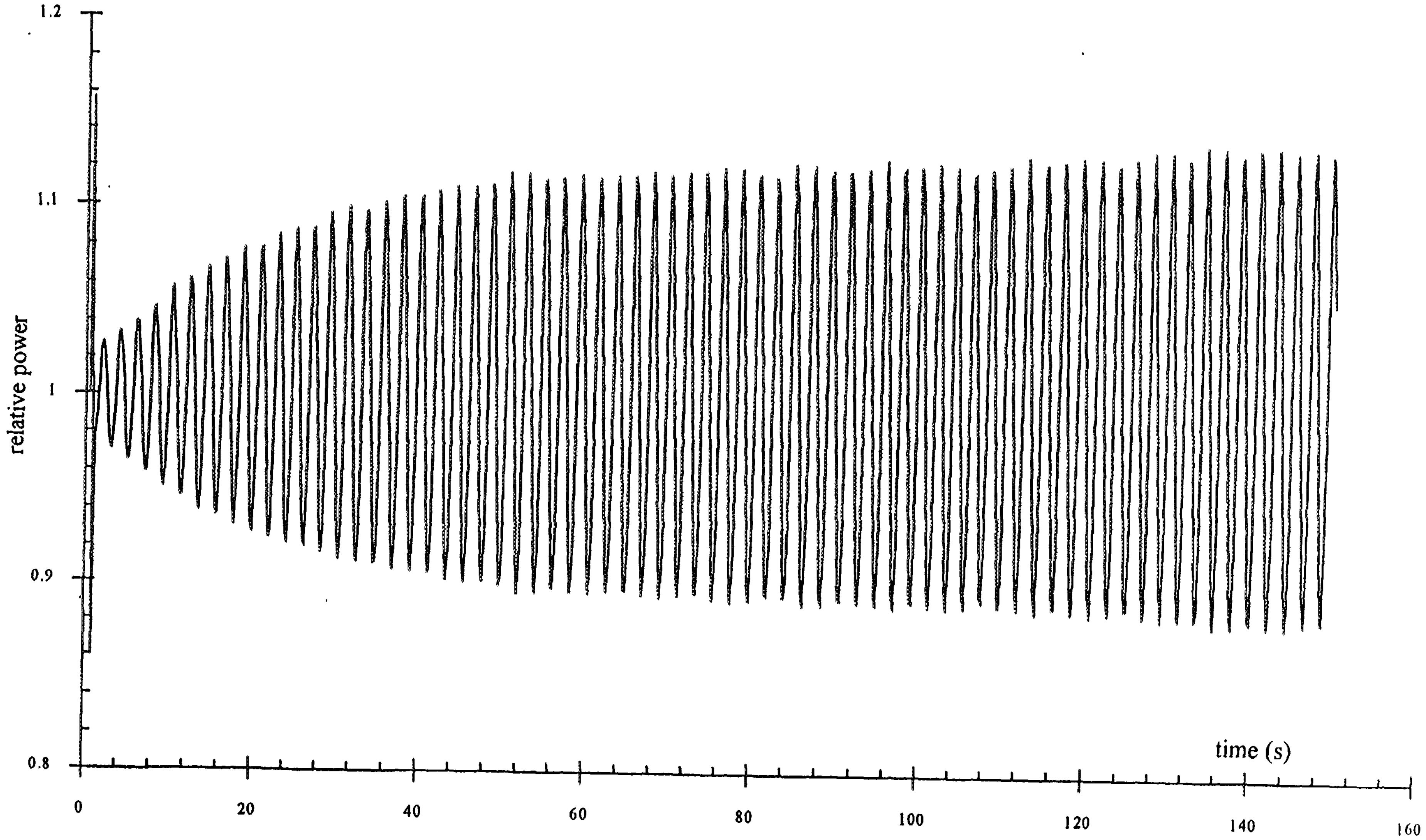


Fig.10 An equi-amplitude oscillation found by calculation

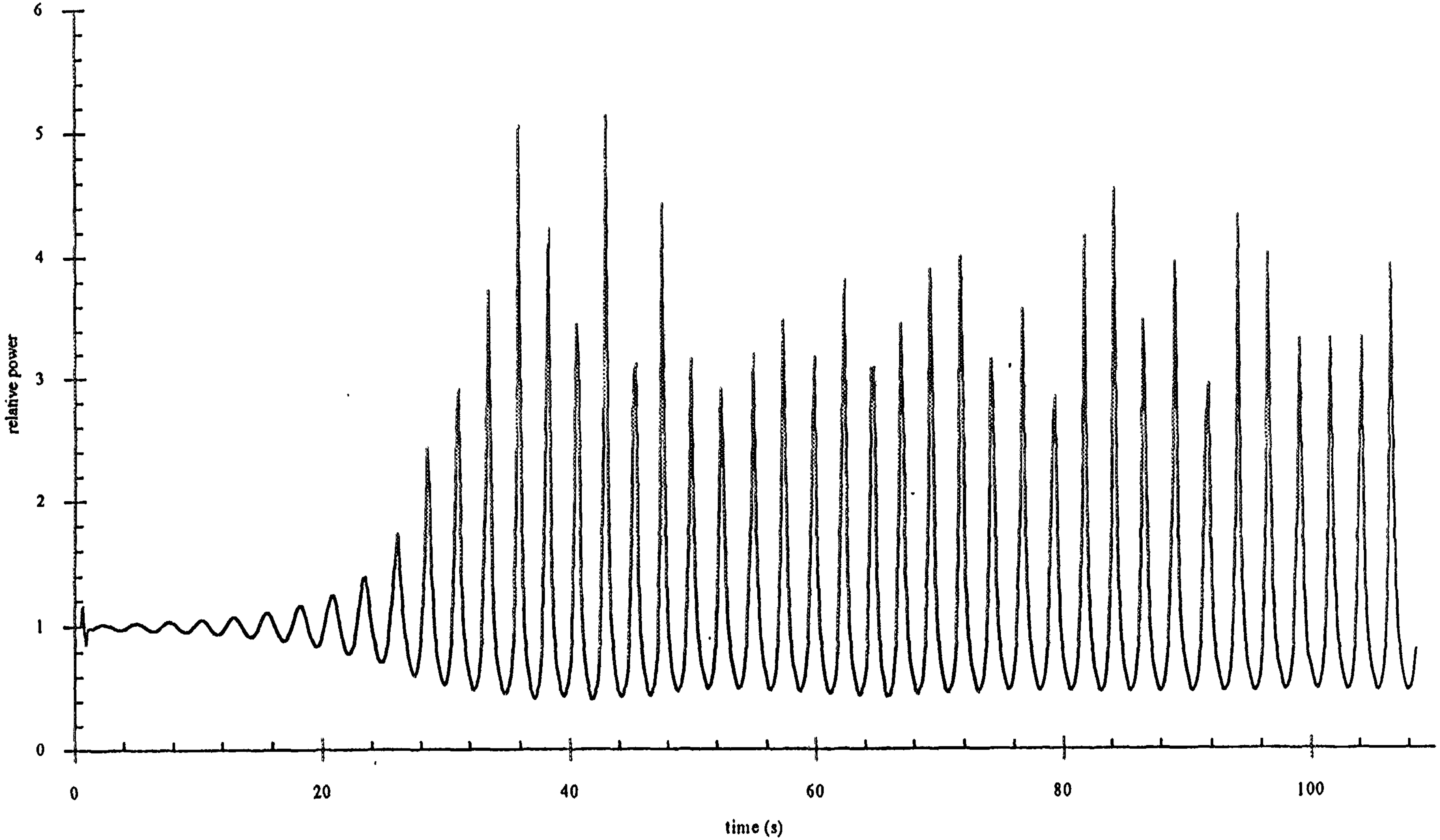


Fig.11 A vari-amplitude oscillation found by calculation

When the declination of the reactor power falls behind that of the flow rate, the core outlet temperature will be superheated and boiling condition may occur, and the system might enter a local two-phase unstable region.

The steady-state operating point is in the stable region. The system enters the density wave instability region after loss of heat sink accident. The power oscillation is illustrated in Fig.12. In order to avoid DWI in the upper heating segment and the riser, the impact of several measures such as increasing system pressure or reactivity feedback coefficient and reducing water storage in riser, were investigated. The conclusion is that a larger density wave oscillation during transients can be avoided by increasing system pressure and by reducing water in the riser.

4.2. DWI during transients from slightly boiling condition

When NHRs are operated at a slightly boiling condition, there exists obviously a steam void in the upper heating segment and the riser. System pressure is maintained by the steam in the upper plenum. After a loss of heat sink accident, the system pressure, temperature and reactor power increase. When the system pressure is high enough, the safety valve opens. Fig.13 and Fig.14 demonstrate the variations of reactor power with time corresponding to two different sizes of downcomer. In the system with a relatively largely downcomer, oscillations occur after the safety valve opening. In contrast, by reducing the downcomer volume such that the core inlet temperature can more quickly follow the rising of the system pressure, DWI during transients can be avoided.

5. STUDY OF PHYSICAL MECHANISM OF DENSITY WAVE INSTABILITY

Stability phenomena also exist in many fields such as mechanical structures, magnetic fluids and thermodynamic equipment. These stability phenomena may be correlated to the tendency of a spontaneous process. Higher energy status isn't stable, but lower one is. For a spontaneous process there is a transition only from a higher energy status to a lower one. The energy principle which describes the minimum energy status of a system is one of the main theoretical tools for the study of above phenomena.

By using the general excess entropy criterion of non-equilibrium thermodynamics, an energy principle of two phase flow density wave oscillations was established in order to provide a convenient and explicit dimensionless criterion for DWI. The spectral entropy feature of DWI was preliminary investigated.

5.1. Energy principle of DWI [11]

In non-equilibrium thermodynamics, I. Prigogine suggested the second order variation of the general entropy as a Liapounov's function to explain the "dissipative structure" of a system from disorder to order and to predict stability of a general thermodynamic system:

$$L=\int_v \delta^2(pz)dV$$

$$=\int_v \frac{\rho}{T}\left[\frac{(\delta p)^2}{\rho^2 c^2}+\frac{T}{C_p}(\delta s)+(\delta U)^2\right]\cdot dV\geq 0$$

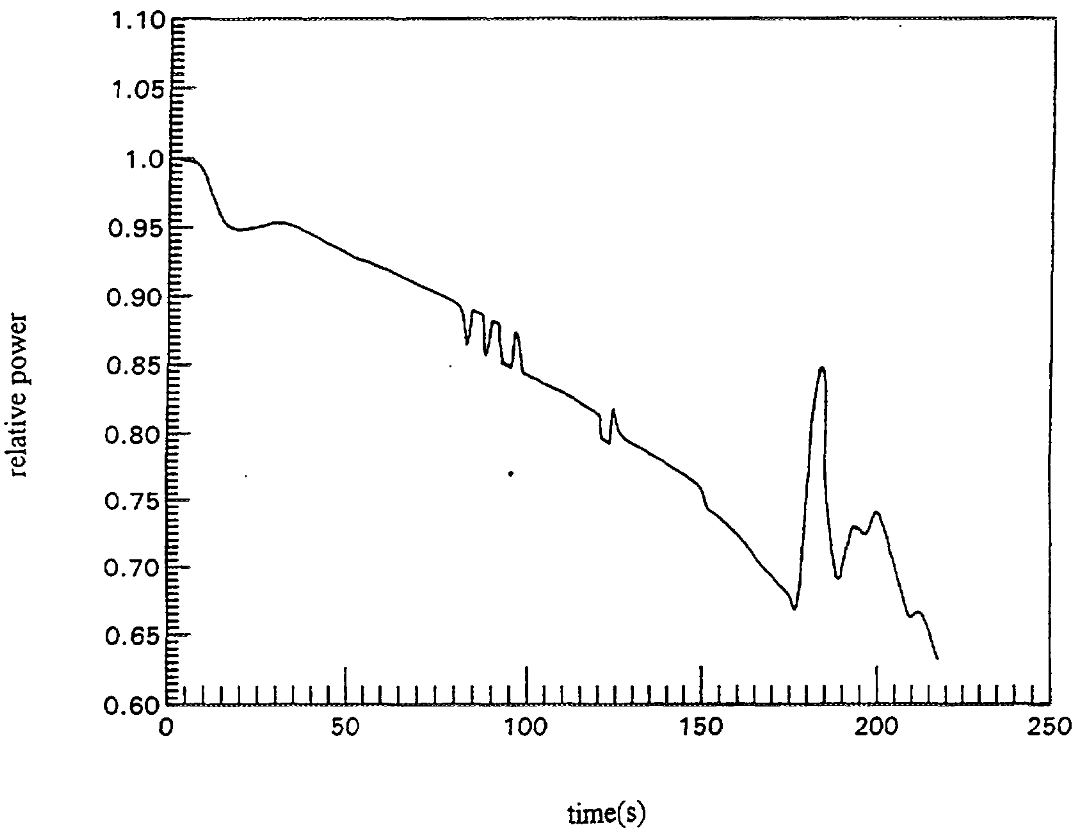


Fig.12 An unstable transient from single phase condition

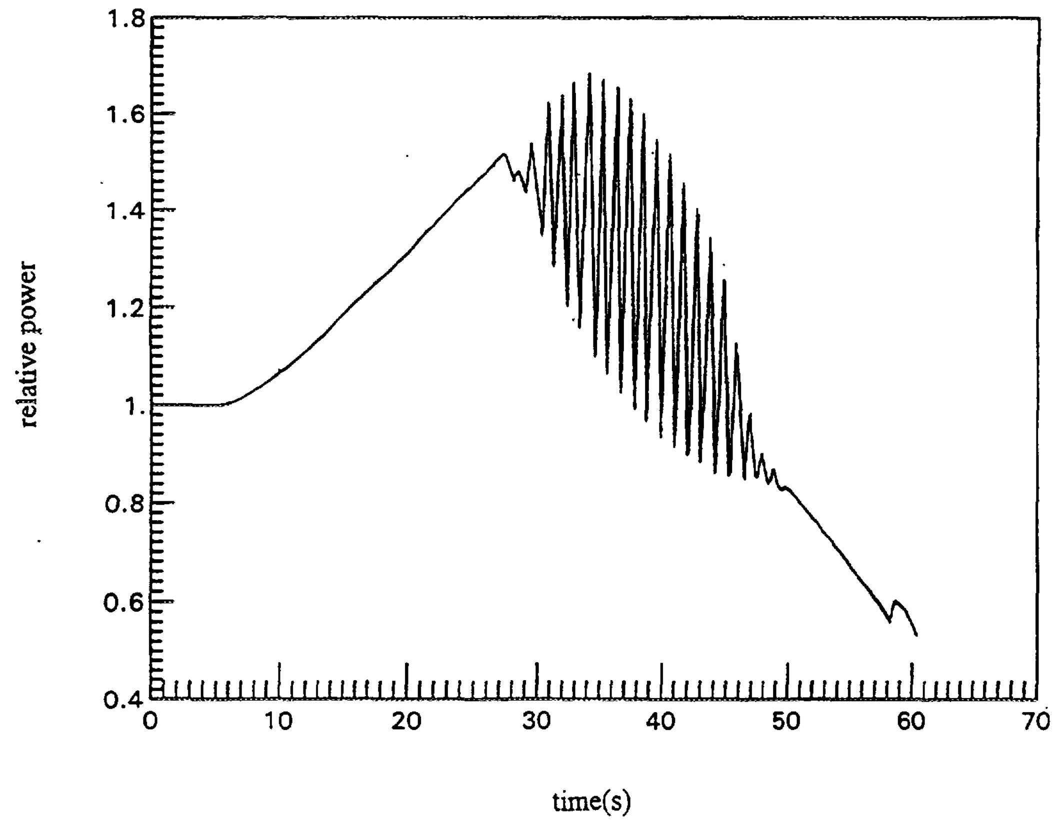


Fig.13 An unstable transient from slight boiling condition
(with large size of downcomer)

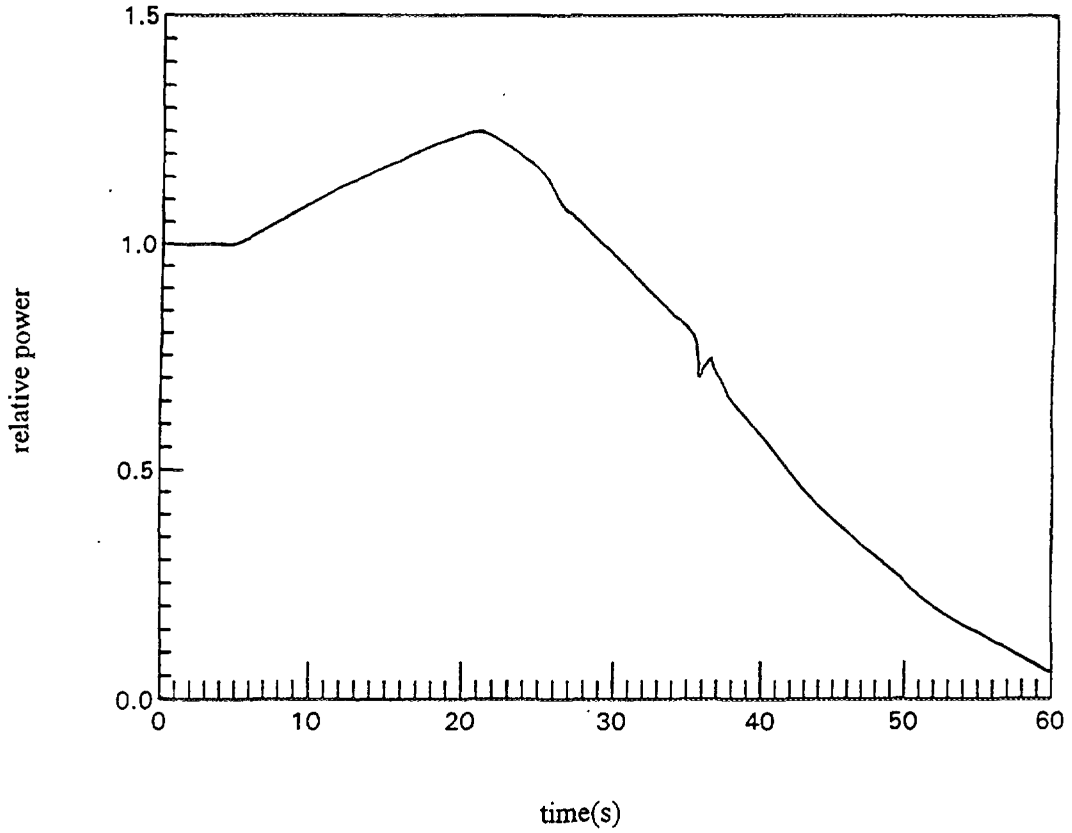


Fig.14 An unstable transient from slight boiling condition (with small size of downcomer)

where L is Liapounov's function, δ^2 is the second order variation operator, s is the thermodynamic entropy, U is the velocity vector, T is temperature and $z=s-U^2/(2T)$ is the general entropy. L consists of three terms: pressure, thermal and kinetic energy. Since two phase flow density wave oscillations (DWOs) belong to low frequency waves and their propagation velocity is far less than the sound velocity, the effects of pressure and kinetic energy terms can be omitted and L function is simplified as:

$$L=\int_{v}\frac{\rho}{C_p}(\delta s)^2dV$$

According to the Liapounov's second theory, system stability can be predicted by sign of the derivative of L with respect of time:

$$dL/dt < 0$$

From this an energy principle, a simplified criterion of the first kind of DWI is [12]:

$$\frac{(1-bX)}{F_f F_\Sigma}(\ln\gamma+\theta_o N_{pch}(N_{sub}-\ln\gamma))-0.4\frac{N_{pch}^2}{N_{sub}N_l}<0$$

(system is stable)

The parameters are listed and explained [12]. Good agreements were obtained between results from this criterion and experiments (Fig.3).

5.2. Spectral Entropy of DWI

By using mathematical proof and spectral entropy calculations for simple examples and a two phase flow system, it is shown that under the same stochastic input, the output spectral entropy of a stable linear system is at a maximum, while for an unstable linear system, its output entropy is at a relatively lower level. Because the spectral entropy describes the output uncertainty of a system, and the second Law of thermodynamics rules the direction of natural tendency, the spontaneous process can be explained as follows: Any deviation from its original state of a stable system will reduce the spectral entropy and violate the natural tendency, thus the system will return to its original state. On the contrary, the deviation from its original state of an unstable system will increase the spectral entropy that will enhance the deviation and the system will be further away form its original state.

A general entropy is:

$$H=-\sum_{i}^{n} P_i \ln P_i$$

where Pi (i= 1, 2, ..., N) is the probability distribute of an information source. For input X and output Y of a general linear system, its spectral entropy density is:

$$h(y)=h(x)+\frac{1}{2}\sum_{k=-N/2}^{N/2}\frac{1}{N}\ln|F(\exp(2\pi k/Nj))|^2$$

where h(X) and h(Y) are the spectral entropy density of input and output respectively. F(z) is the transfer function under Z-transform, and f is the dimensionless frequency.

In order to verify the reliability of the theory, a spectral entropy of the two-phase flow test loop mentioned above was calculated and is shown in Fig.15 [13]. The transfer functions were provided by NUFREQ. A "mountain" of negative entropy rises in the low quality region between single and two-phase conditions. If the contour line of negative entropy 2.5 was projected on a power-subcooling map, it can be found that the projecting line is very close to the stability boundary given by tests (Fig.16).

The flatlands in Fig.15 correspond stable regions, while mountains to unstable ones. When the spectral entropy of a two-phase flow system is located in the "mountain" area, its potential energy is higher, any deviation is enhanced by its spontaneous process and, thus, it is unstable.

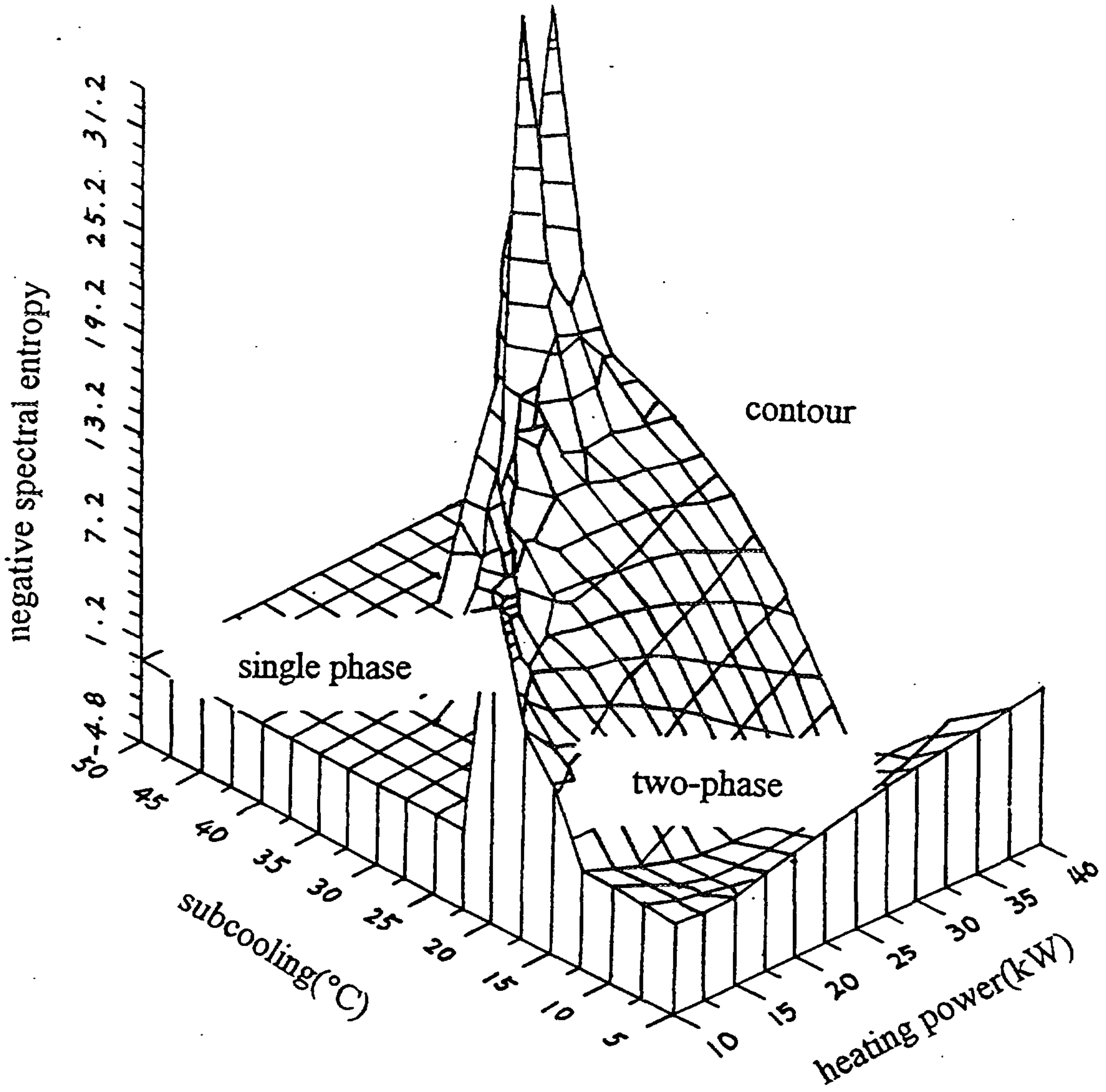


Fig.15 Spectral entropy "mountain" of DWI

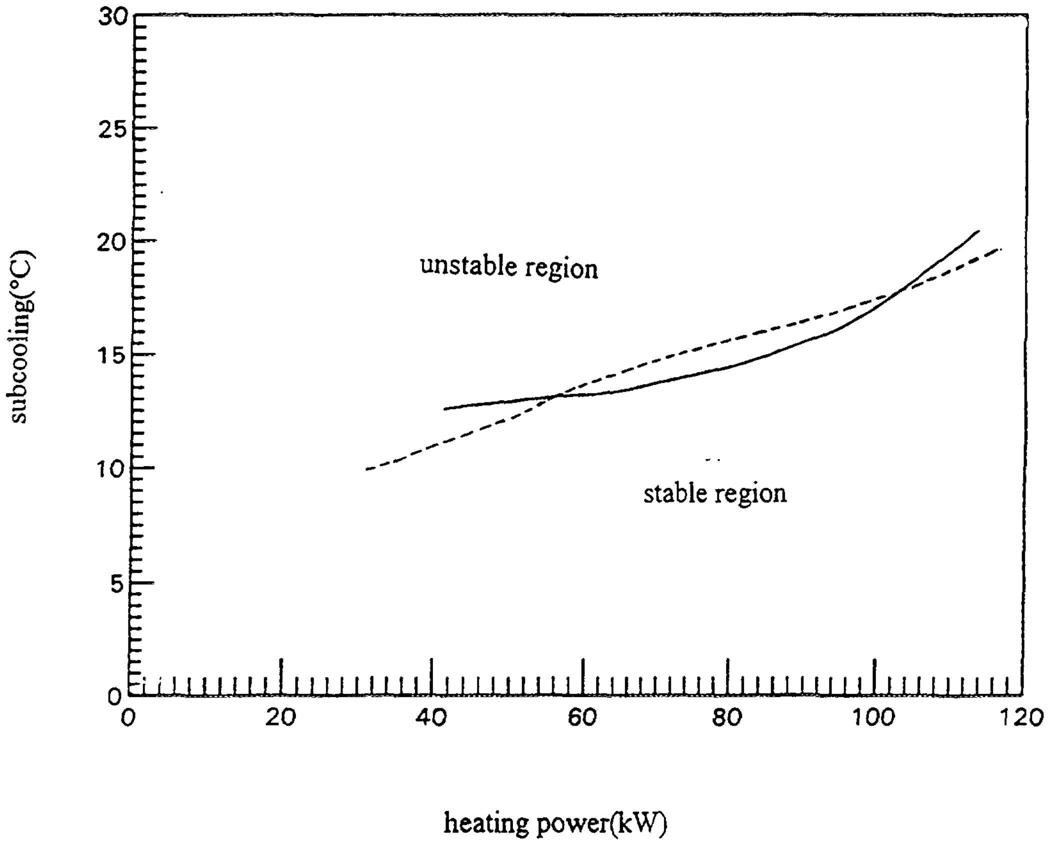


Fig.16 Comparison of results from spectral entropy calculation and test

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BASIC CONSIDERATIONS FOR THE MECHANICAL DESIGN OF HEATING REACTORS

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Abstract

The paper discusses the principal aspects of the mechanical design of the reactor unit for a nuclear district heating plant. It is reasoned that the design must be specifically tailored to the characteristics of the application, and that the experience gained with the design practice of big nuclear power stations must also be incorporated. Some examples of the design solutions developed for the SIEMENS NRH-200 are presented for illustration.

1. GENERAL

This paper reflects the thoughts and work concerning the mechanical design of a heating reactor. The design of a small reactor of low specific cost is a great challenge for the reactor designer. On the basis of the assumption that such a reactor should serve as a nuclear heat source that provides the base load for heating, low temperature industrial processes, air conditioning or water desalination, the exit temperature of the distribution grid can be much lower than that one of the steam cycle of nuclear power plants. This entails in general low pressure levels in the circuits. The size of heating reactors is limited to the local demands which are substantially smaller than the economically reasonable size of reactors for electricity production. Since national heat distribution systems cannot be realized due to size and capital cost, the size of nuclear heat sources is thus depending on the grain size of the local demand. Under economical boundary conditions the minimum size can be seen at a few 100 MW thermal. Under specific circumstances (isolated cities or industrial complexes) also smaller reactors can be taken into account. In the case that CO2 taxes will burden fossil fired heat sources, the economic break even tends to smaller units. In general, at the time being a standard size can be seen at 200 MWth [1, 2, 3, 4].

The design of such reactors requires the consideration of a series of influences, as shown in Fig.1. The dominating influences are the cost of erection and the required development.

In total ten basic concepts of heating reactors are known from the literature. Four of them are at a high degree of development. Two smaller reactors were commissioned and operated successfully. Unfortunately, the development of heating reactors was slowed down in the Western countries.

2. DESIGN AREAS OF MAJOR IMPORTANCE AND STRONG INFLUENCES ON THE DESIGN

Prior to the reflection of design aims and features, some principal trends of thoughts are summarized hereafter.

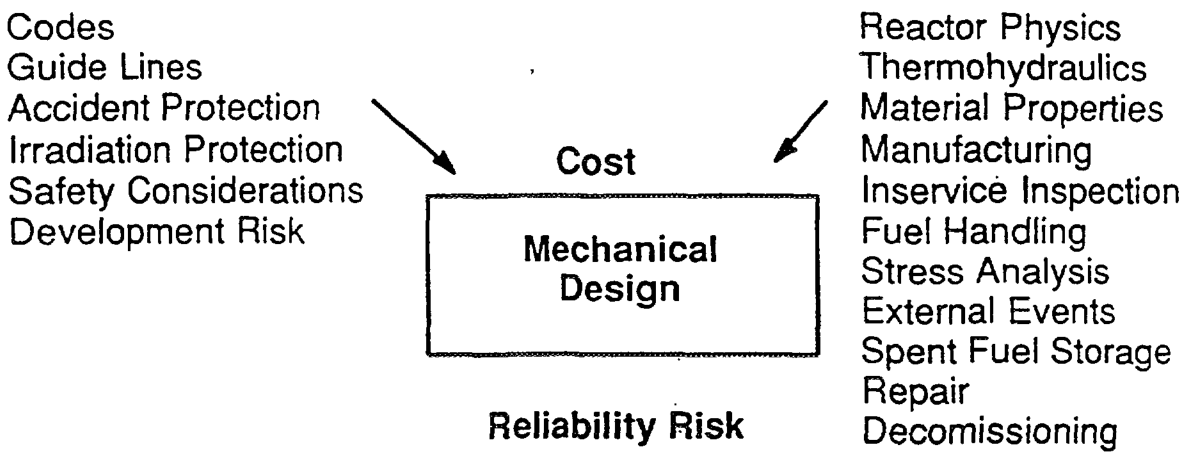


Fig. 1 Influences on the Mechanical Design of Heating Reactors

2.1. Reactor shut down

In addition to the self-evident inherent features of a negative reactivity coefficient, two independent shut down systems are required for a reactor core. One of them is normally a control rod system, while the second system uses soluble boron acid or comparable materials.

The control rod systems have a rather great influence on reactor designs with pressure vessels since the drives are located on top or below the active core. A position below the core increases height and costs. A position above the vessel, or the core, also leads to high buildings, and needs space for an irradiation protected storage (of said component), and equipment for handling which also influences costs in a negative way. In-vessel and in-core drive systems as proposed for the Siemens-NHR, the SHR and the LTHR-200 require high development costs. Due to their relatively small driving forces and reserves in lifting force, the control assemblies must be suited for such drives. Since the friction forces are mainly depending on the spring constants of the guide structure, the control assembly and the drive piston, the mechanical engineer has to take care for a free insertion path under all circumstances.

Mechanical systems under water and in-vessel cannot be built so small that they do not influence fuel handling. Furthermore, electric cables (connectors) for power supply as well as motors and gears lead to serious concerns and developments. In addition, one has to take into account that small cores lead to rather light control assemblies which entail the same problems as mentioned for internal drives.

Problems also arise from the position indication system. Thus, internal drives would be too big in diameter if solenoid type indicators are envisaged. For the required cables there is simply no space and the handling problems during fuel handling are serious. Ultra-sonic transducers behind a stainless steel barrier, incorporated in individual lances have been developed - but they need a bubble-free flow between the tip of the lance and the top of the control assembly. This fact lead to designs with a complete and leak-tight separation of non-boiling water in the control assembly guide structure and the fuel channels over their entire height of the guide structure.

A free control rod path is mandatory. Mechanical drives increase containment height and need a shielded space for deposition during fuel handling. In-core hydraulic drives are burdened by high development costs.

2.2. In-service inspection

Inspection is mainly a question of access. All welds of the primary boundary must be accessible for (simple) manipulators. Their flank angles have to be oriented in a way, that an ultrasonic inspection device can receive the echoes. Adequate irradiation protection, water or other shielding, is required. Furthermore, a coupling medium between the inspection device and the inspected surface is needed. Experiences with BWR demand a minimum handling distance between two components of about 200 mm. It is helpful to draft the tools early, in order to identify obstacles.

All vessel internals must be designed in a way that at least an optical inspection of the connections and seals is possible (preferably without removing other components). If components have to be removed, the connections must be designed to be very reliable and simple, and adequate shielding for transportation is required.

Access to all welds, connections and seals is mandatory.

2.3. Influence of classification of primary circuit components

If, for example, the integrated heat exchangers of the intermediate circuits are nuclear class 1 components, all welds must be accessible for manipulators. This leads in practice to U-tube heat exchangers, because this type has no lower plenum (and feed pipes welded to the latter), and thus no welds which are not accessible for inspections. Class 1 heat exchangers of the intermediate circuit must be designed as U-tube types.

2.4. Fuel handling

In big reactors, fuel handling including spent fuel storage is solved by deep, lined channels, and by a spent fuel pool with a capacity of some cores plus the space needed for a fresh core. Since the cores of heating reactors are rather big due to their low volumetric power density, a classic refueling scheme would entail a serious increase of the reactor building and thus increase cost.

Simplified systems are required to achieve the economic aim. In-vessel storage seemed to be the most attractive solution, but it increases the vessel diameter. Furthermore, the long term fuel storage at high temperatures and very close to the core is not within the experience gained up to now. Thus at least the first heating reactors are burdened by an expensive structure for fuel handling. Pathfinder storage assemblies are requested to investigate the effects of an vessel internal storage and to provide data for in-vessel storage for more advanced reactors.

One of the main disadvantages of an in-vessel storage is that in case of a total removal of the core and the spent fuel plus other internals because of a major defect of, e.g., the core support structure, only the blow down tank is suited for an intermediate storage.

Even in the case that one substitutes the lined concrete channels by temporarily installed steel trenches, and the spent fuel pool by a steel tank with compact racks, the solution is not in the cost optimum.

Other refueling schemes like shielded flasks look in the first moment attractive since trenches are not required, but they are rather complex in detail and have the inherent risk of a drop of the handled fuel assembly or a drop of the heavy flask onto the core. Both events entail serious damages and radioactivity release.

An in-vessel fuel storage allows the design of simple and small reactor buildings.

2.5. Repair

One has to take into account that components can fail. With a sound design no adverse effects on reactor safety should occur in said case. In order to reduce personnel dose and down time, the component must be easy to replace, and there must be a place to which the component can be transported and stored without exposing personnel.

The blow-down tank which is filled with water for condensation of the primary blow down steam is a place which fulfills all the demands for a storage of contaminated components. Also, large pools as shown in the Geyser, the SHR and the Slowpoke NHRs are suited for long term storage, and are advantageous for the handling of contaminated components.

Thus, the mechanical design must take into account that components can fail. All provisions for replacement must be incorporated in the design from the very beginning. Modular assembly leads to small components and thus facilitates handling.

2.6. Sound design of components and systems

A sound design must take into account a perfect tailoring of each individual part depending on its boundary conditions as are:

- pressure
- mechanical load (static and dynamic)
- temperature
- temperature differences across the wall
- medium flow (velocity and temperature)
- neutron flux
- material properties
- dimensional changes during operation

For example, the intermediate circuit inlet is located in the upper area of the RPV. The vessel wall is at the core exit temperature. Serious thermal stresses burden the wall and the weld connection. The incorporation of thermal sleeves hinder inspections. Thus it is worthwhile to design concentric tubes with the exit flow of the intermediate circuit in contact with the RPV wall. Such a design reduces the stresses substantially.

3. DESIGN AIMS AND SPECIFIC FEATURES

3.1. Low capital cost

Low capital cost can be achieved by minimizing the amount of hard- and software, development, and the required quality insurance measures by taking advantage of the specific system parameters. In order to reach this overall optimum, a certain practical experience and creativity is necessary.

A minimum hard- (and soft-)ware is easy to define by a minimum size of the reactor building and a minimum number of components and systems. Such a minimum size cannot be achieved by a simple pantograph reduction of an existing system. The first step in cost reduction is the identification of those components and systems which can be omitted by taking credit of the size and the design parameters of a low temperature (low pressure) reactor. Typical examples for this are the omission of primary circuit loops, pumps and safety injection systems. By omitting those systems, the required subsystems and their control equipment can also be avoided. But the system must be designed as an integral reactor with a tight fitting containment and natural circulation of the primary circuit.

In order to gain the maximum reduction of building height, an integration of the control rod drive system in the reactor pressure vessel is very helpful since the required space for the drive mechanisms can be saved. But on the other hand, such a system needs a substantial amount of development. If the development cost can be distributed over a series of more or less identical reactors the penalty becomes tolerable.

A reduction of the required hardware also reduces the software by a similar fraction. A clear, understandable design of each individual component reduces the required work for quality assurance. If there are experienced suppliers, the individual details should be as far as possible within their field of experience.

3.2. Fuel design

There are different aspects to consider. The principal requirement is the commercial availability of the fuel material. Thus only uranium dioxide and, to a certain extent MOX, in pellet from is tolerable. Size and fuel element structure are of minor importance, they can be suited to the demands of the nuclear heating source. But, in order to reduce development cost, it is worthwhile to design the fuel assembly around an existing spacer grid.

It is self-evident to house the fuel rod bundles in channels if a net steam production is considered. Such channels can be of a small wall thickness because only very small pressure differences between the channel interior and the space between the channels are expected. Due to experience in the FR Germany, the life time of the channels for the NHR can be longer that one of the fuel bundles for power reactors [5] even in the case of high discharge burnups.

A mechanical separation of the fuel rod bundle and the channel leads in a first step of design to a core structure consisting of fuel channels which can remain over the complete service time of the reactor within the pressure vessel. In a second step the channels can be rearranged to cruciform control rod guide structures, forming together with simple additional bent sheets a core cell which contains four fuel rod bundles, with a cruciform control rod in the center. In the case that a hydraulic control rod drive mechanism is applied, this drive can also be incorporated into said structure. Such heating reactor core cells can serve as a one-piece guide structure without any interruptions or offsets over the entire control rod stroke, if the height of the cells is selected to about two core heights plus the necessary space for mechanical overlappings. Furthermore, such core cells can be used as compact storage racks for spent fuel within the reactor vessel (see below). The heating reactor core cells allow refueling without removal of any vessel internals.

For the required pressure tests, the fuel assemblies can remain within the reactor vessel. Because of the low working pressure, the test pressure is also low, and the mechanical stresses are negligible.

In case a heating reactor is designed with core cells, it is advantageous to base the design on proven BWR fuel assembly technology.

3.3. Fuel handling and fuel storage

As mentioned above, the measures for fuel handling and storage require a remarkable space in the reactor building and thus burden the system with high cost. Due to the fact, that heating reactors have a low power density in the core, and because the annual power generation period is rather small (in Western countries about 5000 to 6000 full power hours), the total required number of fuel assemblies is rather small if they are designed for a high average discharge burnup. Due to this fact a vessel-internal storage of spent fuel assemblies appears to be feasible.

In case the reactor is designed on the basis of heating reactor core cells, an in-vessel-handling without removal and storage of control rod drives and guide structures is in principle possible. The core of the SIEMENS NHR-200 with a power density of 20 kW/l consists of 180 fuel assemblies, housed in 45 core cells. An annulus of additional core cells provides enough space for a complete core within the vessel if the fuel rod bundles were stored as two in one channel (two axial layers), which due to the small weight and length of the bundles is no mechanical problem. Also, the storage at primary circuit temperatures and pressure may not cause invincible obstacles.

If the space is not sufficient to store the total number of fuel assemblies needed for the entire life time of the reactor, another (partial) annulus of core cells must be incorporated. The empty space between and below the heat exchangers of the intermediate circuit could be utilized for storage positions.

In case that one plans to withdraw the fuel assemblies out of the reactor, the in-vessel-storage can be used for an intermediate storage of spent fuel. In such a case the reactor pressure vessel and the blowdown tank which is located close to the vessel can be connected with a temporarily installed metal trench. The blow down tank is best suited as a transfer basis for the fuel assembly into a transportation cask for spent fuel. In both cases the reactor building can be minimized. This procedure can also be applied to gain space for new reloads if it is planned to prolong the life time of the reactor.

3.4. Primary circuit design

In heating reactors the primary circuit temperatures and pressures are low compared to BWRs or PWRs. They depend on the end temperature of the distribution grid temperatures, the size of the intermediate heat exchangers and on the operation mode of the primary circuit.

Typical values for the operation pressure are 1,5 to 3 MPa. Reactor vessels for such low pressures can be made from stainless steel or stainless steel-plated ferritic material. Due to the low pressure, the wall thickness of the vessel is rather low. This fact allows a full pressure containment which encloses the reactor pressure vessel, even in the case that the containment is not heat treated after welding.

The reactor pressure vessel should be suspended in the area of its flange. The lower part should be free of any penetration.

In order to reduce the height of the pressure vessel, and thus of the containment and the reactor building, the location of the core should be as close as possible to the bottom of the vessel. The core support structure is advantageously to be affixed by welding to the wall of the vessel. From there, a core shroud and a riser are required to channel the primary water to the inlet openings of the intermediate circuit heat exchangers.

The downward channeling of the fluid must be provided by the individual shells of the heat exchangers. In order to limit the activity level at the loops of the intermediate circuit, a certain distance between the lower edge of the heat exchanger bundles and the tip of the core is required.

It is advantageous to support the heat exchanger bundles at their connections at the (top) headers. Furthermore, it was found that a concentric connection with the cold feed water inlet in a central (removable) tube reduces both the numbers of penetrations (cost) and thermal stresses. A grouping of two bundles at one penetration reduces the number of penetrations to the feasible minimum, but complicates the flow guide structure in the headers. In total, such a design saves cost, however.

Since the sum of the diameters of the heat exchanger bundles is smaller than the perimeter of the annular space in which they are located, space can be gained by grouping of the heat exchangers in two groups. The resulting empty space is well suited for an in-vessel-storage of defective parts, spent fuel and for instrumentation lances during refueling.

The mechanical design of a heating reactor must be influenced mainly by safety, reliability and cost. To fulfil all the requirements, several steps of optimization are required. A consideration of the experience gained with big reactors is mandatory.

3.5. Aspects concerning the intermediate circuit and reactor control

Nuclear reactors for low temperature heat generation can take credit from the demands of the heat distribution grid. Thus the temperature and pressure levels in the primary circuit are much lower than for electricity producing power stations. On the other hand, they are burdened by the fact that radioactivity can be transported by all circuits. In order to prevent leakages to enter the heating grid all designs have an intermediate circuit between the primary circuit and the latter. Most of the concepts show a higher pressure level in the intermediate circuit, thus leakages increase the water content of the primary system. Depending on the requirements of the local temperature demand and the size of heat transfer areas, the pressure level of the primary circuit has to be selected. Typical values are 1,5 to 2,5 MPa for the primary and about 0,5 MPa higher for the intermediate circuit. For the pressurization of the primary circuit different modes are possible, self-pressurization by a net steam production in the core or with single phase circuits by a cover gas such as nitrogen. Furthermore, individual steam producing fuel assemblies or electrically heated inserts can serve as a steam source.

With all steam pressurized reactors, a pressure control by condensation is required. Some of the concepts show heat exchangers which project into the steam plenum and thus provide a sufficient condensation area. Furthermore, separate control condensers are feasible. In the latter case an additional degree of freedom for control is gained and the pressure control becomes easier. From the mechanical design such a control condenser is comparable to the other heat exchangers of the intermediate circuit. If in nuclear class 1, they have to be designed as U-tube bundles, but with the header at the bottom. For the penetrations it is also worthwhile to follow the same philosophy as for the heat exchangers.

For reactivity control in single phase reactors boron acid and control rods can be applied, whilst for boiling systems only control rods are feasible. It is self-evident, that for both cases burnable poisons can be used. Since for economical reasons the nuclear heat sources are designed for the base load of the distribution grid, long down times are expected in most of the applications. The time constants of the grid are rather big, thus fast load changes are not expected. This facilitates the design of the components because the load cycles are mild and limited. Also the number of control rod motions is small. Depending on the local requirements only some 100 000 steps over the entire lifetime are expected.

4. SPECIFIC EXAMPLES OF THE DESIGN OF THE SIEMENS NHR 200

4.1. Heating reactor core cell

The key design feature of the Siemens NHR is the heating reactor core cell. This component is a novel design element which was developed for small natural circulation reactors. The base structure of the heating reactor core cell consists of a cruciform control rod guide structure which is rigidly connected to the core support grid. It consists of thin bent zircaloy sheets, at their ends welded together with straight zircaloy bars. This structure is located in the center of a square and subdivides said square into four small squares arranged in the corners of the large one. The small squares accommodate the fuel rod bundles. In order to facilitate fuel handling (and to provide an adequate flow guidance) these smaller squares are wrapped with additional bent zircaloy sheets. To gain space for the hydraulic control drive, in the center of the cruciform guide structure the sheets show an outward bow, thus reducing the four peripherical squares to pentagons with the fifth small side in the center of the hole structure. Fig. 2 shows a schematic artist view of the core cell arrangement.

The core cell is extended over the entire height of the fuel assembly. The control assembly stroke and about 500 mm for necessary overlappings of the control assembly and the drive piston, as well as the position indicator lance and the guide structure are fully accommodated.

In order to center and orientate the individual core cells, square frames are pinned to the bars of the guide structure at their upper end and directly above the active core. Those frames are made of stainless steel. To allow differential thermal expansion between individual core cell frames, the outer surface of the frames are equipped with keys. These keys of adjacent frames fit one into the other. Thus the complete core arrangement consists of interlinked individual frames which allow only very small displacement from their initial position but do not hinder any differential growth.

At the periphery of the complete arrangement, a former is located which interferes with the edge frames in the same manner as between the frames themselves.

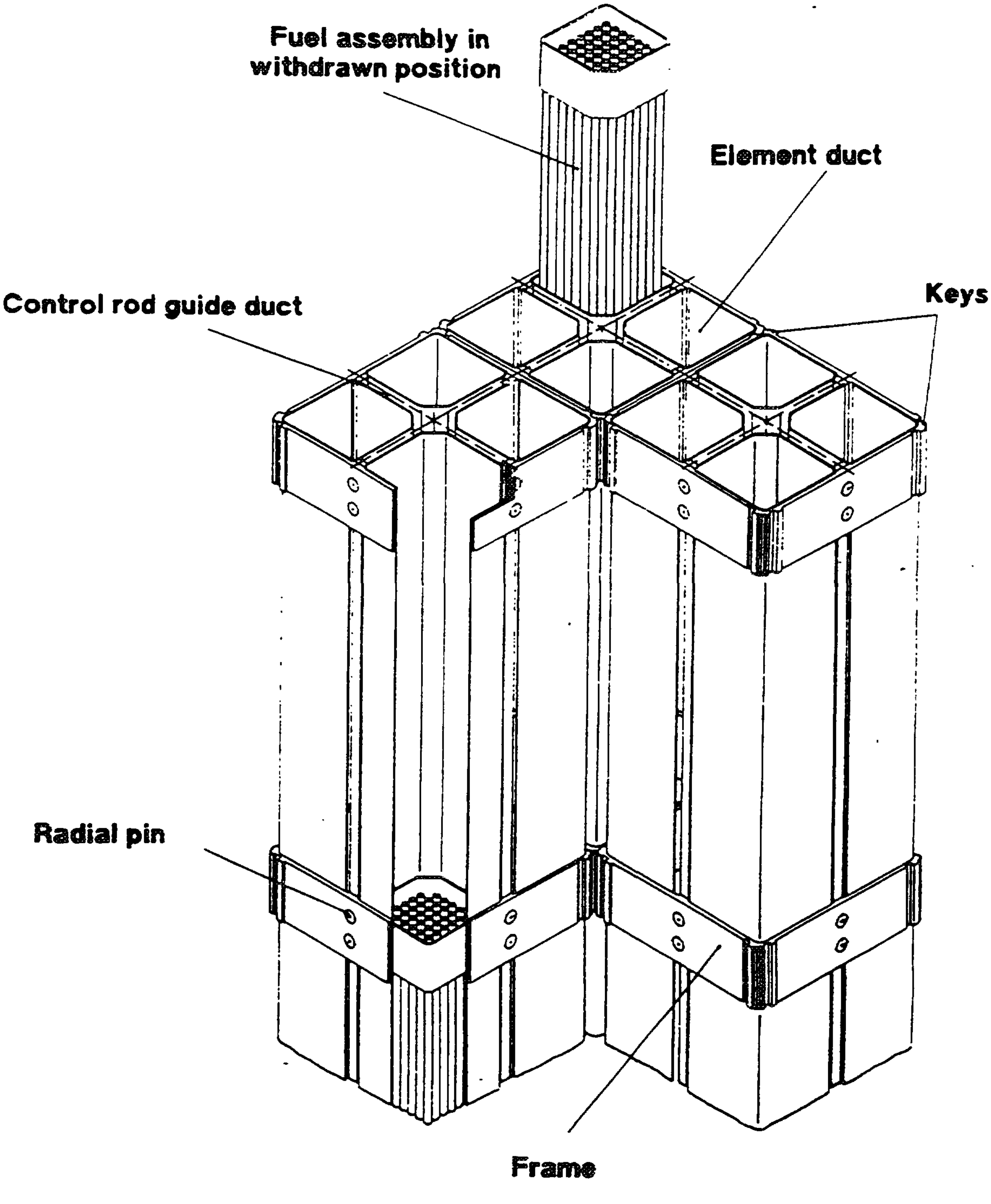


Fig. 2 **Core Cell, Cross Section**

4.2. Hydraulic control rod drive

As mentioned above, the control rod drives are located in the centers of the core cells. Fig. 3 shows a cross section of a core cell with inserted fuel rod bundle, control assembly and control rod drive.

A hydraulic control rod drive was selected since such a mechanism is fully compatible with the primary circuit and core conditions. The drive unit consists of a thin hollow piston and a cylinder which serves also as the spine of the control assembly. This drive unit is fed with primary circuit water. In order to hold defined positions, and to move defined steps upward or down, the piston is equipped with indents. The same is provided at the lower end of the cylinder also, as is schematically shown in Fig. 4. There are wide or narrow gaps depending or their relative position. Fig. 5 shows the related mass flow and Fig. 6 the flow scheme of the drive. Since the regular mass flow for maintaining a certain position is adjusted by a throttle in a control unit. An additional mass flow provided by an opening of the withdrawal valve results in an outward motion of the control assembly. A bypass (controlled by a second valve) between the drive unit and the exit of the throttle reduces the mass flow in the drive unit and initiates an (stepwise) insertion of the control assembly.

Such a full size drive system was tested at the Siemens facilities in Erlangen, under primary circuit temperatures and pressure over more than 500 000 steps and some thousand scrams. The test results showed no adverse function even under a high content of iron oxides and other dirt added purposely to the circuit (part of the tests). There was also no interference with second other drive in the rig. Furthermore, a fluidic ejection protector was tested with great success and incorporated into the design. The complete drive system was assessed by the TÜV Bayern [6, 7].

Figures 7 and 8 show the lower and the upper end of the core cell with integrated control rod drive.

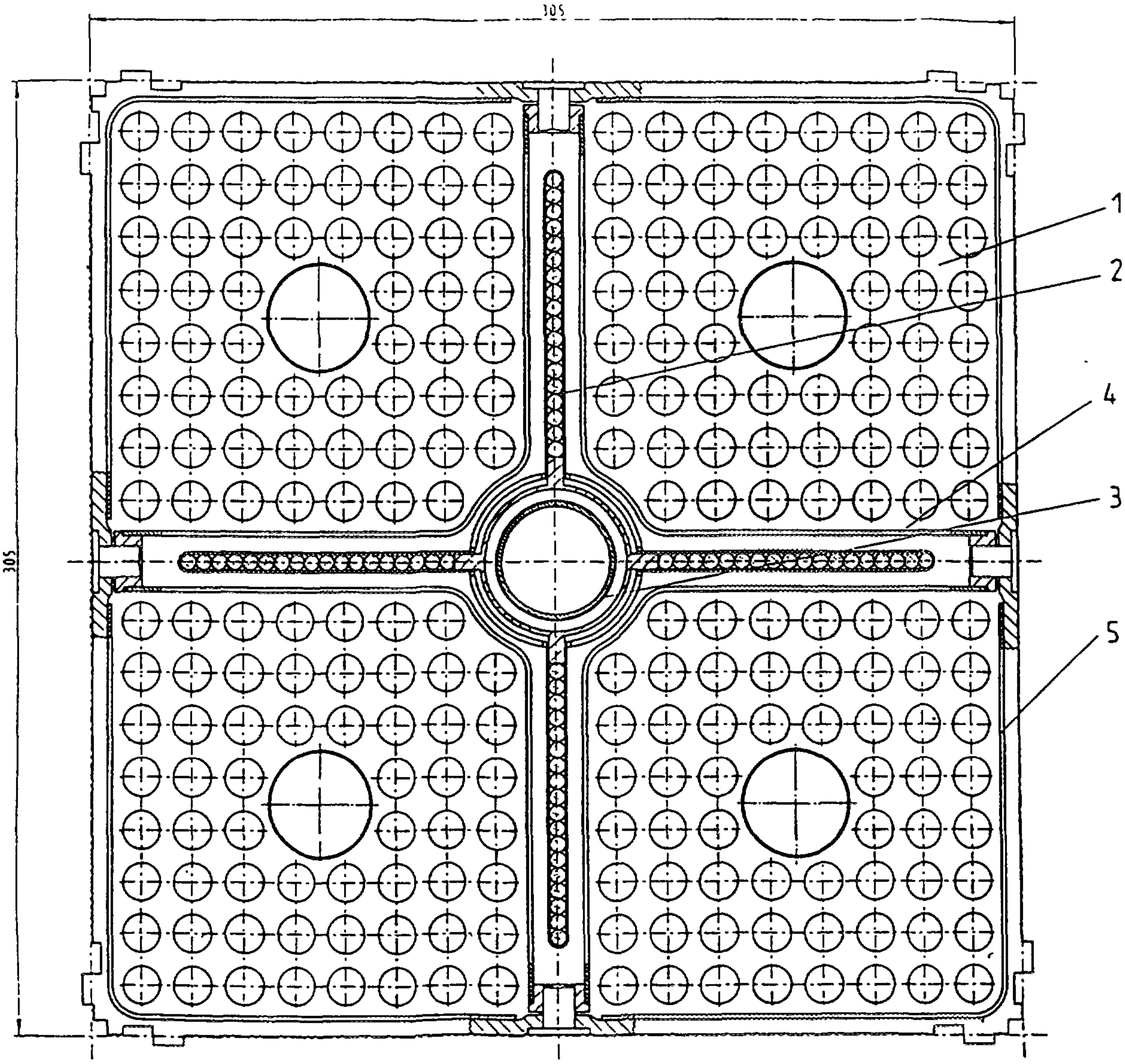
Together with the core cell, the hydraulic drive forms a module. Each individual part is removable for service. The core cell entity allows a free and straight access to the fuel assemblies after removing the closure of the reactor pressure vessel.

4.3. Heat exchanger

As mentioned under 2 and 3, the number of the required penetrations can be reduced to ¼ if two of the bundles are grouped together and a concentric connection is applied. With U-tube heat exchangers (which allow a complete in-service inspection) the central inward flow of the secondary circuit must be channelled to the downward branch of the U-tube which are at the outer side of the bundles (lower thermal stresses). Fig. 9 shows a cross section through the header of such a twin bundle.

4.4. Refueling scheme

In Fig. 10 the refueling scheme of a NHR with an intermediate spent fuel storage in the reactor pressure vessel is shown. The pressure vessel is equipped with a refueling annulus. A U-shaped trench, temporarily mounted between the refueling annulus and the blow down tank provides a shielded channel between the reactor and the spent fuel cask, which is inserted into the blow down tank. After closing and decontaminating, this tank is ready for transportation to a reprocessing plant or to a central storage of spent fuel.



1 Brennelement / Fuel Bundle
2 Steuerelement / Control Rod Assembly
3 Hohlkolben / Piston
4 Führungskreuz / Control Rod Guide Structure
5 Kernzelle-Rahmen / Core Cell Frame

Fig. 3

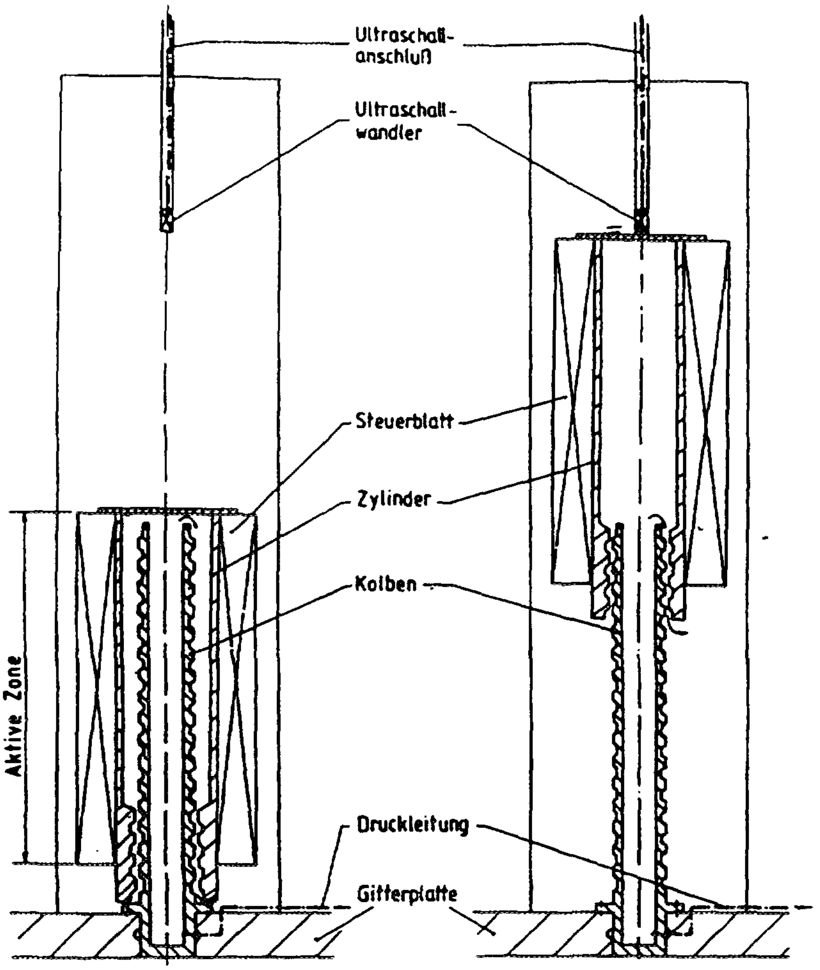


Fig. 4 **Heizreaktor 200 MW**
Hydraulischer Steuerstabantrieb, Funktionsprinzip

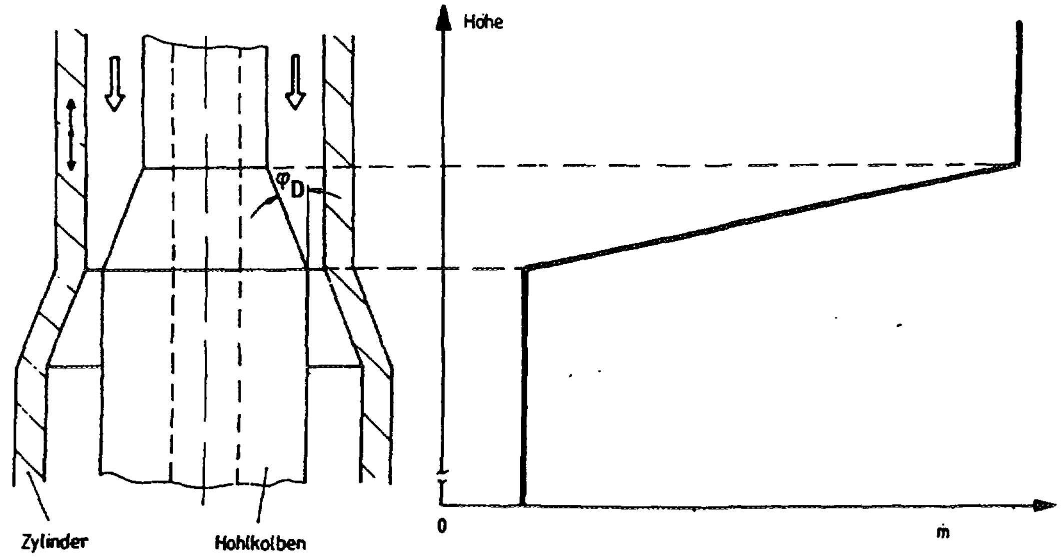


Fig. 5 **Heizreaktor 200 MW**
Zusammenhang zwischen Durchsatz und Steuerelementposition

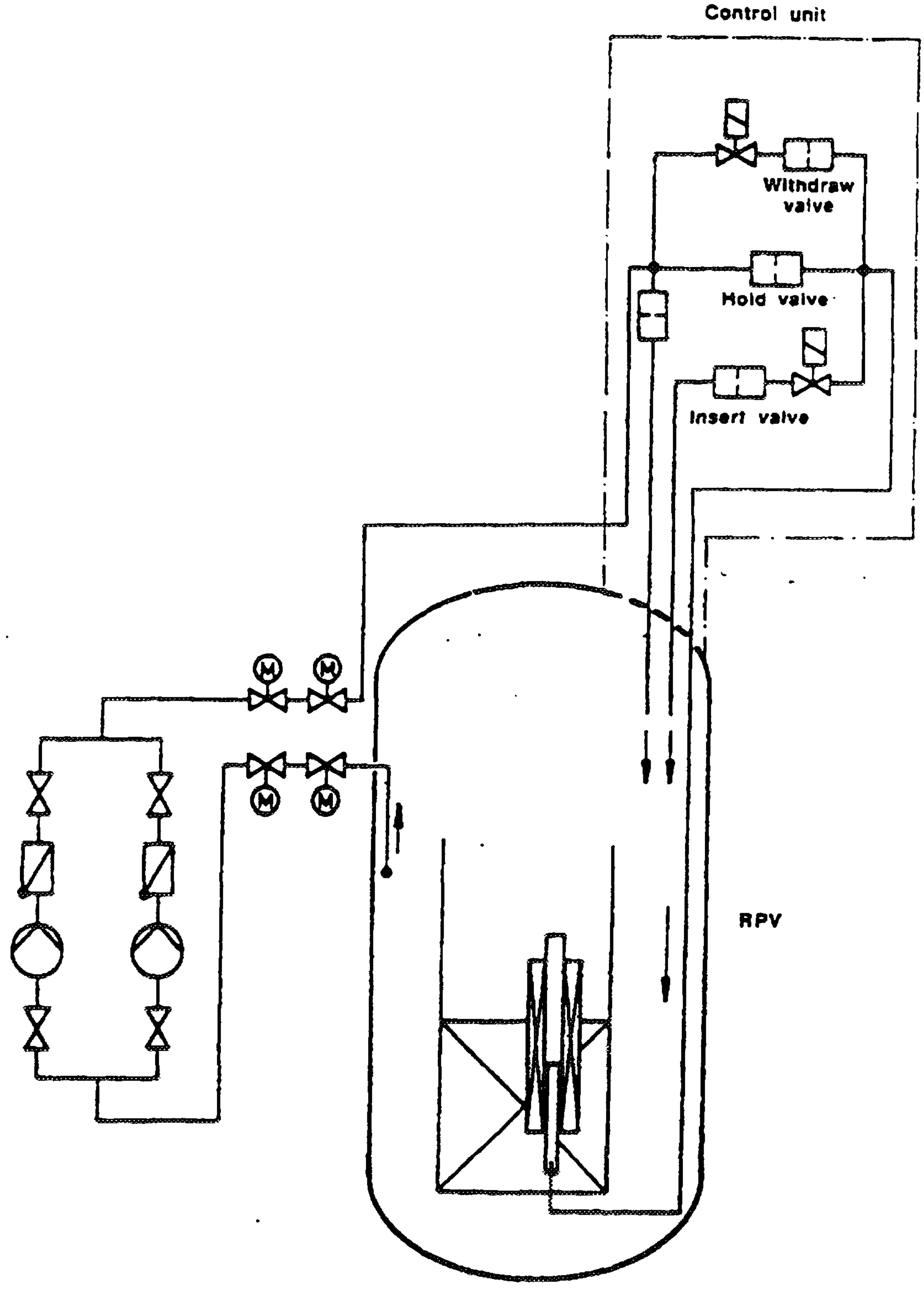
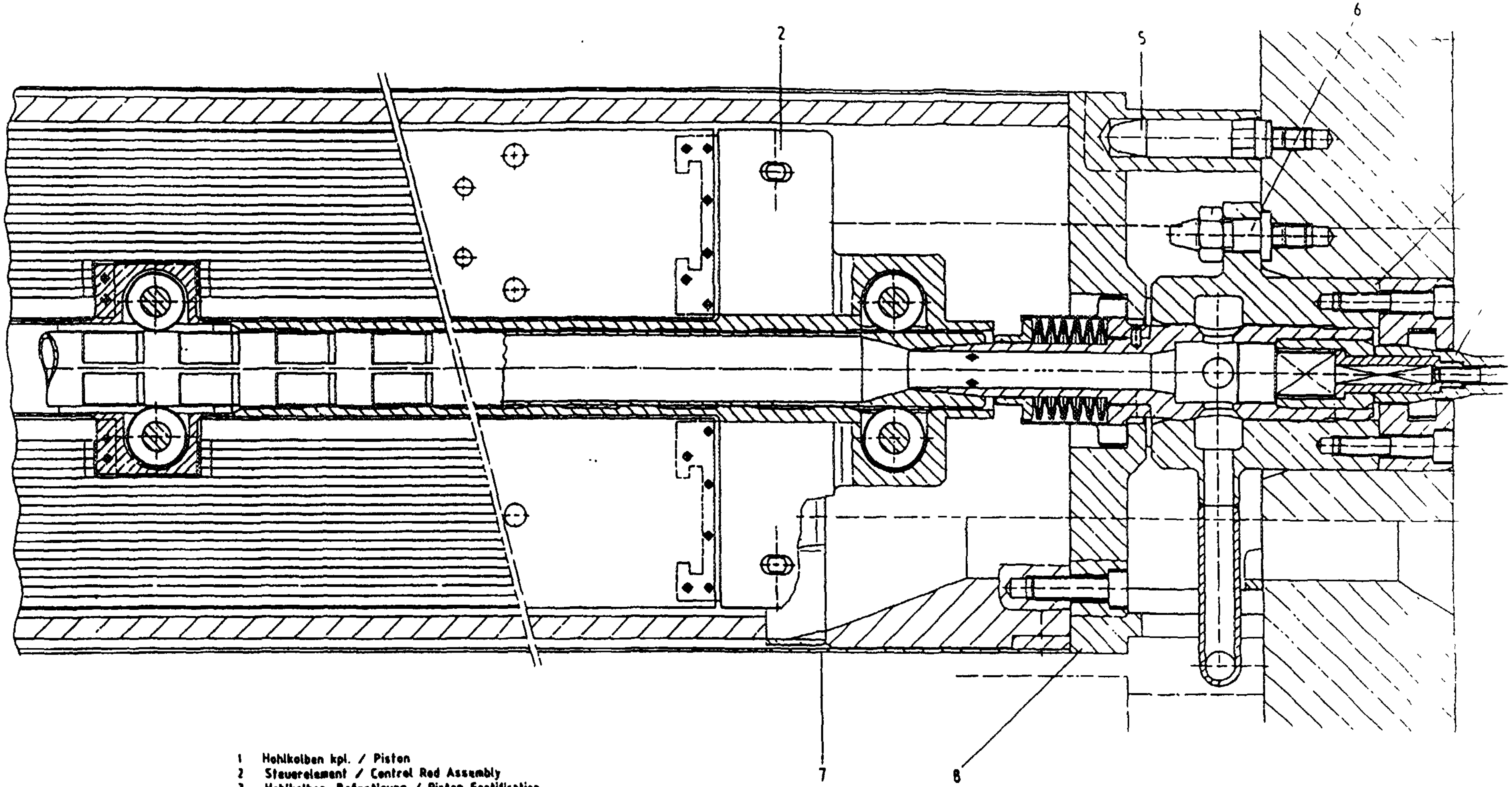
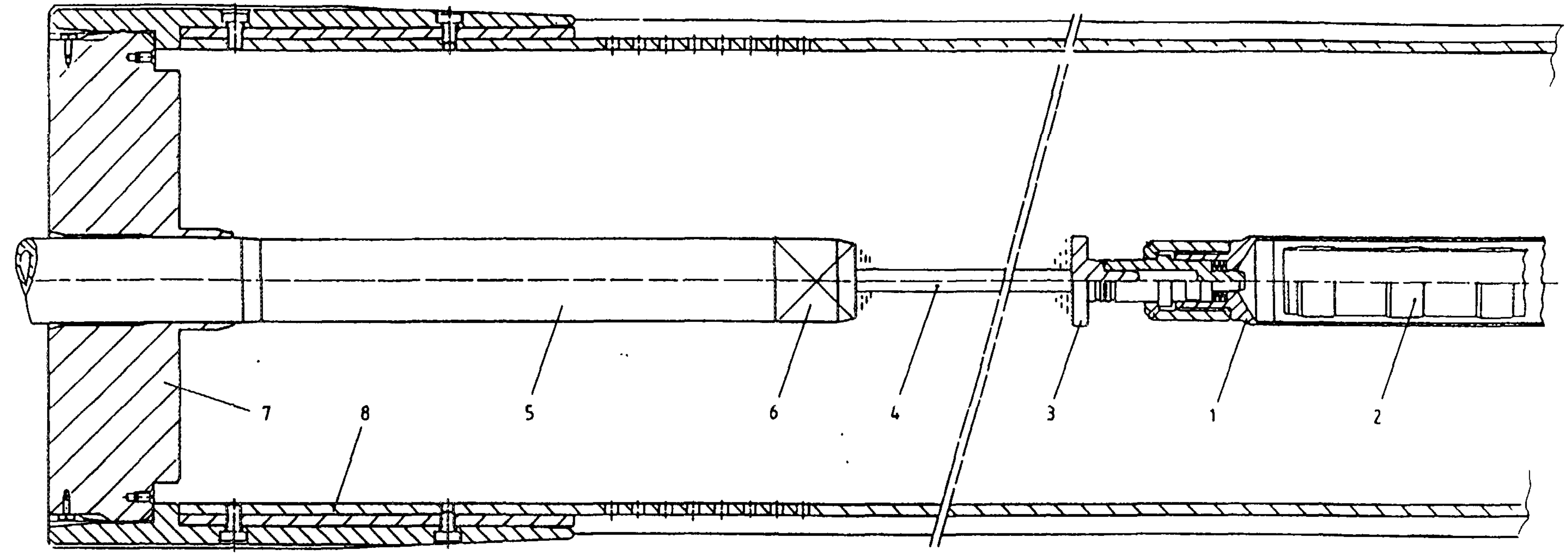


Fig. 6 **Hydraulic Control System Circuit Diagram**



1 Hohlkolben kpl. / Piston
2 Steuerelement / Control Rod Assembly
3 Hohlkolben-Befestigung / Piston Fortification
4 Rost mit Tragplatte / Grid support plate
5 Zentrierstift / Centering pin
6 Schraube / Bolt
7 Kernzelle - Rahmen / Core cell Frame
8 Bodenplatte / Base plate

Fig. 7



1	Oberteil- Steuerelement / Top section control element	5	US-Lanze / US-Lance
2	Hohlkolben / Hollow piston	6	Ultraschallsensor / Ultrasonic sensor
3	US-Reflektor / US-Reflector	7	Führungskanaldeckel / Guide duct cap
4	Steuerelement-Führungskanal / Control element guide duct	8	Kernzelle - Rahmen / Core cell frame

Fig. 8 **Drive Unit/Core Cell, Mechanical Design**

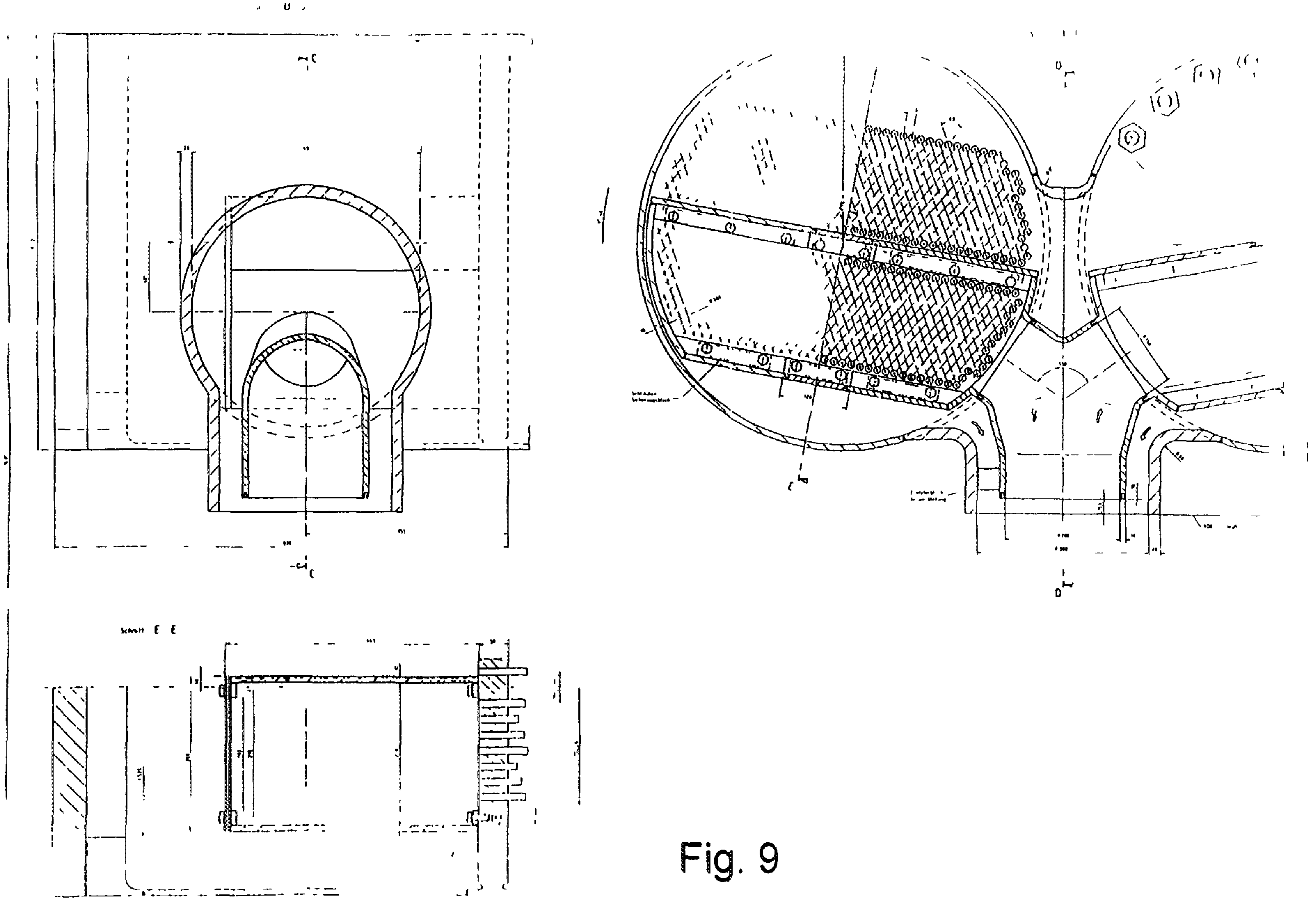


Fig. 9

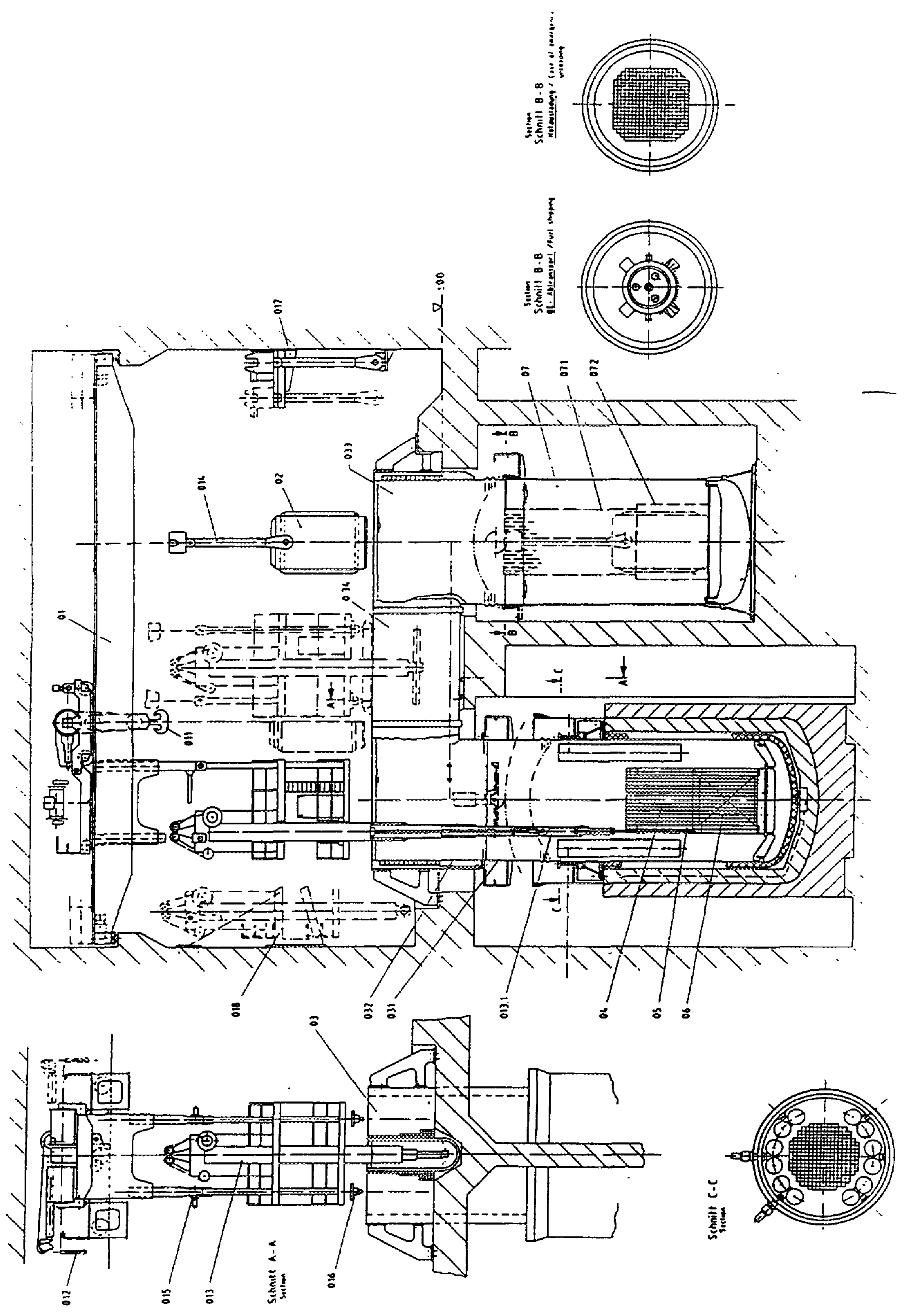

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Fig. 10

5. CONCLUSION

The mechanical design of a heating reactor must be influenced mainly by safety, reliability and cost. To fulfil all the requirements several steps of optimization are required. A consideration of the experience gained with big reactors is mandatory.

The utilization of core internal hydraulic control drives in connection with heating reactor core cells are a sound and already developed basic for a very attractive heating reactor concept which is distinguished by the potential of low capital cost.

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BASIC DESIGN DECISIONS FOR ADVANCED AST-TYPE NHRs

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Abstract

On the basis of the AST-500 reference design decisions and of the experience gained in the RF during the pilot NDHPs development and construction, the advanced NHR AST-500M has been developed recently by OKB Mechanical Engineering, as well as a whole series of heating and co-generation reactor plants of various unit power. All the designs represent enhanced safety reactor plants meeting the contemporary national requirements and international recommendations for nuclear plants of the new generation. The main objectives for the advanced NHR development are considered. New design decisions and engineering improvements are described briefly.

1. INTRODUCTION

To the end of the 1980s solid experience had been gained in the ex-USSR resulting from the extensive activity on the development, fabrication, construction and installation of several nuclear heating reactors of the AST-500 type on two twin-unit pilot stations near the large cities of Nizhny Novgorod (former Gorky) and Voronezh.

In-depth analyses of AST-500 NHR safety performed after the Chernobyl accident, activities on design upgrading in compliance with the revised safety codes and new licensing procedures and, at last, the joint work together with the IAEA safety review team during the Pre-OSART mission at the Gorky NDHP (1989) not only proved undoubtedly a sound basis of the design and of its safety concept, but also revealed the potential for further improvement of the design engineering decisions and characteristics.

Taking the AST-500 as a reference NHR design with in-depth-developed inherent safety features, a series of nuclear heating reactors of a unit power ranging from 30 to 600 MW has been developed recently at OKB Mechanical Engineering [1,2]. These designs rely upon the OKBM experience in the development of various types of nuclear reactors, including, besides NHRs, propulsion reactors for nuclear ice-breakers and marine ships. They accumulate the most promising engineering novelties and know how. They also take into account the contemporary world trends in development of the new generation of nuclear power reactors and the generally recognized international requirements for the safety of advanced NPPs.

2. MAIN OBJECTIVES FOR AST-500 IMPROVEMENTS

The following principal objectives have been adopted for the AST-500 NHR upgrading and for the development of its advanced modifications:

- simplification and standardization of engineering decisions in the framework of the whole series of AST-type reactor plants;

- gaining better economics;
- improving the basic equipment reliability and life-time extension;
- enhancement of safety by improving the reactor immunity to plant personnel errors, as well as to external and internal impacts.

The following indicators were accepted as the design goals (compared to the reference design characteristics):

Goal	Indicator
Extension of the reactor pressure vessel and main equipment service life	1.5 - 2 times
Reactor power uprating at the same dimensions	1.2 times
Fuel life-time extension	1.3 times
Fuel burn-up increase	1.7 times
Cut of equipment items in engineered safety systems	1.4-1.5 times
Decrease of auxiliary power, including that for responsible consumers	1.5 times
Cut of electrical controlling safety systems (CSS)	transition from 3-channel to 2-channel structure of CSS
Enhancement of plant immunity to common cause failures (fire, flooding, etc.)	use of self-actuated passive systems
Increase of design margins for strength	with account of hydraulic shocks and emergency increase of pressure in confinement systems
Standardization of engineering decisions for all NHRs in series	90%

The major portion of the advanced design decisions, particularly those regarding the reactor plant components service life extension, reliability and safety enhancement has been already implemented in the upgraded design of the Voronezh NDHP.

Besides, the results of the activity on the improvement of AST-500 reactor characteristics were used in the new designs of small and medium power reactors for electricity and heat co-generation power plants being under development now.

3. IMPROVING AST-500 NHR RELIABILITY AND ECONOMICS

To illustrate the progress in the reference design (AST-500) advancement the following decisions may be highlighted.

3.1. Increase of heat output to the heating grid

The reactor uprating potential is validated for the pilot plant AST-500 with no need for changing the composition of the plant equipment. This goal can be attained at the expense of working parameter variation in periods when the grid water temperature required for the consumer is below the design value. The reactor thermal power can be increased up to 600 MW at the grid water temperature decrease below 128/32°C (in summer, spring and autumn).

Increase in grid water flow by 25% G_{nom}, in grid heat exchanger surface by 25%, as well as in the primary heat exchanger surface allow to provide the delivery of 600 MW into the grid for the entire heating period. Investigations are underway of the capability to further increase the total heat output without deterioration of the reactor plant safety level (Fig.1). A considerable increase in heat-exchanger surface built into the reactor without changing the reactor dimensions has been achieved by the use of smaller tubes with a close pitch (tubes of 10x1.2 mm, square array with 14 mm pitch).

3.2. Reactor plant design simplification

The lower main joint of the reactor pressure vessel is eliminated in the advanced NHR along with the related complex of in-service control means.

The reactor plant systems simplification may be illustrated with the example of the primary coolant purification system (Fig.2). In the pilot plants the reliability of this system operation determined the reliability of the heat delivery from the NDHP to the consumers, which can be explained by the following reasons:

- in the AST-500 the VVER-440 reactor CRDM of the ARK-type were used with the need of assured filling with water of all internal cavities on the primary circuit side;
- in the integral reactor, the aforementioned requirement is met by continuous feeding of sealing water into the drives by pumps of the reactor coolant purification system;
- an emergency protection signal is generated at loss of sealing water flow.

The reactor coolant purification system is the only system in the pilot AST-500 having a continuous circulation of primary water beyond the reactor boundary. The requirements for the flowrate in the system (and hence, the diameter of related tubes and penetrations in the reactor vessel), and the continuous mode of operation are dictated by the CRDM operation and not by maintaining the reactor coolant chemistry.

The operational experience gained from the OKBM-designed marine reactors testifies a sufficiency of intermittent operation of the primary coolant purification system and a possibility to reduce flow in the system.

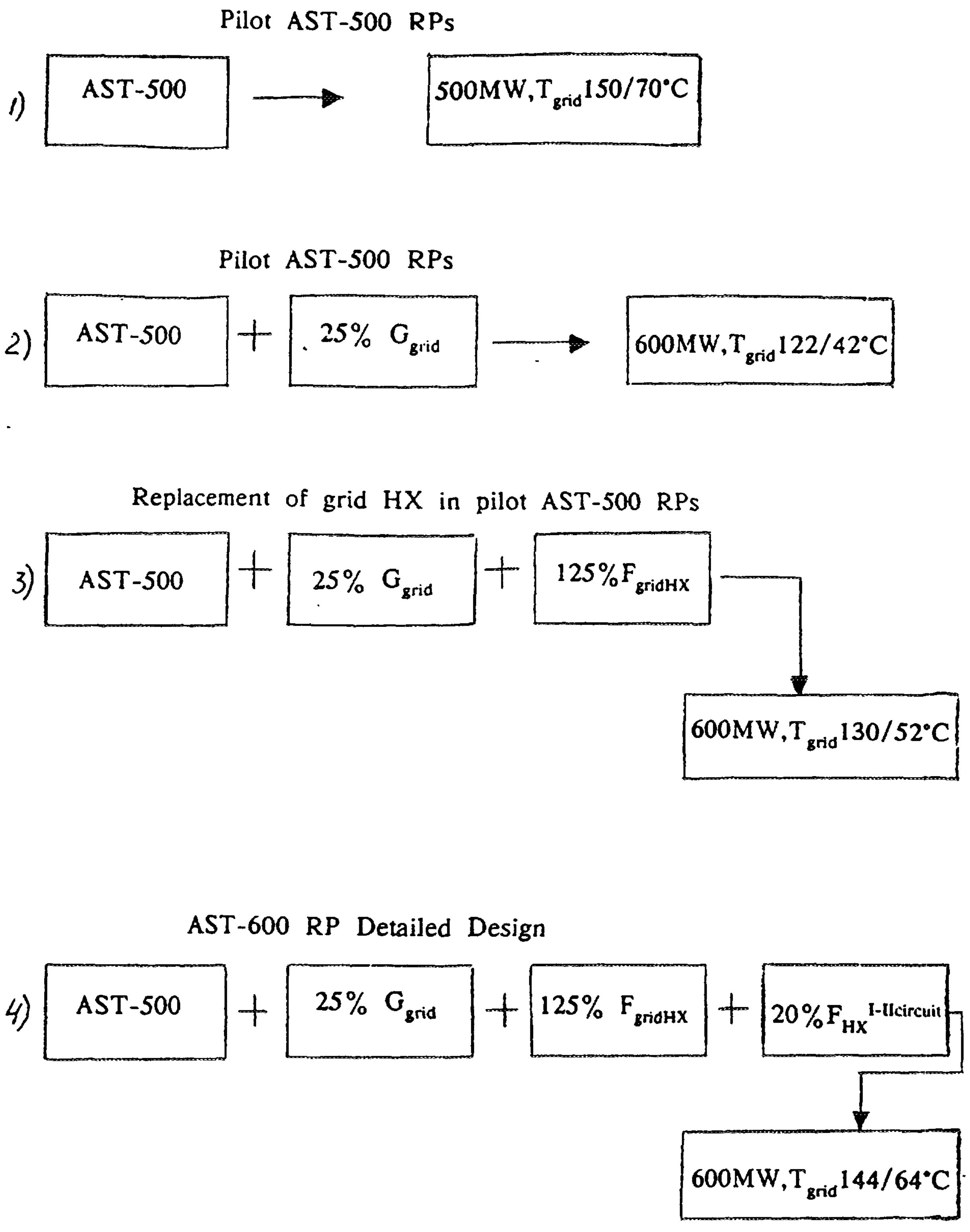


Fig.1. AST-NHR Heat Output Uprating Potential

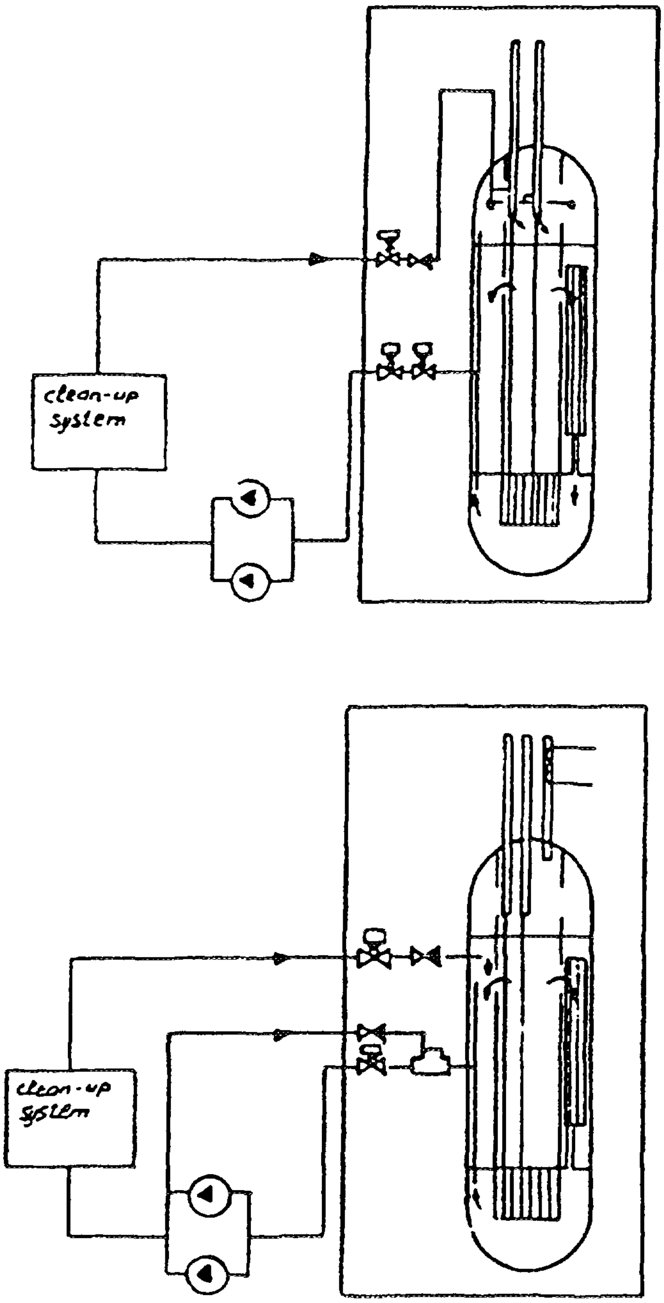


Fig.2. AST-500 Reactor Coolant Purification System Upgrading

The following decisions were adopted to solve the problem:

- upgraded CRDM were developed and tested successfully; they work in a steam-gas medium; besides, their service life was extended;
- the diameters of penetrations in the reactor pressure vessel and of pipelines in the system are decreased (to 40 mm for the penetration);
- in-reactor header and sealing water distributing pipes are eliminated;
- the periodicity of the system connection to the reactor is not more than once per three months;
- the system's pump delivery flow and consumed power are reduced 2 times. A modernized pump of the "impeller upward" type is used without a special gas removal system;
- a self-actuated hydro-controlled valve is used in the intake pipeline of the system within the guard vessel boundaries.

4. SAFETY ENHANCEMENT AND USE OF SELF-ACTUATED SYSTEMS AND SAFETY DEVICES

4.1. Self-actuated ERHR channel on the reactor

The followings factors gave the impetus for development of the additional ERHR channel:

- strive for independence and diversity of ERHR trains, that is an ERHR channel was needed which would be completely arranged within the guard vessel and which would not be connected to secondary loops. It is a structural accessory of the reactor and uses a self-actuation principle;
- evidence of testing the CRDM in a steam-gas medium that showed practically zero heat transfer into the CRDM cooling circuit.

So the idea emerged, and was then realized, to install immediately on the reactor closure head the condensers whose size is similar to that for the CRDM and whose inner cavity is connected like the CRDM with the steam-gas volume of the in-reactor pressurizer.

In the stand-by mode the partial pressure of the gas in the condenser's cavity exceeds 0.9 P_1, thermal losses into the cooling circuit are practically absent. The condenser self-actuation is provided by gas blown off through the self-actuated device in response to signals of a primary pressure build-up or a coolant level lowering in the reactor.

4.2. Self-actuated devices for safety systems

The premises for the development and implementation of these devices were the following (Fig. 3):

- preservation of their operability at external and internal impacts, including fires, flooding, seismic effects, etc. Impossibility to disable the fulfillment of protective functions by the plant personnel;
- design simplicity and relatively low cost;

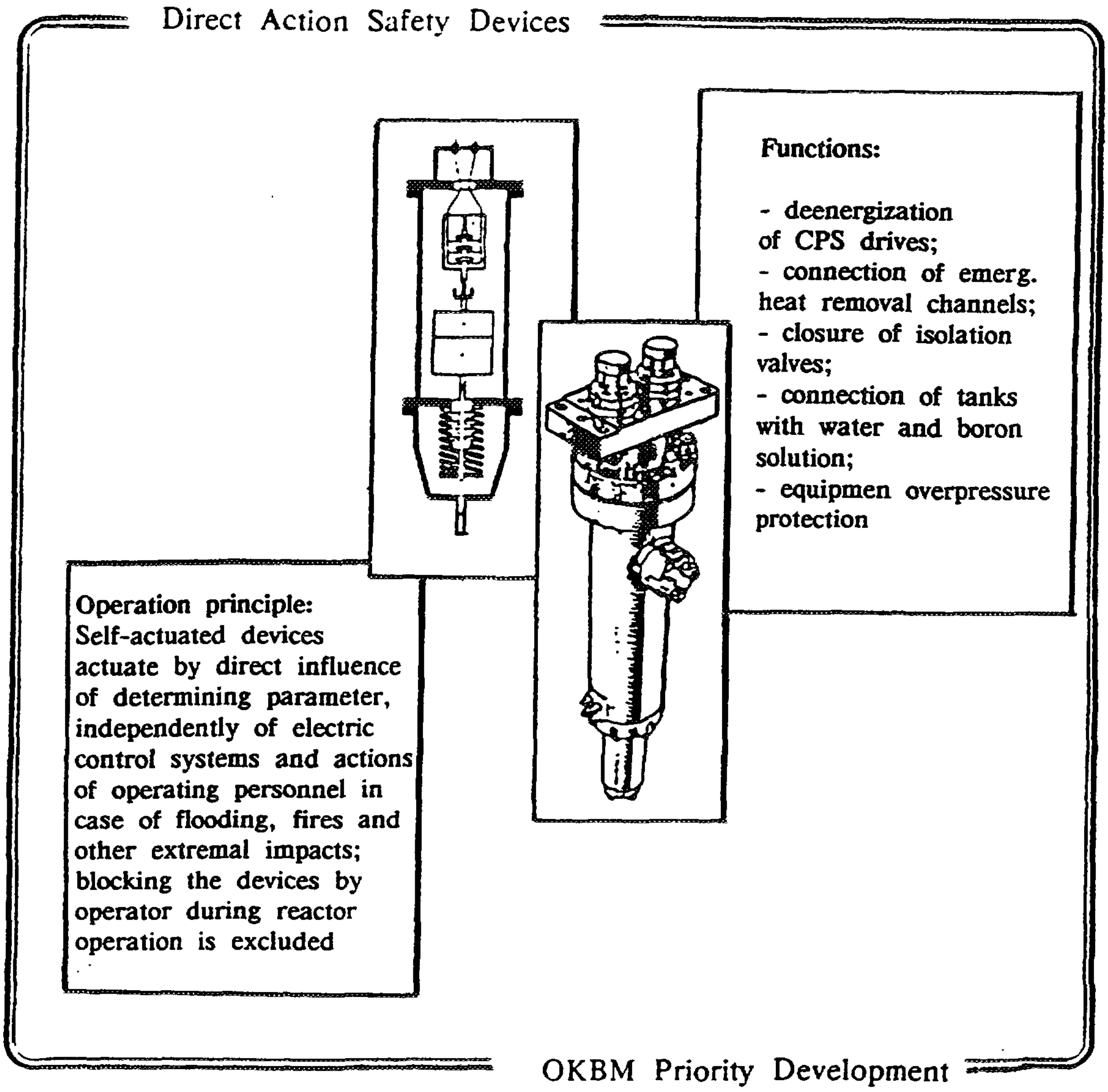


Fig.3. Direct Action Safety Devices

7. PRIMARY CIRCUIT OVERPRESSURE PROTECTION

The heat removal principle for the primary system overpressure protection without dumping of radioactive medium is used in the design of advanced AST, as well as in the pilot AST-500. This became possible due to the reliable means used for reactor shut down and for residual heat removal, and due to the properties intrinsic for the given reactors.

This design approach meets the modern requirements of regulations for safety of NPPs.

8. IN-REACTOR CONTROL

The following parameters are monitored continuously by the control means provided for the reactor:

- neutron flux density (neutron power, reactor period and reactivity); there are 8 ionization chamber suspended in the reactor;
- coolant level in the reactor; there are 6 level indicators;
- pressure in the reactor; pressure gauges are located in the guard vessel;
- temperature in the reactor; thermocouples are arranged at the heat exchanger inlet (core outlet);
- energy release and temperature at the core inlet/outlet; instrumentation probes are located immediately in the core.

Besides, the radioactivity of the primary circuit medium (in the steam-gas plenum, coolant), the primary coolant chemistry and gas regime indicators (hydrogen, oxygen and the others), leak-tightness of the main reactor joint and primary circuit valves are monitored on-line.

The metal of the reactor vessel is checked when the reactor is shut down.

9. FUEL HANDLING AND STORAGE

Fuel is reloaded by means of the universal refuelling machine ("dry" method), the same as in the pilot units.

The refuelling machine moves on a rail track laid on the reactor hall floor and attends the area between rails, where the reactor pit, the reactor internals pits and the spent fuel storage pool are arranged.

The refuelling machine is intended to perform the following operations:

- transport of the reactor removable internals (control rod guide tubes connecting device unit, in-vessel barrel) from the reactor to the corresponding pits and backwards;
- transport of fuel assemblies inside the reactor core, between the core and the storage pool, inside the storage pool, (with fuel cladding integrity control), between the storage pool and universal seat (shipping cask, etc.).

Spent fuel is held up in the storage pool for radioactivity and residual power decreases, then it is removed in special container-cars from the site to a reprocessing plant.

10. PRIMARY CIRCUIT WATER-GAS REGIME

The steam-gas pressurizer provided for the primary circuit pressure control uses a explosion-proof helium-hydrogen mixture (96% He + 4% H_2). The hydrogen content in the primary coolant is specified (not less than 2 g/kg at core inlet) and monitored. A special system is provided for the primary circuit make-up with a He-H_2 mixture. The given hydrogen content in the coolant ensures water radiolysis suppression in the reactor.

In LOCAs the gas mixture is blown off from the pressurizer through the bubbler.

11. INSPECTIONS AND REPAIRS

The reactor unit design allows to perform location and plugging failed tubes in the primary heat exchanger, as well as a tubing section replacement after their lifetime is expired.

The reactor vessel metal and the weld joints are tested (using a particular ultrasonic technique) when the reactor is shut down.

12. DECOMMISSIONING

The design service life of the AST power units was initially determined to be 30 years. However, the design peculiarities of this reactor plant (thick water layer around the core) provide low irradiation and induced radioactivity of the reactor internal structures and construction concrete which gives ground for an extension of the reactor unit lifetime up to 50-60 years. It also facilitates its decommissioning.

Because the service life of the reactor building structures is determined to be as much as 100 years, a possibility is provided for dismantling the reactor units and installation of new ones.

Taking into consideration the aforementioned peculiarities of the reactor unit, equipment dismantling work could be started immediately following its operation life completion and reactor unit shut down (without delay for a plant conservation and observation period).

The quantity of radioactive materials during decommissioning of the AST unit subjected to disposal amounts to 50 m^3.

13. CONCLUSION

The AST-500 reactor plant, being the first in a family of integral PWRs developed by OKBM, represents a new generation reactor plant of passive safety. It is mastered in serial production and is used as a part of Voronezh NDHP under construction.

- positive operation experience for prototypes and similar devices used in the marine reactors;
- possibility to solve on a system level the problem of reactor emergency protection, emergency cooling and radioactivity confining systems actuation.

So, the main goal of implementing such devices is to improve the controlling safety systems reliability, to simultaneously reduce a number of electrical controlling safety systems' trains and to reduce their price.

5. DESIGN DECISIONS CONTINUITY AND STANDARDIZATION

For the development of a series of AST-type reactors the principle of design decision standardization was adopted. Among the advanced decisions which are to be implemented in the pilot NHRs, the following ones are developed:

- upgraded core with four fuel cycles of two-year duration;
- advanced protective system's control and power supply subsystems;
- use of self-actuated devices;
- equipment life extension;
- heat output increase, etc.

All AST-type designs contain a set of proven engineering decisions which are implemented into small power nuclear plants that are being developed now, including ones for electricity and heat co-generation. Particularly, decisions ensuring enhanced safety were implemented which allows to realize a single concept of the new generation reactors.

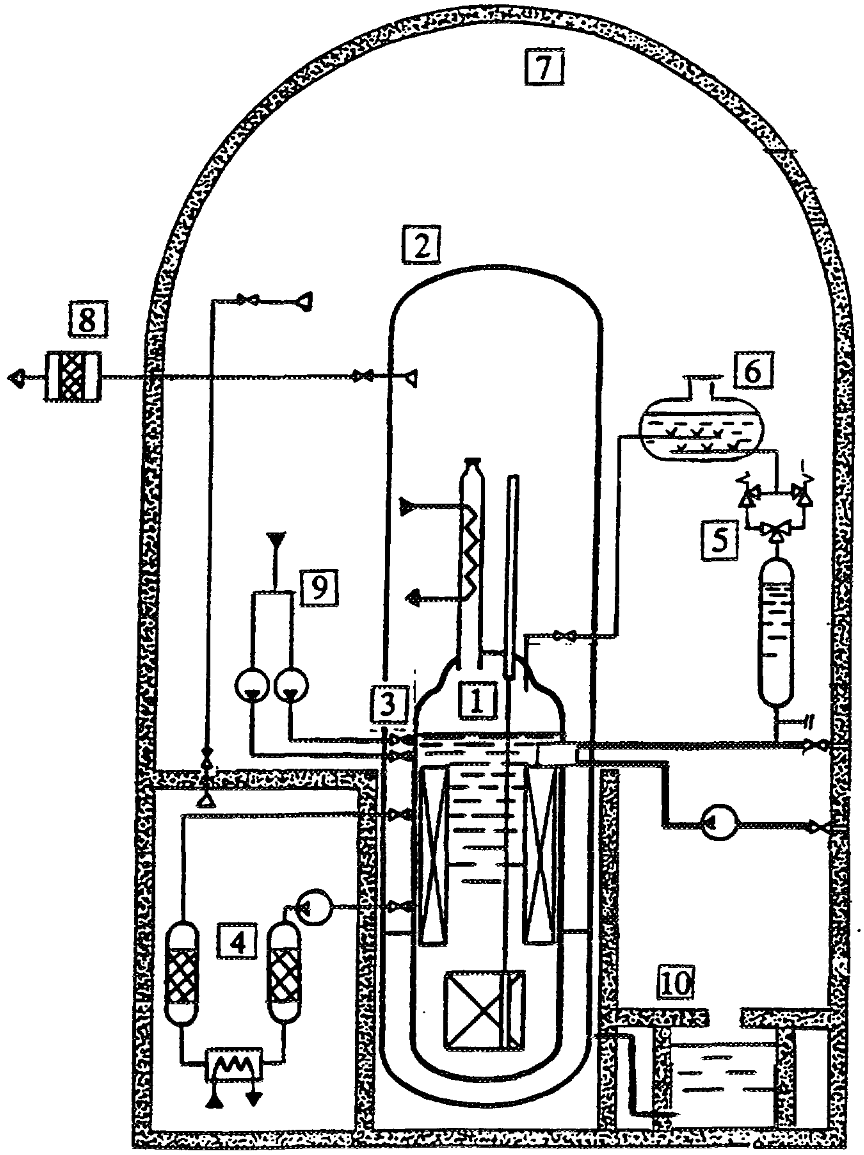
This regards the design of the reactor unit and the basic equipment, structures and equipment of passive safety systems, spectrum of self-actuated devices used, emergency control algorithms and the usage of back-up systems and devices.

6. CONFINEMENT AND MITIGATION OF RADIOACTIVE DISCHARGES, CORE PROTECTION AGAINST UNCOVERING

The following protective barriers (Fig.4) are used effectively to confine radioactive products, and to prevent and mitigate their release at accidents [3]:

- "cold" UO_2 fuel;
- fuel element claddings;
- leak-tight reactor with large margins for strength;
- guard vessel;
- intermediate circuit, designed for primary coolant pressure;
- leak-tight compartments, concrete reactor pit;
- bubbling of emergency discharges before their release into the containment;
- containment with aerosol filters;
- decreased radioactivity of emergency discharges due to keeping the core covered with water.

The high safety level of this reactor plant is recognized by national review and regulatory bodies, and is confirmed by the international safety review team (IAEA Pre-OSART mission).



1. Reactor
2. Guard vessel
3. Quick-acting licalization valves
4. Leak-tight rooms
5. PORV
6. Bubbler
7. Containment
8. Filter
9. 1-ry circuit makeup system
10. Water storage tank

Fig.4. Radioactivity Release Mitigation and Core Uncovery Prevention Means

The basic engineering decisions used in the AST-500 were then borrowed and further developed in a series of AST-type advanced NHRs ranging from 30 to 600 MW. Therewith, problems are being solved for assuring AST plant competitiveness compared with fossil-fuelled heat sources, for simplification of plant systems, equipment and operation, along with the provision of a safety level exceeding that already reached and substantiated for the pilot AST-500 NHR.

A significant portion of the decisions that have been developed for the advanced AST is planned to be used in the pilot NHRs.

Simultaneously, the whole set of engineering decisions providing reliability, economic efficiency and safety of the AST reactor is realized on a system level in reactor plant designs for nuclear power, particularly for the electricity and heat co-generation plants of the ATETS-150 type.

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MECHANICAL AND STRUCTURAL DESIGN OF THE 200 MW NUCLEAR HEATING REACTOR (NHR-200)

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Abstract

In this paper, some mechanical and structural design features of NHR-200 are briefly described focussing on: design and technical features of internals; a new type hydraulic control rod driven system; spent fuel storage around the active core; design and safety characteristics of pressure vessel; discussion on in-service inspection of pressure vessel.

1. INTRODUCTION

- The 200MWt Nuclear Heating Reactor (NHR-200) was designed, following the mechanical and structural design experiences of pressurized water reactors (PWR), boiling water reactors (BWR) and nuclear heating reactors (NHR), and the experience from operating the 5MWt test heating reactor (NHR-5) with excellent performance.

- The main criteria for the NHR-200 design are the corresponding chapters or clauses in the ASME code, the relevant National Standards and the relevant Stipulations for the Qinshan 300MW nuclear power plant.

- The main components of the NHR-200 [Fig. 1], i.e., reactor pressure vessel (RPV) and internals, are described in this paper.

2. THE DESIGN AND TECHNICAL FEATURES OF INTERNALS

A detailed description of NHR-200 internals is shown in Fig. 2.

- **The design of internals**

Self-pressurized performance, natural circulation and integrated arrangement are adopted.

Primary components, such as primary heat exchangers (PHEs) and hydraulic control rod drives etc., are fully incorporated in the RPV. The pressure of the primary coolant is maintained by the volume variation of the space occupied by a pressurizing gas and vapor in the upper part inside the RPV. The active core and the PHEs are separately arranged inside and outside the core barrel. The barrel stands between the core and the RPV to split the annular gap into two parts that serve as the circuit of primary coolant natural circulation.

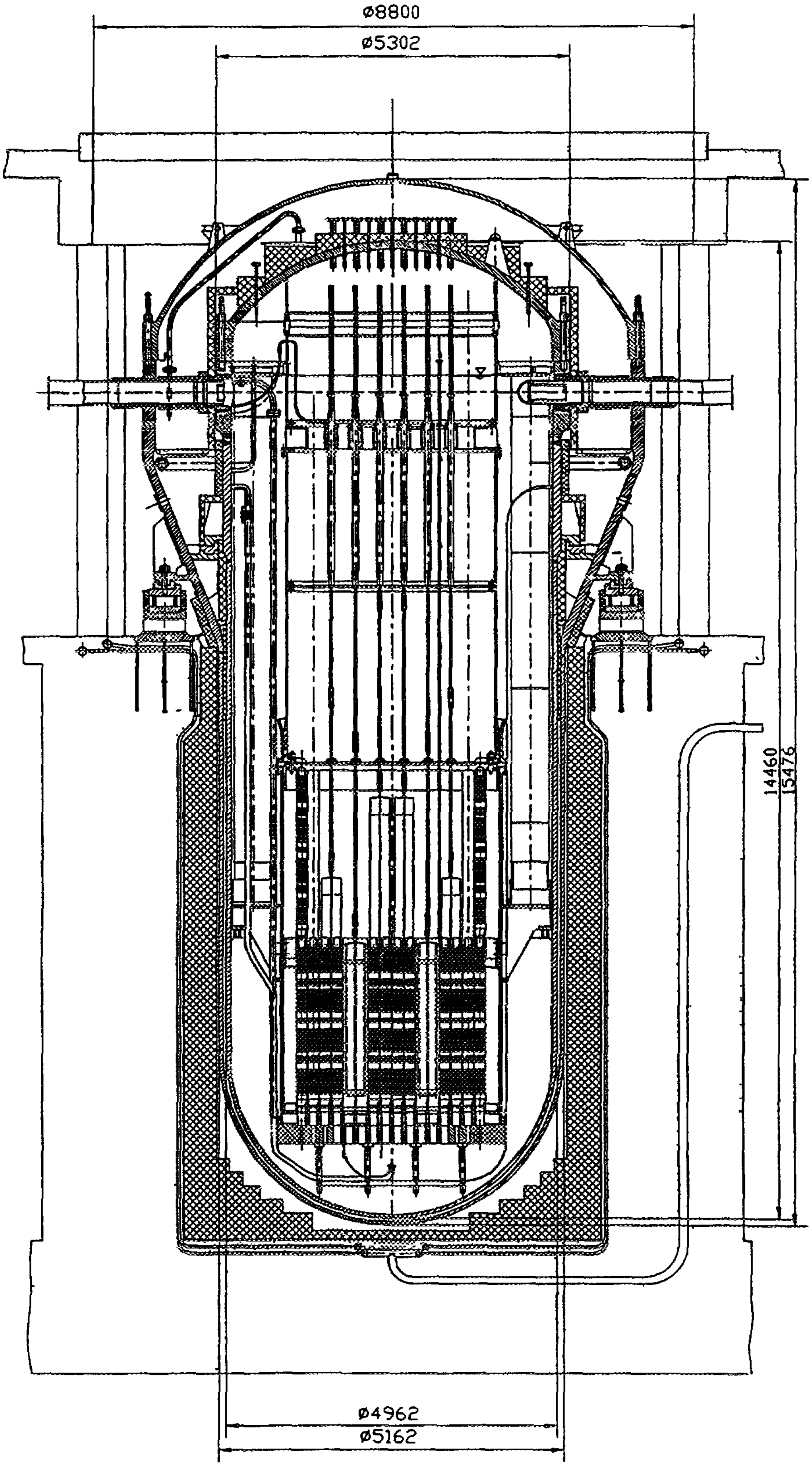

Fig.1 The primary system arrangement

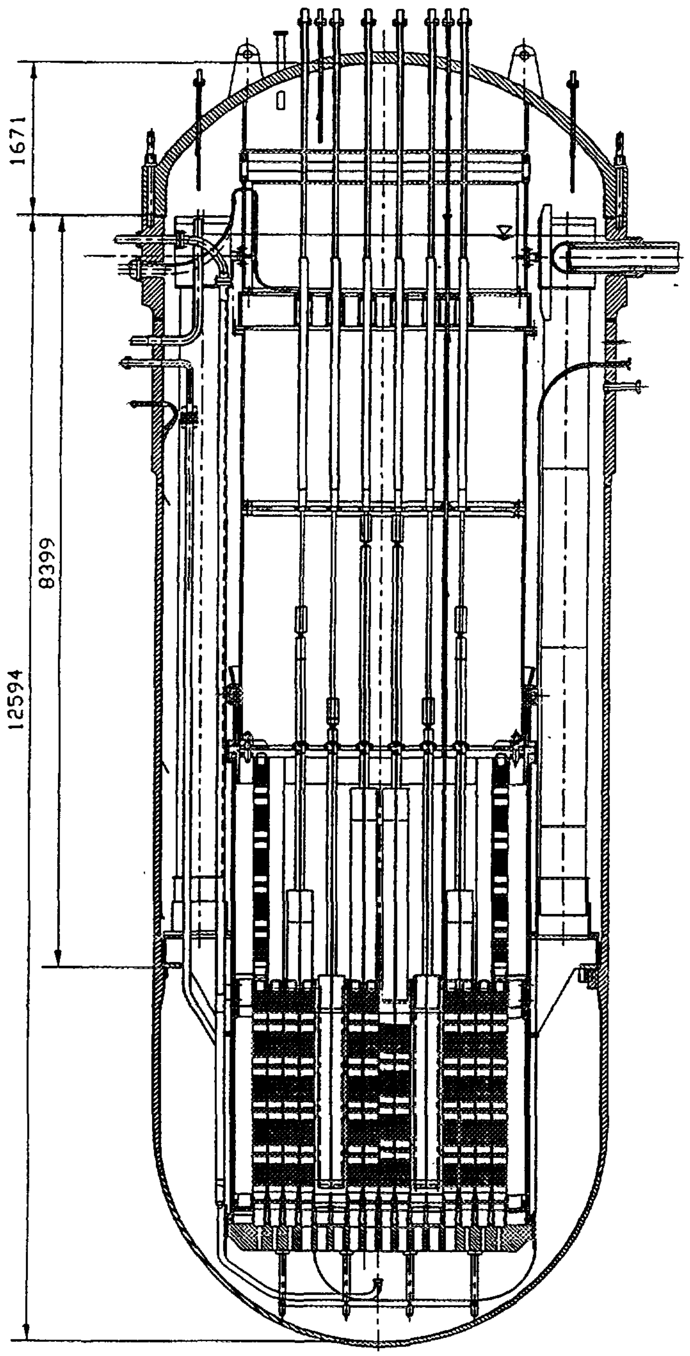

Fig.2 The pressure vessel and internals

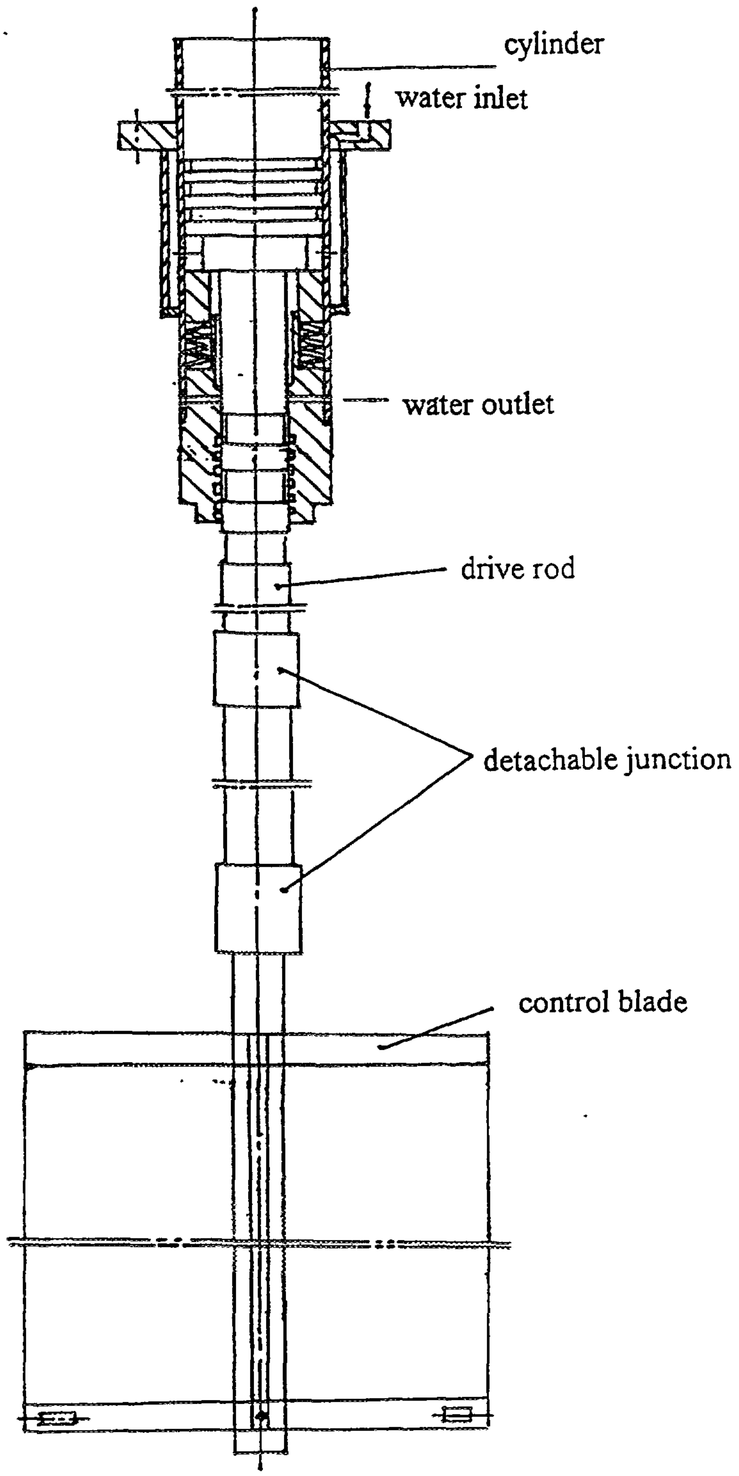

Fig.3 The control rod system

Fuel channels are used

A fuel assembly consists of a fuel bundle and a Zircaloy-4 channel. The active core contains 96 assemblies which are arranged and supported in a way that is similar to that in BWRs. There are 110 lattices for spent fuel storage around the active core. And there are 32 cruciform type control rods in the active core. Each of them inserts into the gap between four fuel channels which serve as the guide for the control blades. The cruciform type control rod is driven by a hydraulic system which follows the fail-safe principle.

- **Three technical features of internals**

A new type of hydraulic control rod drive system is adopted

A new type of control rod drive system has been developed that can be fully incorporated into the RPV. The control rod system consists of a stationary hydraulic drive cylinder and a movable drive rod that carries the absorber blades [Fig. 3]. Primary coolant, as the actuating medium, is pumped to a set of control armatures, holds and drives the rods.

The differences between control rod drive of NHR-200 and that of NHR-5 can be summarized as follows:

1) The step drive cylinder was changed to "gap-hole" type from "hole-hole" type. It makes the alignment of the cylinder and rod, and the fabrication more easy.
2) The cylinder is fixed at the upper core grid plate. The junction between drive rod and absorber blades is detachable.
3) The absorber drive part is a drive rod rather than a drive cylinder. The drive rod has many indents on its outer surface with steps of 50 mm.

Spent fuel storage around the active core

The storage of the spent fuel assemblies around the active core is available because the integrated arrangement was adopted, operation temperature and pressure are low, and because the number of damaged fuel element would be small. After storage in the RPV for a long time, decay heat and radioactivity of the spent fuel assemblies would be greatly reduced. It could simplify the refueling equipment, so the safety and economics of the reactor would be improved.

In many countries, the concept of spent fuel storage inside RPV is under development to increase the burn-up of fuel, and to eliminate post-processing. For the NHR-200, the storage of the spent fuel around the active core aims at increasing the average burn-up and at reducing the differences of burn-up in fuel assemblies.

The experiences of reactor operation and the results of studies regarding fuel cladding corrosion by some researchers have proven that in-core spent fuel storage is feasible. In Canada, some assemblies, that had been stored outside the RPV for 10, 9, 5 years respectively, were put into the reactor again with a power density of 4.2kW/m, gave a good performance. The fuel assemblies of the NPD reactor had been working well for 16 years under full power operating condition. It is pointed out that after storage in water at a temperature of 300°C for 100 years, the thickness of fuel cladding decreases by only 10% of its original value.

Anti-seismic behavior of internals

The chimney for primary coolant natural circulation is at least 6 m, therefore, the height of the barrel would be larger. The hydraulic cylinders of the control drives are installed in the upper part of the barrel, above the core. So they can be easily examined and repaired, but the distance between the drive cylinder and the absorber will be increased. Deformation and higher stresses might occur in the drive rods because guidance is only provided by the fuel channels in the core. A series of guide rolls is arranged in the upper core grid to improve the anti-seismic behavior. In this way deformation and stresses of the drives could be limited.

A large number of nonlinear problems was related to the complicated structure. Based on full dynamic analyses and calculations, two test experiments are planned. The first is a core cell including a control rod drive and its supporting guide structure with scale of 1:1 to the actual one. It is considered to test and verify the anti-seismic capability of the internals and the reliability of control rod scram under earthquake conditions.

The second experiment, with model scale 1:10 to actual the size of the structures, including RPV, containment, internals and PHEs, etc., is designed to test the overall anti-seismic behavior of the reactor and to obtain the necessary input data for the first test.

3. DESIGN OF THE RPV AND DISCUSSION OF IN-SERVICE INSPECTION

- **Design of the RPV**

According to the operating parameters of the NHR-200, the arrangement of internals and requirements for installation and repair, the RPV is designed as shown in Fig. 4. The main parameters of the RPV are:

operating pressure	2.5 MPa
operating temperature (max.)	213°C
design pressure	3.1 MPa
design temperature	250°C
inner diameter	4820 mm
thickness of RPV	65 mm
outer diameter of main flange	5302 mm
main bolt size	M78x4
number of main bolts	88

- **The material of the RPV is SA516-70**

The selection depends briefly on the following:

- SA516-70 has low strength, high toughness and can be easily welded.
- The operating pressures and temperatures are low, the wall thickness of the RPV is smaller and the mechanical properties of normalized material are good.
- The integral fast neutron flux affecting the RPV is small. Over the lifetime of the reactor, the integral flux of fast neutrons will be no more than 10^{16}n/cm^2.
- SA516-70 has been used in some PWR and BWRs.

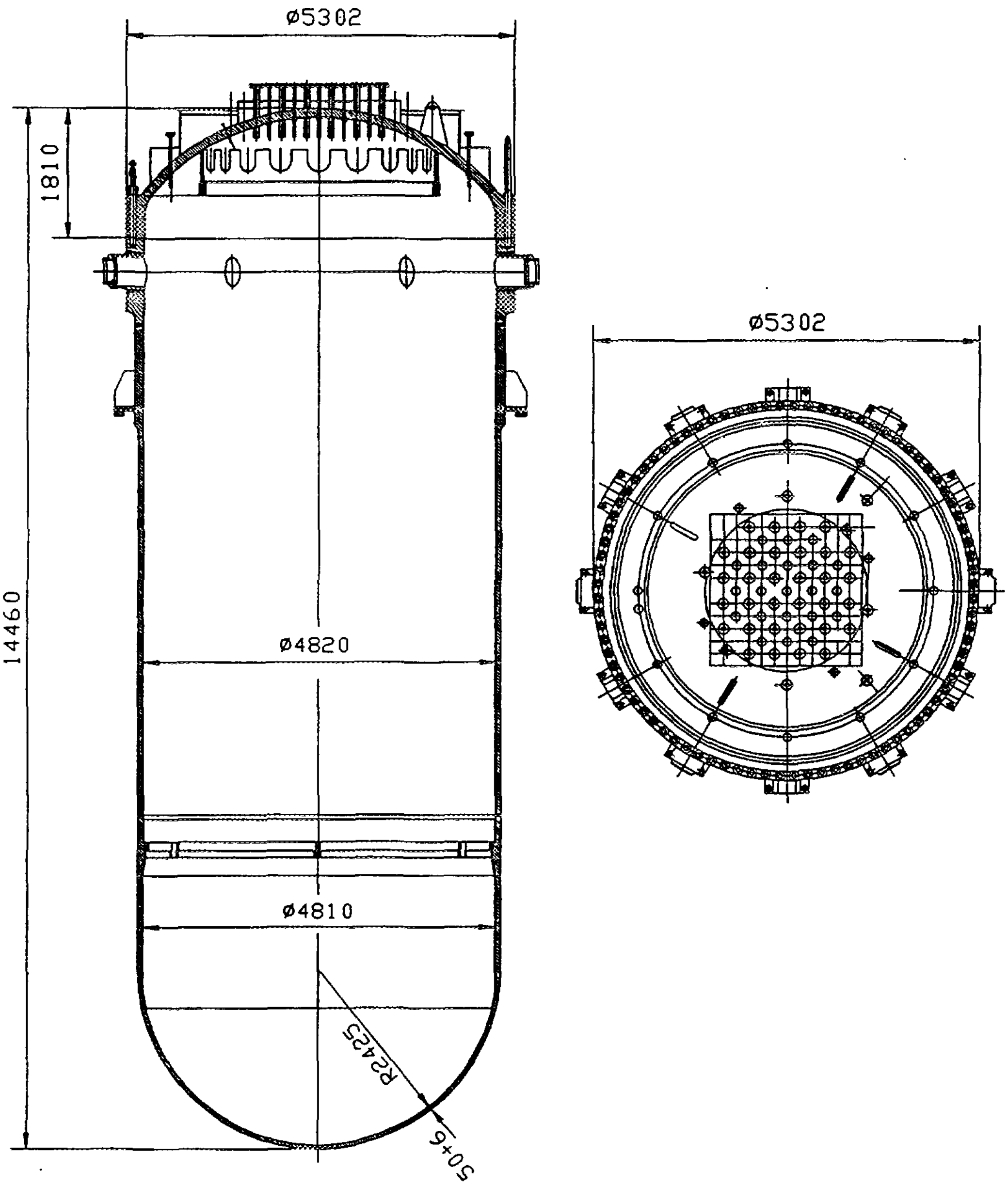

Fig.4 The NHR-200 pressure vessel

The material of the RPV of an American reactor in Puerto Rico was SA516-70, being referred to as A212-13 in that time. It is still allowed by ASME to be used as material for components belonging to class A.

- **Safety characteristics of the RPV**

High toughness

SA516-70 has a high toughness itself. Even more, because the integral neutron flux to the RPV is low over the lifetime, the material will retain its good performance at any time.

Low stress level

The overall membrane and bend stresses of the RPV are no more than 120 MPa. They are much lower than in case of PWRs. For cracks of any dimension, its propagation speed is only 5%-10% of that of the RPV in PWRs. Therefore the damage probability is much lower.

The RPV allows application of the LBB (Leakage Before Break) principle

The LBB behavior of the RPV depends on the toughness of the material, its stress level and its wall thickness. The material of the NHR-200 RPV has good toughness, the stress level is low and the wall thickness is small. According to the fracture analysis, the barrel of the RPV and closure meets the LBB principle.

- **Discussion of in-service inspection for NHR-200 RPV**

Main targets of in-service inspection

Maintaining the integrity of the primary pressure boundary, thus avoiding the loss of primary coolant and release of radioactivity.

- For reactors of different type, or of the same type, but with different criteria, the requirements and methods of in-service inspection are different.
- The metallic containment fits tightly around the RPV. Because of the small volume, even if the most serious accidental break occurs on the bottom of the RPV, the leakage from the pressure boundary will be limited. The water inside the RPV still covers the active core. There is no possibility that the accident of core melt down occurs. This safety characteristics of NHR-200 are different from PWRs.
- The RPV of NHR-200 confirms to the requirements of LBB.

For the reasons mentioned above, and from the safety point of view, the main task of in-service inspection for the NHR-200 is the inspection of leakage from the primary pressure boundary. X-ray and ultra-sonic inspections are not necessary.

- With the traditional method of in-service inspection, flaw detection applies only in some local parts of the primary pressure boundary. Even more, some cracks and flaws could not be found due to the limited sensitivity of inspection devices. Therefore, the break probability of RPV will be reduced only to a certain extent, but the accident can not be avoided absolutely. For PWRs, the accident of RPV break is not acceptable. But for NHR-200, it cannot lead to a serious accident. It is thus clear that the overall safety of NHR-200 is greatly improved.

INVESTIGATION OF IN SERVICE INSPECTION FOR PRESSURE VESSEL OF THE 200 MW NUCLEAR HEATING REACTOR

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Abstract

The Nuclear District Heating Reactor (NHR) is a new type of reactor. There are some differences in the arrangement of the primary circuit components and in safety features between NHR and PWR or other reactors. In this paper the safety features of the 200 MW NHR are described. The failure probability, the LBB property and the in-service inspection requirement for the 200 MW NHR pressure vessel are also discussed.

1. INTRODUCTION

An integrated arrangement of the primary components is adopted in the design of the 200 MWt Nuclear Heating Reactor (NHR-200). The reactor core and the primary heat exchangers are incorporated in the Reactor Pressure Vessel (RPV). There are no pumps and main pipes in the primary circuit. All the heat power generated in the active core is removed by natural circulation. Therefore, the pressure vessel represents the main part of the primary pressure boundary. Based on these features, a compact metallic containment is designed to fit around with the pressure vessel. Accident analyses show that the active core remains water flooded under any operating or accident condition. A core-melt accident is impossible.

Because the arrangement of the primary system and the safety characteristics are quite different from those of PWRs, the requirements and the adopted methods for in-service inspection for the NHR-200 are to be modified accordingly.

2. PRIMARY SYSTEM ARRANGEMENT, PRESSURE VESSEL AND METALLIC CONTAINMENT OF THE NHR-200

2.1. The primary system arrangement

Fig.1 shows the primary system arrangement of the NHR-200. A compact metallic containment, fitted tightly around the pressure vessel, is arranged inside the reinforced concrete reactor building.

The pressure vessel is supported inside the containment vessel which is located inside the concrete biologic shielding. The control units of the hydraulic control rod drive system, as well as various water or gas pipes, electric cables, thermal insulation of pressure vessel and the isolating valves of the containment, etc., are arranged in the annular space between the pressure vessel and the containment.

On every pipe penetrating the containment, there are two isolating valves located inside and outside of the containment, respectively. The drive-means of the two valves are different. Due to the small volume of the containment, if a break occurs in any pipe inside

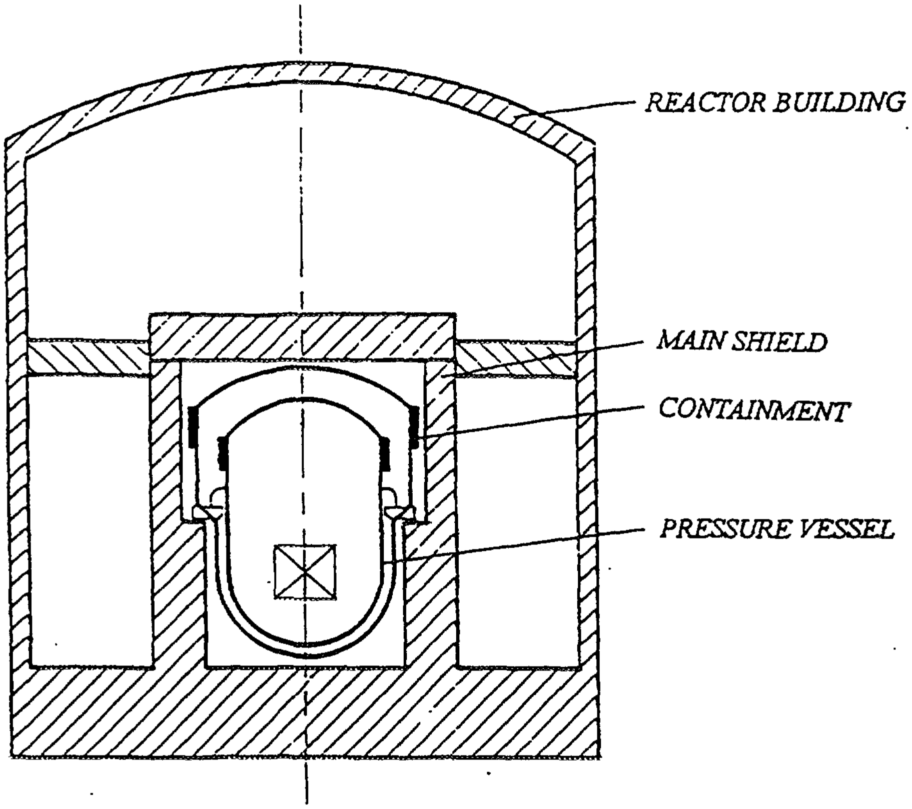


Fig.1 The primary system arrangement of the NHR-200

the containment, or even at the bottom of the pressure vessel, the leakage from the primary pressure boundary will be always limited. The active core will still be covered by coolant. It is impossible that a core-melt accident would take place.

2.2. Reactor pressure vessel

The structure of the pressure vessel is illustrated in Fig. 2. The RPV material is SA516-70. The main part of the pressure vessel is a cylindrical shell with a thin wall. It is supported by 12 support brackets on the conical part of the containment. The head closure and the cylindrical shell are connected together with 84 main studs (*M804*). The main flange is sealed with two metallic "O" rings. The inner surface of the pressure vessel wall is a layer of stainless steel cladding with a thickness of 6 mm to prevent corrosion.

The main parameters of the pressure vessel are as follows:

operating pressure	2.5 MPa
design pressure	3.1 MPa
operating temperature	213°C
design temperature	250°C
wall thickness	65 mm
inner diameter	4,820 mm
outer diameter of flange	5,320 mm
overall height	14,460 mm
total weight	2.04 x 10^5 kg

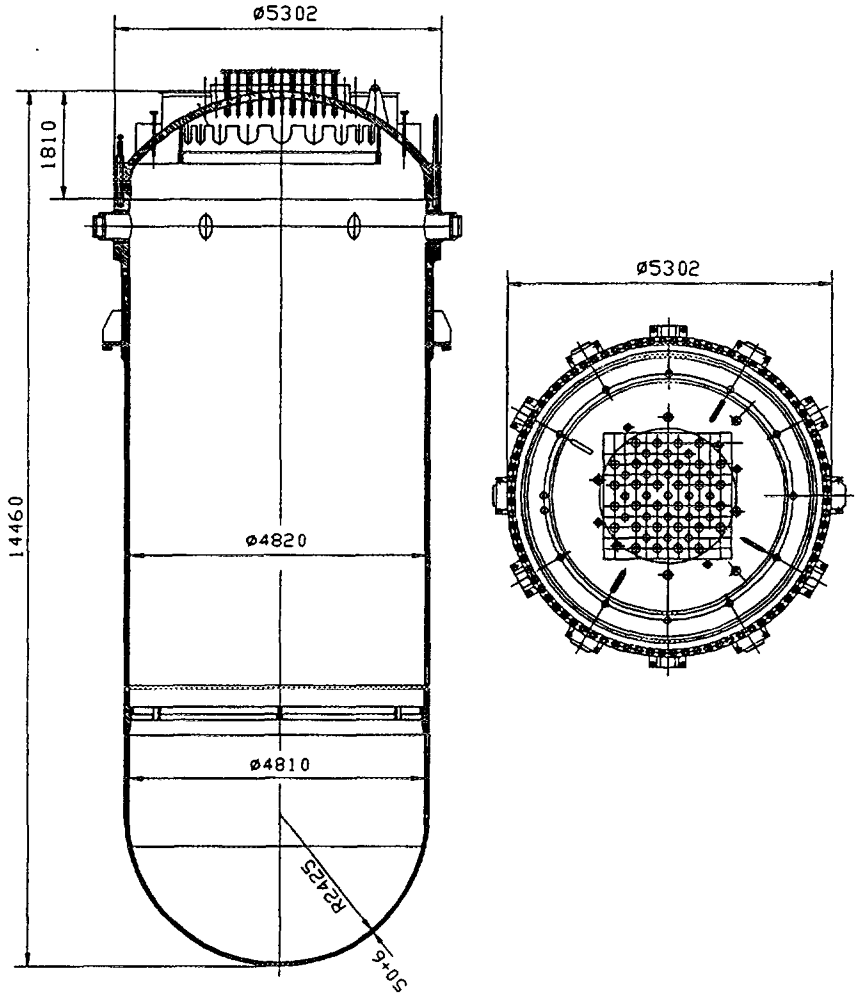


Fig.2 The pressure vessel of the NHR-200

2.3. Metallic containment

The metallic containment is shown in Fig.3. The material of the containment is also SA516-70. The design pressure is 2.3 MPa. Because the penetrations are concentrated in the upper part and the head closure of the pressure vessel, the inner diameter of the upper part of the containment (7,000 mm) is larger than the lower part. The height between the two head closures of the pressure vessel and the containment is no less than 800 mm. The inner diameter of the lower part of the containment is 5,062 mm, and only a narrow gap of 50 mm exists between the containment and the pressure vessel. The upper and lower parts of the containment are connected by a conical cylinder. All penetrations through the containment are arranged in the upper part under the sealing surface of the flange. There are 104 connection bolts with the size of *M804* to connect the containment head closure and the shell. A rubber ring is adopted for flange sealing.

There are 12 support seats for supporting the pressure vessel, and 12 support brackets for the containment itself. All of them are welded inside or outside the conical part of the containment, respectively. Each of them has the capacity to bear vertical and horizontal loads of more than 2 MN.

The overall height and total weight of the containment are 15,476 mm and 2.1×10^5 kg separately.

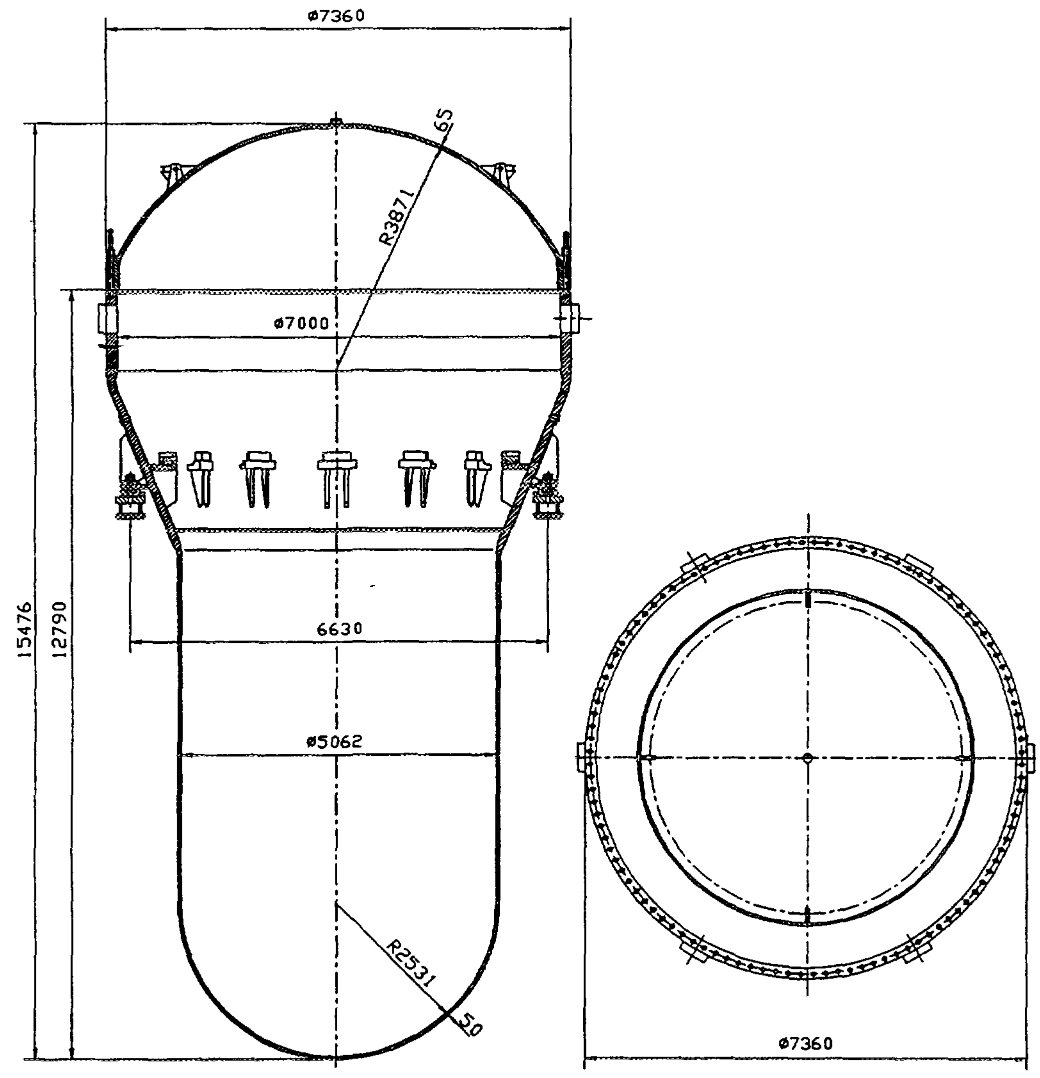


Fig.3 The metallic containment of the NHR-200

3. AIMS, EXAMINATION METHODS AND REQUIREMENTS OF IN-SERVICE INSPECTION OF THE PRIMARY PRESSURE BOUNDARY COMPONENTS

3.1. Aims

The fundamental aim of in-service inspection is to maintain the integrity of the primary pressure boundary, in other words, to avoid the accident of large breaks in the primary boundary components over the full operation time of the reactor, so that the probability of a loss of coolant accident (LOCA) is minimized.

Many aspects, such as material selection, design, fabrication, examination, tests and operating condition control, etc., influence the integrity of the primary pressure boundary. The in-service inspection is also an important aspect.

3.2. Examination methods

Some in-service inspection methods, specified by Codes or Stipulations in some countries, are as follows.

3.2.1. Visual examination

Direct visual examination shall be performed with naked eyes, or where necessary using a magnifying glass or a remote observation device, to observe the surface appearance and leakage vestige of liquid on the inspected component.

3.2.2. Surface examination

A surface flaw can be detected by liquid penetrant examination or magnetic particle examination.

3.2.3. Volume examination

Ultrasonic, radiographic or eddy current examinations can be used to find cracks or other flaws in the inspected component.

3.2.4. Continuously monitoring

The leakage monitoring for liquid or gas is generally chosen for this purpose. There are various monitoring methods, such as pressure, leakage, temperature and radioactivity monitoring, etc., serving to the complements of the in-service inspection methods mentioned above, to ensure and monitor the integrity of primary pressure boundary. The crack, if it appears, will be monitored.

3.2.5. Some other methods, such as hydraulic test, etc.

The hydraulic test will be done not only in the in-service inspection periods, but also before the reactor is restarted after every refueling.

3.3. Requirements

The in-service inspection requirements closely depend on the characteristics of the reactor and the accident consequence of breaks in the primary boundary. They are specified in Codes or Stipulations, and are different depending on the reactor type.

For a pool-type sodium-cooled fast breeder reactor, visual examination is enough because the damage of the primary boundary cannot lead to a serious accident with radioactivity release or loss of coolant.

The primary components of PWRs or BWRs consist of the pressure vessel, main pipes, pumps, pressurizer, penetrations, etc.. A damage of the pressure vessel will result in a serious accident of a core-melt. The break of a main pipe is also serious. Therefore, volume examination (ultra-sonic examination) is required. But for pipes with small diameter, (smaller than 60 mm or 33 mm defined by ASME Vol. XI), the requirement of in-service inspection is visual or surface examination instead of volume examination [1]. The leakage induced by break of these smaller pipes can be compensated by normal or accident water makeup systems, and does not lead to a loss of coolant accident or a core-melt.

4. SAFETY CHARACTERISTICS OF THE NHR-200 PRESSURE VESSEL

Compared with the pressure vessels of PWRs from the safety point of view, the NHR-200 pressure vessel has two features.

4.1. Lower rate of crack propagation

It is known that, usually, the thicker the vessel wall, the larger the flaw size existing in the wall. And the larger the flaw size, the more rapidly the flaw propagates. The wall thickness of the NHR-200 pressure vessel is 65 mm and that of the pressure vessel of a PWR or a BWR is usually larger than 160 mm.

Now, take an example by comparing the crack propagation rate of the NHR-200 pressure vessel with that of the Biblis B reactor vessel [2]. For possible comparison, suppose that cracks with the same size exist in the pressure vessel wall of the two reactors.

The pressure vessel of Biblis B is shown in Fig. 4. Its main parameters are listed as follows:

wall thickness	250 mm (with 6 mm s.s. lining)
inner diameter	5,000 mm
material	22MnMoNi37
operating pressure	15.8 MPa
operating temperature	292.5°C ~ 329.6°C
design pressure	17.5 MPa
design temperature	350°C

According to Appendix A, ASME XI, the fatigue crack propagation rate is:

$$da \,/\, dN = C_o \,(\Delta K_I)^n \qquad (1)$$

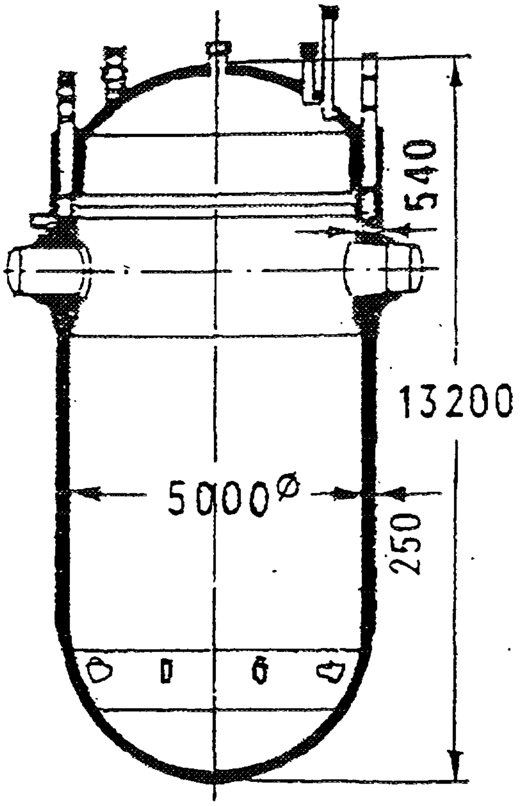


Fig.4 The Biblis B pressure vessel

In equation (1), n and C_0 are determined by the fatigue crack propagation rate curves of Fig. A-4300-1 in reference [3]. The parameter a represents the depth of surface crack, and the crack stress intensity factor K_I is obtained from membrane stress and bending stress in the crack defects by applying the following equation [3]:

$$K_I = \sigma_m M_m \pi^{1/2} (a / Q)^{1/2} + \sigma_b M_b \pi^{1/2} (a / Q)^{1/2} \qquad (2)$$

where:

σ_m: membrane stress, MPa;
σ_b: bending stress, MPa;
a: depth of surface crack, m;
Q: defect shape coefficient from Fig. A-4300-1,
M_m: modification coefficient of membrane stress, determined from Fig. A-3300-3,
M_b: modification coefficient of bending stress, determined from Fig. A-3300-5.

Considering a surface crack with a depth a of 10 mm and length l of 50 mm under three different pressure fluctuations, respectively, from shutdown pressure 0 to normal operation pressure *Po*, from 0.5 *Po* to *Po*, and a fluctuation in a range of ±5% *Po*, the fatigue crack propagation rate *da/dN* of the Biblis B pressure vessel and the NHR-200 pressure vessel are calculated. The results are given in Table 1. *P1* and *P2* are the maximum and the minimum pressures in pressure vessels, respectively.

The calculation results show that for a crack with a depth of 10 mm and a length of 50 mm, under the three different conditions mentioned above, *da/dN* of the Biblis B pressure vessel is, respectively, over 6, 30, and 42 times that of the NHR-200 pressure vessel.

TABLE 1: FATIGUE CRACK PROPAGATION RATE OF THE TWO PRESSURE VESSELS

Status	Reactors	Pressure	σ_m (MPa)	σ_b (MPa)	K_I $(MPa\sqrt{m})$	K_I $(MPa\sqrt{m})$	da/dN (mm)
P1=0 P2=Po	NHR-200	P1=0	0.	0.	0.	16.23	2.36×10^{-4}
		P2=2.4	92.32	1.20	16.23		
	Biblis B	P1=0	0.	0.	0.	28.76	1.49×10^{-3}
		P2=15.8	158.4	7.90	28.76		
P1=0.5Po P2=Po	NHR-200	P1=1.2	46.19	0.60	8.024	8.21	4.08×10^{-6}
		P2=2.4	92.32	1.20	16.23		
	Biblis B	P1=7.9	79.18	3.45	14.13	14.63	1.27×10^{-4}
		P2=15.8	158.4	7.90	28.76		
P1=0.95Po P2=1.05Po	NHR-200	P1=2.28	87.70	1.14	15.36	1.651	2.92×10^{-10}
		P2=2.52	96.94	1.26	17.01		
	Biblis B	P1=15.01	150.4	7.50	27.22	3.106	1.26×10^{-8}
		P2=16.59	166.3	8.29	30.33		

The crack propagation rate of the NHR-200 pressure vessel is also much lower than that at Biblis B under other conditions of operational pressure changes. Thus the safe operation time of the NHR-200 pressure vessel can be much longer.

4.2. The NHR-200 pressure vessel meets 'LBB' condition

In the last decades, with the development of fracture mechanics and the improvement of material properties, the research in the techniques of leak before break (LBB) of pressurized pipelines and pressure vessels, and the application of LBB techniques in component design has gained a lot of development. LBB analysis has been used in the design of pressurized pipelines. For example, the LBB analysis of a main pipe has been complemented to the related regulations of USNRC [4].

It is quite different for a pressure vessel whether it has the characteristic of LBB. If a pressure vessel meets LBB condition, small through-wall cracks may occur and cause a leak of medium, the brittle break accident will be thus prevented. In case of some large, high energy pressure vessels, LBB condition can virtually improve their safety, and it is more attractive. So the LBB characteristic has already become one of the aims pursued by designers and researchers in the field of pressure vessels.

The reasons that the NHR-200 pressure vessel meets LBB condition comes mainly from the following features of the pressure vessel.

(1) Low operating pressure and small wall thickness

The operating and design pressure of the NHR-200 pressure vessel are 2.5 and 3.1 MPa, respectively. In case of a PWR, the values are no less than 15 and 16 MPa. As a result, the wall thickness of the NHR-200 pressure vessel is only a quarter of that of a PWR. A small wall thickness is very beneficial for meeting LBB condition.

(2) Small integral flux of fast neutron suffered by the pressure vessel

The high integral flux of fast neutron (10^{19} n/cm^2) suffered by pressure vessels of PWRs makes the non-ductile transition temperature of the material rising clearly, and the toughness decreasing greatly. For the NHR-200, the pressure vessel experiences no more than 10^{16} n/cm^2 after 40 year's operation [5]. So the brittle break of material by neutron irradiation can be ignored. The material will retain its good toughness in all its life time. This is also significant.

(3) Low stress level

Because the stress level in the pressure vessel is relatively low, the material of SA516-70 with low strength and high toughness is selected for the NHR-200 pressure vessel [5]. The maximum membrane stress occurs in the cylinder part. It is only 92.5 MPa under operating pressure and 115 MPa under design pressure. But for a PWR, the two values are at least 150 and 180 MPa.

(4) Great heat capacity and thermal inertia

The operating pressure and temperature change very slowly due to the great heat capacity and thermal inertia.

(5) Large margin of material toughness

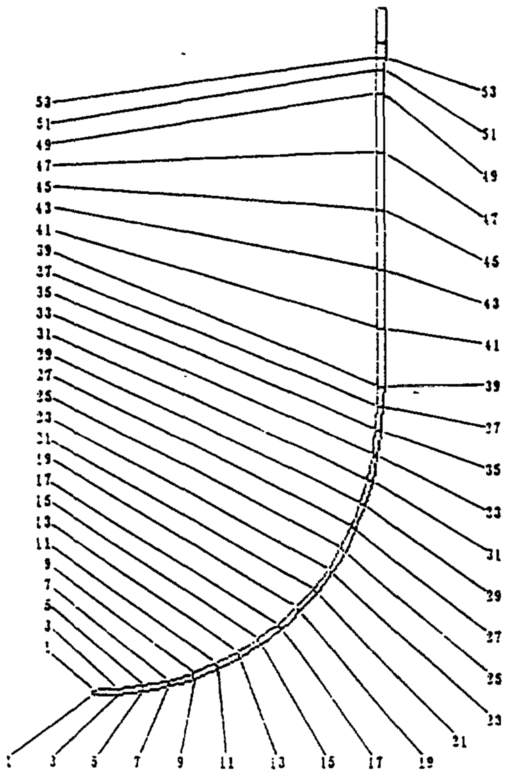
It is proven that the NHR-200 pressure vessel meets LBB condition by analyses with different methods. When a crack with a length of 0.65m (over 10 times of the wall thickness) occurs in the cylinder part under the highest operating pressure, the stress intensity factor is still under the conservative limit of critical crack propagation [6]. The critical length of a crack on the other parts of the pressure vessel is even larger.

The stress distribution curves of the NHR-200 pressure vessel are shown in Fig. 5.

The weak point of the NHR-200 pressure vessel is the cylinder part where the stress is higher than in other parts. It can easily be proven that the cylinder part meets LBB condition. The Mode I cracks in the cylinder part are discussed in the following.

When a small through-wall crack just occurs in the vessel, the leak rate is too small to be judged correctly.

USNRC stipulates the principles and the methods of LBB analysis for the main pipes of a nuclear reactor on the basis of enormous theoretical and experimental research [4]. The analysis method is established on the principles of minimum detectable leak rate. According to reference [7], the leak rate of the minimum breach inspected by the monitoring systems in an hour after a leak is no more than 1 G/min. In accordance with reference [4], for LBB



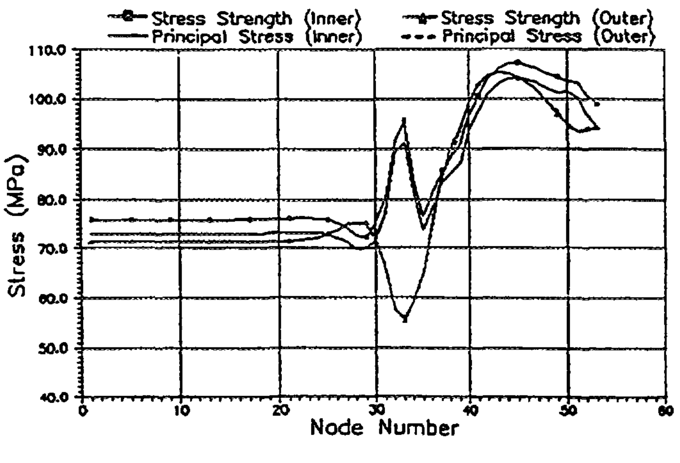


Fig.5.1 The stress distribution of the bottom closure

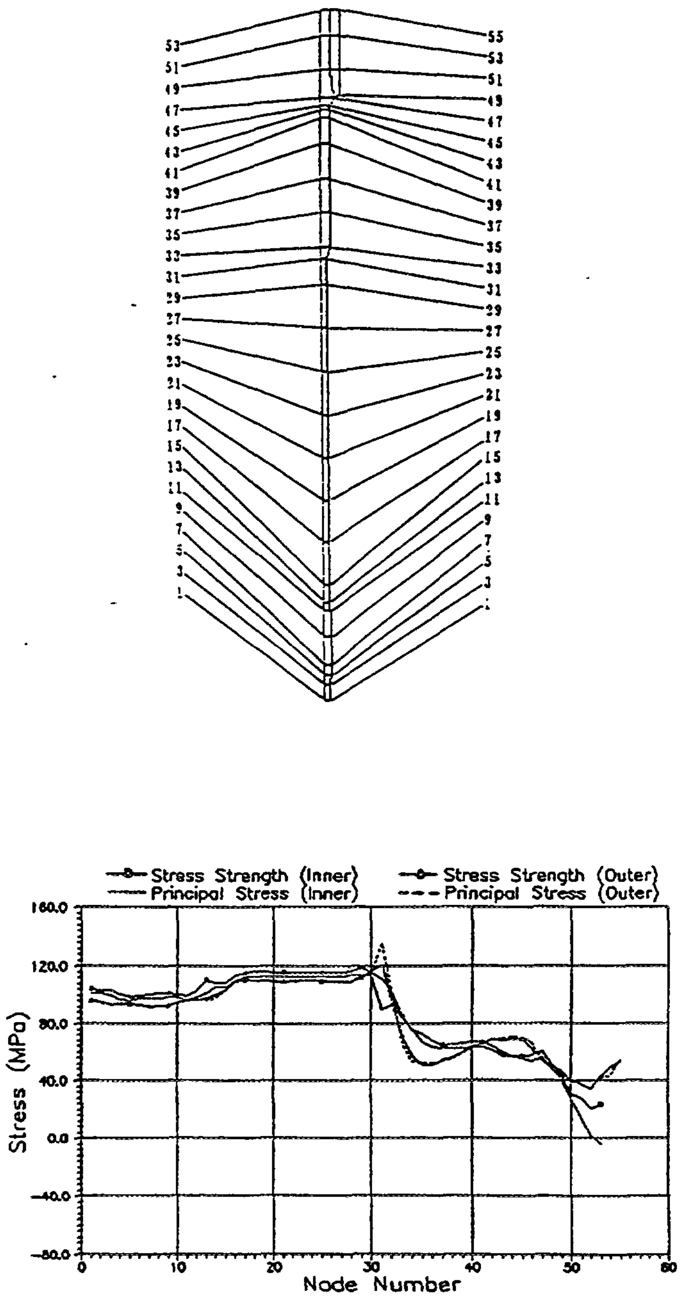
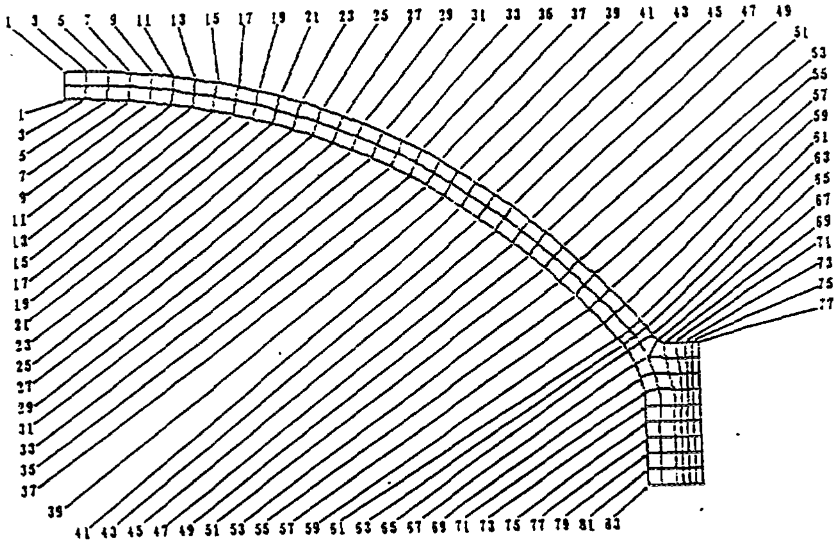


Fig.5.2 The stress distribution in the cylinder



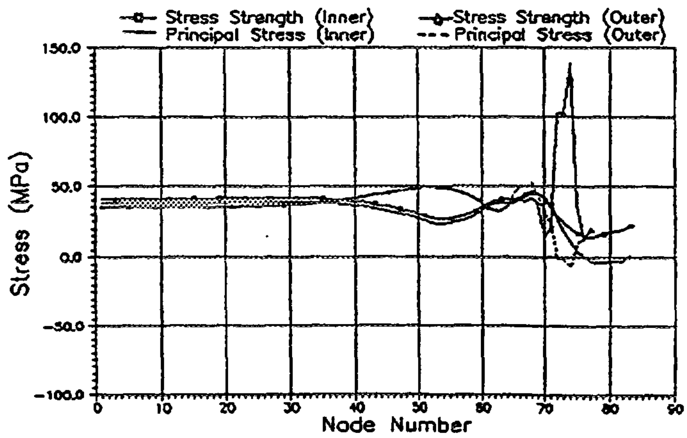


Fig.5.3 The stress distribution of the head closure

analysis, take the detectable leak rate as 10 times of the minimum detectable leak rate of the monitoring system. Then take a safety factor of 2 for break length. That is, when the break leak rate reaches 10 times of the minimum detectable leak rate, the LBB analysis is based on the condition that the break length at this moment is no larger than half of the critical length of unstable break propagation. This principle is obviously very conservative.

4.2.1. The critical length of unstable through-wall crack propagation for the cylinder part of the pressure vessel

In the following analysis, the methods of linear elastic fracture mechanics and Double Criteria are used to determine the critical length of unstable through-wall crack propagation. The toughness of SA516-70 is very good and the analysis methods used here are conservative.

4.2.1.1 The crack stress intensity factor K_I

The stress intensity factor of longitudinal through-wall crack of cylinders is:

$$K_I = \sigma\sqrt{\pi a M} \qquad (3)$$

where σ is the annular stress on the cylinder part, a is the semi-length of the crack. M will be obtained as follows:

$$M = M(a) = [1+\frac{5\pi}{32}\lambda^2(a)]^{1/2} \qquad (4)$$

$$\lambda^2(a) = [12(1-\upsilon^2]^{1/2}\frac{a^2}{Rt} \qquad (5)$$

where v is Poisson ratio, R is the radius of the cylinder part.

4.2.1.2. Dugdale Model [8]

According to the Dugdale Model in elastic-plastic fracture mechanics, consider a Mode I crack with the length $2a$ in an infinite plate. Near the crack tip exists an ideal plastic zone with the length r, as shown in Fig. 6, r can be obtained from:

$$r = a[\sec(\frac{\pi\sigma}{2\sigma_s})-1] \qquad (6)$$

where the yield strength σ_s is the compression stress in the plastic zone in the vertical direction to the crack face.

Take the crack semi-length $a+r$. The outside compression stress with a value of σ_s is exerted on the upper and lower faces of the crack in the plastic zone. Then at the actual crack tip there exists an opening displacement δ_i.

4.2.1.3. Double Criteria Method [8]

In an infinite plate, the critical crack semi-length can be obtained from:

$$\frac{S_r^2}{K_r^2} = \frac{8}{\pi^2} ln[\sec(\frac{\pi}{2} S_r)] \tag{7}$$

where,

$$S_r = \frac{\sigma}{\sigma_s}, \quad K_r = \frac{K_I}{K_{IC}} = \frac{\sigma}{K_{IC}} \sqrt{\pi a} \tag{8}$$

The critical semi-length of unstable crack propagation can be worked out from K_I.

For the NHR-200 pressure vessel, because of its very thin wall, large diameter and small bulging-effect, equation (7) can still be regarded as an approximate formula of Double Criteria.

4. 2.1.4. The calculation results of critical length of unstable crack propagation

The condition of unstable crack propagation is

$$K_I \geq K_{IC} \tag{9}$$

where K_{IC} is the fracture toughness. K_{IC} of SA516-70 is shown in Fig. G-2210-1 of App. G, ASME XI. The minimum operating temperature of the NHR-200 is no less than 50°C. From Fig. G-2210-1, taking the RT_{NDT} = 0°C for SA516-70, and a safety factor of 0.9 for K_{IC} , then the value of K_{IC} is no less than 180 MP$\sqrt{m}$.

Under the design pressure, the annular stress of the cylinder part is 115.4 Mpa. Under operation pressure, the stress is 92.3 MPa.

According to linear elastic methods, a crack will propagate unstably when K_I reaches K_{IC}. In this case, we can work out the critical crack semi-length a under the design pressure and the operating pressure using equation (3).

According to Double Criteria, Sr, Kr and K_I can be obtained by using equation (7) and (8), and the critical crack semi-length a can be obtained by using equation (3).

The calculation results are given in Table 2.

TABLE 2: THE CRITICAL SEMI-LENGTH OF CRACK UNSTABLE PROPOGATION/(mm)

Methods	Design condition	Operation condition
Linear elastic	349.	429.
Double Criteria	334.	419.

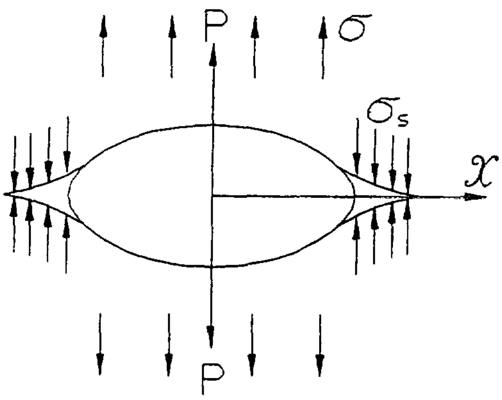


Fig.6 The Dugdale model

4.2.2. Crack Opening Area (COA)

With linear elastic theory and the Dugdale Model, the crack central opening can be derived. The shape of the crack opening is conservatively considered as rhombic. For the crack semi-length a and the crack central opening u, the COA is:

$$S = ua \tag{10}$$

4.2.2.1. Crack Central Opening (CCO) [9]

4.2.2.1.1. Method of linear elastic fracture mechanics

The stress intensity factor of a crack with semi-length x (resulting from stress) is (Fig. 6):

$$K_I = K_I(x) = \sigma\sqrt{\pi x}M(x) \tag{11}$$

where $M(x)$ is given in equation (4).

A pair of postulated force P act vertically on the center points of the upper and lower surfaces of the crack depicted in Fig.6. For the crack with a semi-length x, the stress intensity factor is:

$$K_{IP} = K_{IP}(x) = P\sqrt{\frac{1}{\pi x}}M(x) \tag{12}$$

The CCO can be obtained as follows:

$$u = \frac{4}{E}\int_o^a K_I \frac{K_{IP}}{P}dx = \frac{4\sigma a}{E} + \frac{2.16}{E}\frac{\sigma a^3}{Rt} \tag{13}$$

where the first term on the right side of the equation is CCO in an infinite plate, the second term is the bulging-effect, and it is very small.

4.2.2.1.2. Dugdale model

According to Dugdale elastic-plastic model, the CCO is:

$$u_{\sigma} = \frac{4}{E}\int_{o}^{a+r} K_I \frac{K_{IP}}{P} dx \tag{14}$$

The crack stress intensity factor resulted by σ_s mentioned in 4.2.1.2 is:

$$K_I^{'} = K_I^{'}(x) = -2\sigma_s\sqrt{\frac{x}{\pi}}\cos^{-1}(\frac{a}{x}) \tag{15}$$

The CCO is:

$$u_{\sigma s} = \frac{4}{E}\int_{a}^{a+r} K_I^{'} \frac{K_{IP}}{P} dx \tag{16}$$

Put equation (6), (11) and (12) into (14), and put equation (6) and (12) into (16), the CCO can be obtained:

$$\begin{aligned} u &= u_{\sigma} + u_{\sigma s} \\ &= u_o + \frac{2.16\sigma_s a^3}{\pi E R t}[\sec\frac{\pi\sigma}{2\sigma_s} tg\frac{\pi\sigma}{2\sigma_s} + \ln(\sec\frac{\pi\sigma}{2\sigma_s} tg\frac{\pi\sigma}{2\sigma_s})] \end{aligned} \tag{17}$$

where

$$u_o = \frac{8\sigma_s a}{\pi E}\ln[\sec\frac{\pi\sigma}{2\sigma_s} tg\frac{\pi\sigma}{2\sigma_s}] \tag{18}$$

The u_0 is the CCO in an infinite plate. The second term on the right side of equation (17) reflects the bulging-effect.

The semi-length a of a through-wall crack and the opening u of the crack center obtained from equation (13), (17) are listed in Table 3.

4.2.2.2. Calculation of COA

Depicted in Table 3 is the COA which is calculated by applying equation (10).

TABLE 3. CRACK SEMI-LENGTH a, CCO u AND COA S

a(mm)		100		200		300	
u(mm)	S(mm^2)	u	S	u	S	u	S
operating condition	Linear elastic	0.182	18.2	0.398	79.6	0.685	205
	Double Criteria	0.192	19.2	0.425	85.0	0.738	221
design condition	Linear elastic	0.227	22.7	0.498	99.6	0.857	257
	Double Criteria	0.249	24.9	0.551	111	0.973	292

The actual shape of the break is not rhombic. It is elliptical in accordance with linear elastic theory. Comparing with the actual break areas, the results obtained from equation (10) are relatively conservative.

4.2.2.3. Detectable leak rate

The volume of the NHR-200 metal containment is very small. A leakage from the pressure boundary can be easily detected by continuously monitoring the pressure and the temperature in the containment.

The normal operating pressure of the NHR-200 containment is only 0.1 MPa. When the pressure goes up to 0.15 MPa, a warning signal will be given.

If a break happens in the pressure boundary, water in the pressure vessel will leak into the containment and cause the pressure in the containment to rise from 0.1 MPa to 0.15 MPa in an hour after the leak. Then the leak rate will be less than 0.2 G/min. Taking a safety factor of 10, the detectable leak rate is less than 2 G/min.

4.2.2.4. The COA and the leak rate

The CCO is taken as the equivalent size. When the equivalent size is less than one twelfth of the vessel thickness, the leak rate can be approximately calculated by means of the extended Henry-Fauske/Moody critical flow model [10] given in reference [11], Fig. 4.

Considering the roughness of break surface, the actual break leakage rate is smaller than that obtained from Fig. 4. Reference [12] shows that the fraction factor of the crack surface is between 0.01 and 0.08.

According to the actual minimum detectable leak rate of 0.2 G/min and the stipulated minimum detectable leak rate in reference [1], 1 G/min, taking the safety factor of 10, the detectable leak rate 2 G/min and 10 G/min are used in the LBB analysis of the NHR-200 pressure vessel. According to the Henry-Fauske/Moody model, the leak rates corresponding to different crack length mentioned above, and in case of smooth crack surface (friction factor f=0) and rough crack surface (f=0.08), are calculated by using RETRAN-02 code.

The results are given in Table 4. It can be found that when friction factor is 0.08 under operating conditions, the crack semi-length corresponding to the detectable leak rate of 2 G/min (0.1261 Kg/s) and 10 G/min (0.631 Kg/s) are about 55 mm and 136 mm. Under design conditions, the crack semi-length corresponding to leak rate 2 G/min (0.1261 Kg/s) and 10 G/min (0.631 Kg/s) are about 50 mm and 120 mm.

TABLE 4: THE CRACK SEMI-LENGTH A AND THE LEAK RATE

crack semi-length (mm)	operating condition		design condition	
	leak rate (Kg/s)		leak rate (Kg/s)	
	f=0	f=0.08	f=0	f=0.08
50	0.1098	0.0944	0.1694	0.1132
100	0.5832	0.3419	0.8846	0.4068
150	1.8030	0.7488	2.6992	0.9882
200	4.0254	1.4579	5.2050	1.9543

4.2.3. The LBB analysis results

It can be shown, by comparing Table 2 with the analysis results of 4.2.2.4, that the length of a through-wall crack corresponding to the detectable leak rate is far less than half of the critical length of unstable crack propagation. The NHR-200 pressure vessel satisfies the LBB condition.

5. IN-SERVICE INSPECTION REQUIREMENTS FOR THE NHR-200 PRESSURE VESSEL

5.1. Summary of the safety characteristics of NHR-200 pressure vessel

(1) The margin of the toughness of the pressure vessel material is big enough for the full operating time. The low rate of fatigue crack propagation of the NHR-200 pressure vessel makes also a contribution to the safe operation time. On the basis of the operation experience with PWRs and BWRs, and the analysis results for the NHR-200 pressure vessel, the probability of a break accident for the NHR-200 pressure vessel is considered to be very low (10^{-7} per reactor year).

(2) The NHR-200 pressure vessel meets LBB condition.

Even if a break occurs, it should be small in size. Large brittle break accidents for the vessel can be excluded.

(3) The break accident of a light water reactor pressure vessel is not considered. In fact, this kind of accident is unacceptable for light water reactor pressure vessels up to now. For the NHR-200, this beyond design basic accident is acceptable.

5.2. In-service inspection requirements

Based on the reasons mentioned above, the in-service inspection requirements for the NHR-200 pressure vessel are determined as follows.

5.2.1. The in-service inspection period is defined from Unit 1 in ASME XI

5.2.2. In-service inspection for the pressure vessel and its penetrations

Visual inspection, continuously monitoring and hydraulic tests are selected for inspections of the vessel. For main studs, nuts and other related parts of the pressure vessel the in-service inspection will be performed in accordance with ASME XI [1].

5.3. Explanation on the in-service inspection

5.3.1. Hydraulic test and its functions

The hydraulic test for the pressure vessel is an important measure of in-service inspection. After every refueling, the test will be performed.

The test temperature is far below the operating one, the test pressure exceeds that of operation. An appropriate temperature To is chosen, such as $To=50°C+RTNDT$. If no break occurs in the test condition, the pressure vessel still has enough safe operating lifetime. So, as a kind of effective examination methods for a pressure vessel, the hydraulic test is still important [13].

5.3.2. Monitoring system for primary pressure boundary

There are water level monitoring systems for the pressure vessel, and temperature, pressure, humidity and radioactivity monitor systems for the containment. These systems can detect if a break occurred in the primary pressure boundary and give tone or light signals. If a break developes in the pressure boundary, the reactor can be shut down.

As mentioned above, the containment is of small volume. So the leak monitor system for the primary pressure boundary will perform with high sensitivity.

5.3.3. Material radiation surveillance capsule

Even though the flux of fast neutron suffered by the pressure vessel is considerably low, the radiation surveillance capsule is still adopted for mechanical property tests of the pressure vessel material.

6. EVALUATION OF IN-SERVICE INSPECTION OF THE NHR-200 PRESSURE VESSEL

6.1. Comparisons with other type reactors

The NHR-200 pressure vessel is somewhat similar to that of sodium-cooled fast reactors. Breaks of the primary boundary do not lead to a core-melt accident. Large breaks will not occur in the vessel, and small breaks do not result in a serious accidental radioactivity release. Therefore, volume examination is not required for the in-service inspection of sodium-cooled fast reactors

The volume examination is required for PWRs and BWRs. It will reduce the probability of a break. The result of a break accident for a PWR or BWR pressure vessel is so serious that there is no way to deal with. This kind of accident is considered as incredible. But the risk is still there.

For the NHR-200, the in-service inspection measures, except volume examination, are as same as that for PWR or BWR. The break probability of the pressure vessel is certainly very low. And the result of a break is considered in accident analysis and the consequence is acceptable.

6.2. Safety characteristics of NHR-200

The capacity of keeping integrity of the primary pressure boundary for the NHR-200 is good enough as mantioned above. As compared to a PWR or BWR, the material toughness of the NHR-200 pressure vessel has more margin, and the sensitivity of the leak monitoring system is higher. Unstable propagation of cracks will not occur, and loss of coolant accidents and radioactivity releases can be avoided.

Because a compact metallic containment is adopted, the accident analysis have proven that, if any believable break accident occurs on the primary pressure boundary, including the maximum penetration break of the pressure vessel, a break on the bottom of pressure vessel and the safety valve stuck open, the residual water level will be still above the active core.

The reasons are as follows:

(1) The water leakage is limited by the small volume of the containment.

(2) The probability of two vessel breaking simultaneously is very small, and can be ignored.

The metallic containment serves as a guard vessel for the pressure vessel. Only in case that the two vessels would break simultaneously, a core-melt accident could happen. However, the probability of this accident is too small to be considered [14].

(3) Because pressure, temperature, and gas radioactivity monitor systems are installed in the space between the pressure vessel and the containment, a very small break on the pressure vessel could be detected in a short period, and counter-measures can be taken to depressurize the pressure vessel in order to terminate the crack propagation.

(4) The NHR-200 has a decay heat removal system with reliable and good inherent safety.

6.3. Inservice inspection, integrity of primary pressure boundary and core-melt accident

The integrity of the reactor primary pressure boundary depends on many measures including design, material, manufacturing, examination, test, in-service inspection and so on. Because of the limitations of space, time, technique or other unforeseen factors, the in-service inspection can not be performed perfectly. For instance, the volume inspection is not performed on the whole of parent material, but only on the welding zone. In fact, some weld seams in the primary pressure boundary of PWRs are not accessible for inservice inspection [15][16]. Historically, some break accidents occurred on the connection between main pipe and pressure vessel of BWRs. That is to say, the accident probability can only be decreased to a certain extent by in-service inspection, but can not be excluded.

It has been the effort of the authors that in designing the NHR-200, the integrity of the primary pressure boundary is made to rely to a much lesser extent on inservice inspection, especially on volume inspection, and that the integrity of the fuel elements and other core structures are not vulnerable to small breaks in the primary pressure boundary.

6.4. Brief summary

Low failure probability of the reactor pressure vessel, acceptability of a small break in the wall of the reactor pressure vessel, only small LOCAs, and no core-melt accidents are the safety features of the NHR-200 and its pressure vessel. Based on these features the inservice inspection requirements are drafted for the NHR-200 pressure vessel.

Although the pressure vessels of PWRs or BWRs are inspected by means of volume examination and other in-service inspection measures, their operational safety can't achieve the safety level of the NHR-200. We can say that the in-service inspection requirements of the NHR-200 pressure vessel can meet the basic safety requirements.

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WATER CHEMISTRY AND BEHAVIOUR OF MATERIALS IN PWRs AND BWRs

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Abstract

Water chemistry plays a major role in corrosion and in activity transport in NPP's. Although a full understanding of all mechanisms involved in corrosion does not exist, controlling of the water chemistry has achieved good results in recent years. Water chemistry impacts upon the operational safety of LWR's in two main ways: integrity of pressure boundary materials and, activity transport and out-of-core radiation fields.

This paper will describe application of water chemistry control in operating reactors to prevent corrosion. Some problems experienced in LWR's will be reviewed for the design of the nuclear heating reactors (NHR).

1. INTRODUCTION

The IAEA co-ordinated research program entitled "Investigation on Water Chemistry Control and Coolant Interaction with Fuel and Primary Circuit Materials in Water Cooled Power Reactors (WACOLIN)" was organized and carried out from 1987 to 1991. The reports of this work summarize the present understanding on good coolant chemistry (IAEA Technical Reports Series No. 347 and IAEA-TECDOC-667):

"Good reactor coolant chemistry, corrosion control and minimum of activity build-up are indispensable for the optimum performance of nuclear power plants. Without these the system integrity may be jeopardized and the activity transport may create various problems".

For a nuclear power plant the capability to operate with an optimal chemistry regime is determined by the design including materials, construction and effectiveness of the water purification system. The choice of materials defines water quality requirements and dimensions of water treatment systems. Some features of design like the use of copper-base alloys or Inconels or carbon steels presents limitations for the optimum water chemistry. Therefore recommendations for water chemistry must be established together with the design and material specifications.

Reliable water chemistry specifications have been developed for the existing water cooled reactors taking into account the common material practice for high temperature operation (T > 250 °C); however, there is still room for improvement. Water cooled power reactor experience shows that even under normal operating conditions some undesirable phenomena can occur like stress corrosion cracking and corrosion fatigue, erosion corrosion or deposition of corrosion products on heat transfer surfaces.

2 PWR COOLANT SPECIFICATIONS

The goal of recommendations for chemistry in operating PWR's is to limit radioactive transport and thereby reduce out-of-reactor radiation fields. Radiolytic oxygen formation in PWR's is suppressed by maintaining an overpressure of hydrogen. Generally 25 - 30 cc/kg are required to keep oxygen below 5 ppb. Control of pH in PWR primary systems is complicated by the use of boric acid as a chemical shim to control nuclear reactivity, which results in a need to adjust lithium or potassium hydroxide content to avoid a continually changing pH through the fuel cycle.

Typical water chemistry specifications used in PWR's are shown in Tables 1 and 2 as well as in Fig. 1. The three options being consistent with the guidelines are:
- Elevated Li-B-Chemistry
- Modified Li-B-Chemistry
- Co-ordinate Li-B-Chemistry.

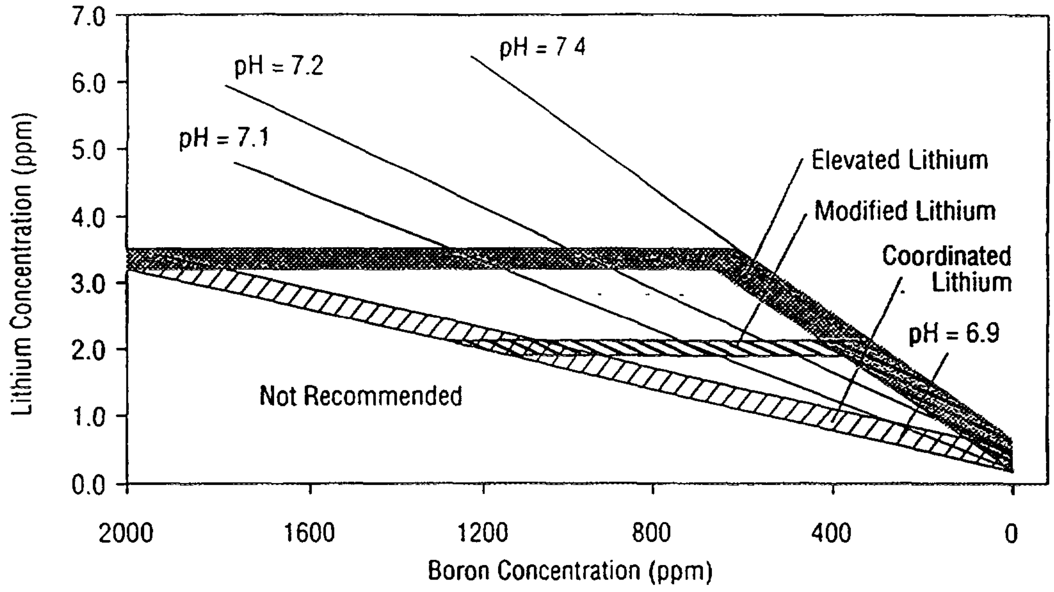

Fig. 1. PWR primary system chemistry control regimes (EPRI NP-7077 1990).

The operational experience of the last years has shown that the modified Li-B-Chemistry should be preferred over the other two options. Corrosion product transport and solubility has been lowest and the estimated relative activity build-up on cold regions of the primary circuit, as a result of precipitation and activation on hot regions and subsequent dissolution and redistribution throughout the system, is minimized (Riess 1993).

PWR primary water chemistry changes greatly at shut-down, when the coolant is oxygenated and borated during cooling period. This change will release activated corrosion products which can be collected by clean-up systems operating at full power.

The guidelines give a possibility to great variation especially concerning pH optimization. This optimization must be done plant-specific. Generic principles for optimization of primary system pH according to EPRI Guidelines (EPRI NP-7077 1990) are presented in order of priority as follows:

1. Operate at or above $pH_{t(ave)}$ = 6.9 (t_{ave} = average of t_{inlet} and t_{outlet}) to minimize crud deposition on fuel and enhanced Zircaloy oxidation. If such operation requires greater than 2.2 ppm lithium, the impact on heat exchanger materials should be assessed. If plant-specific considerations require operation below $pH_{t(ave)}$ 6.9, a fuel surveillance program should be considered.

2. For operation above 2.2 ppm lithium for the purpose of achieving pH > 6.9, plant-specific fuel and materials reviews should be performed, and a fuel surveillance program considered.

3. Once lithium has been reduced to 2.2 ± 0.15 ppm consistent with principles #1 and #2, lithium can either be controlled so as to continue operation at $pH_{t(ave)}$ = 6.9, or maintained constant at 2.2 ± 0.15 ppm until a specified $pH_{t(ave)}$ between 6.9 and 7.4 has been reached. The plant-specified pH should be selected on the basis-specific impacts on fuel and materials integrity and radiation field control. There are limited data available on which to base operation above $pH_{t(ave)}$ = 7.4. Therefore, it is recommended that $pH_{t(ave)}$ = 7.4 (±0.15 ppm lithium) be considered the upper operating band.

4. Maintain the specific pH at ±0.15 ppm lithium until the end of the operating cycle, noting that lithium variations have greater effect on pH at lower boron concentrations.

5. Attempt to minimize pH fluctuations during power operation. However, during power-level changes, some fluctuation in pH changes may be unavoidable.

These guidelines with specified pH values are applicable only for reactors operating at high temperatures and, thus, for nuclear heating reactors (NHR's) the specifications should be reviewed with respect to the fuel cladding failures, stress corrosion and corrosion product release phenomena shown in Fig. 2. The minimum solubility pH of magnetite at temperatures below 200 °C is pH_T > 7.5. This can be obtained by having the total Li-content < 2.2 ppm depending on the desired boric acid content needed as a moderator. In Table 3 some examples of the amount of Li needed to obtain the coolant pH providing the minimum solubility of magnetite at different temperatures and boron contents are presented. However, the potential risk of material failures caused by using higher concentrations of Li (i.e., 2.2 ppm) than specified in the guidelines for reactors operating at high temperatures should be evaluated for temperatures to be used in NHR's.

With respect to the NHR's the choice of the BWR water chemistry raises questions concerning general corrosion and following activity levels and SCC problems if the oxygen content is not reduced, e.g., by using hydrogen addition. Anyway, BWR water chemistry has to be optimized also before NHR application as shown in Fig. 3.

Table 1. EPRI guidelines for primary coolant in PWR's (EPRI NP-7077 1990).

Hydrogen (cm^3) (STP)/kg H_2O [a)]	25 - 50
chlorides (mg/kg)	< 0.15
fluorides (mg/kg)	< 0.15
dissolved oxygen (mg/kg)	< 0.01
lithium (mg/kg)	consistent with station lithium program

[a)] STP, standard temperature and pressure (0 °C, atm)

Table 2. Specifications of reactor water quality for PWR's of VVER-440 and VVER-1000 type (Rieβ 1993).

Indicator	Values	
	VVER-440	VVER-1000
pH (25°C)	6.0 - 10.2	5.7 - 10.2
K^+, L^+, N^+ (mmol/kg) (depending on H_3BO_3 concentration)	0.05 - 0.45	0.05 - 0.45
NH_3 (mg/kg)	> 5.0	> 5.0
hydrogen (cm^3/kg)	30 - 60	30 - 60
chlorides and fluorides (μg/kg)	≤ 100	≤ 100
H_3BO_3 (g/kg)	0 - 9.0	0 - 13.5
oxygen (μg/kg)	≤ 5	≤ 5
copper (ng/kg)	< 20	< 20
iron (ng/kg)	< 200	< 200

Table 3. Required Li concentrations in the coolant with constant boron concentration (1000 ppm as B) needed to obtain the pH of minimum solubility for magnetite at different temperatures.

Li / ppm	B / ppm	pH_T	pH_T for solubility minimum
3.8	1000	6.57 (150°C)	8.0
100	1000	7.99 (150°C)	8.0
3.8	1000	6.65 (200°C)	7.5
30	1000	7.51 (200°C)	7.5
3.8	1000	7.21 (300°C)	7.2

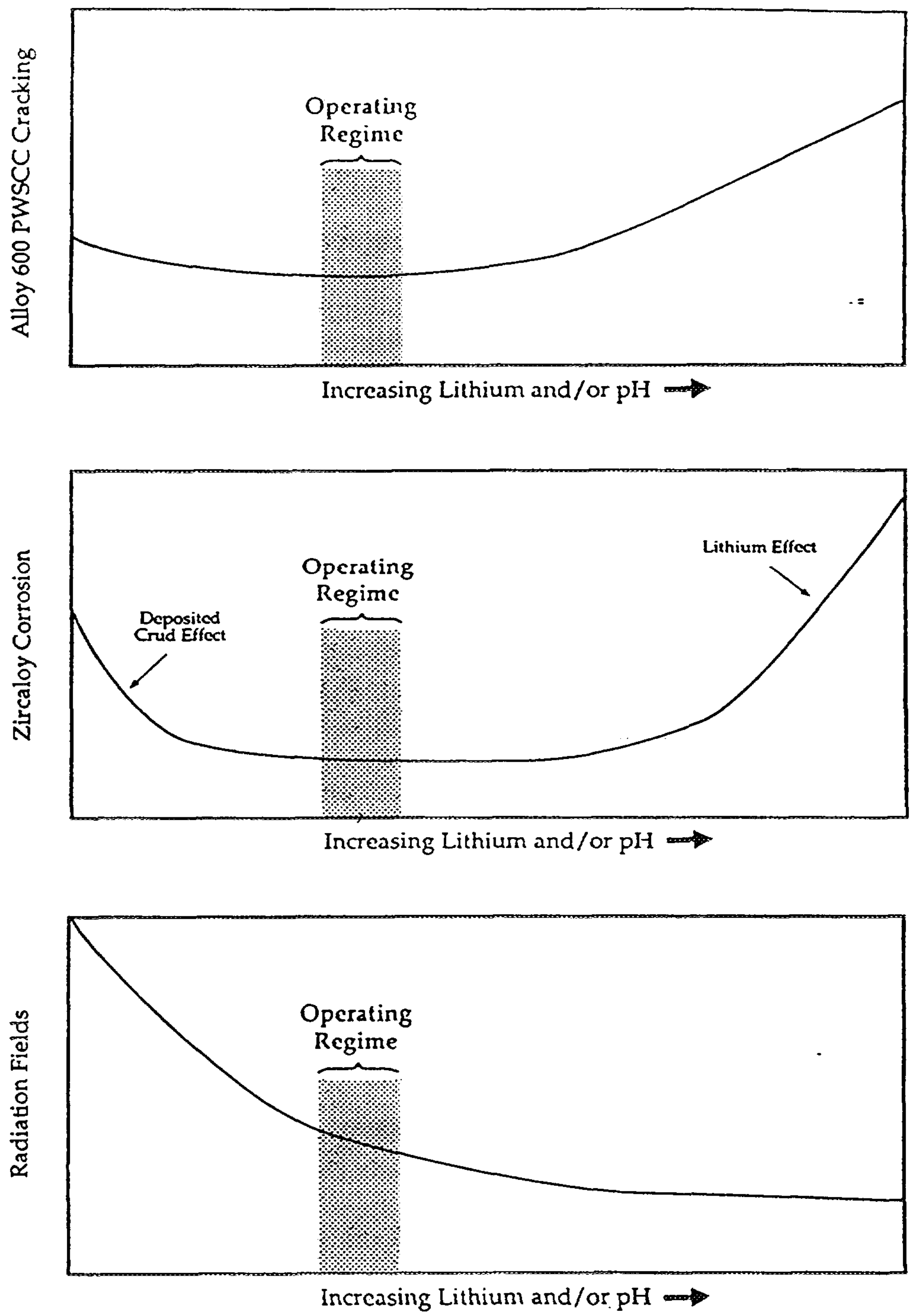

Fig. 2. Schematic presentation of the PWR primary chemistry optimization problem (EPRI NP-7077 1990).

Table 4a. EPRI water chemistry guidelines for BWR's (EPRI NP-4946-SR 1988).

Control parameter	Frequency of measurement	Achievable value	Action levels 1	2	3
Reactor water during power operation					
Conductivity (μS/cm at 25 °C)	continuously	≤0.20	>0.30	>1.0	>5.0
Chloride (ppb)	daily	≤15	>20	>100	>200
Sulphate (ppb)	daily	≤15	>20	>100	>200
Diagnostic parameter, silica (ppb)	daily	≤100			

Table 4b. EPRI water chemistry guidelines for BWR's (cont'd) (EPRI NP-4946-SR 1988).

Control parameter	Frequency of measurement	Achievable value	Action levels 1	2	3
Reactor feedwater / condensate during power operation					
Feedwater conductivity (μS/cm at 25 °C)	continuously	≤0.06	>0.07		
Condensate conductivity	continuously	≤0.08	>0.10		>10
Feedwater total copper (ppb)	weekly integrated	≤0.10 ≤0.30	>0.50 >0.50		
Feedwater total iron (ppb)	weekly integrated	≤2.0	>5.0		
Feedwater dissolved oxygen (ppb)	continuously	20-50	<10 >200		

2.1 BWR COOLANT

BWR's operate under constant oxygen chemistry throughout the fuel cycle and so there is no changes taking place like in PWR's during shut-down affecting the transport of activated corrosion products. Thus, the deposition rate of activity is mainly controlled by the corrosion rate during steady state operation. The aim of the BWR water chemistry is to control the corrosion product input in the feedwater, to reduce crud build-up and to control and minimize radiation field build-up on recirculation piping. Both the above aspects of BWR radiation control require that feedwater and reactor water are kept as pure as possible (EPRI NP-4946-SR 1988).

BWR water chemistry specifications according to the EPRI guidelines are listed in Table 4. These and corresponding guidelines are now worldwide under review. The existing operational problems, specially the IGSCC problem, require more stringent values. For example, it is currently discussed to fix the chlorides and sulphate values of the reactor water for action level 1 at 5 ppb. Also, the iron level of the feedwater seems to be too high. BWR water chemistry has also been modified in order to moderate the SCC problems. Hydrogen water chemistry (HWC) has been used to control intergranular stress corrosion cracking. However, applied HWC has resulted in two radiation effects. The first effect is the increase in N-16 radiation. The second is the Co-60 shut-down radiation field mainly caused by a similar change in the redox environment like in PWR chemistry. Zinc injection has been studied in consideration for reducing cobalt-60 fields. Noble-metal coatings (Pt, Pd) or noble metals as an alloying element have been demonstrated to improve the effects of hydrogen water chemistry. Noble metals even at low concentrations enhance the cathodic reactions and, thus, decrease of corrosion potential takes place faster when hydrogen is induced.

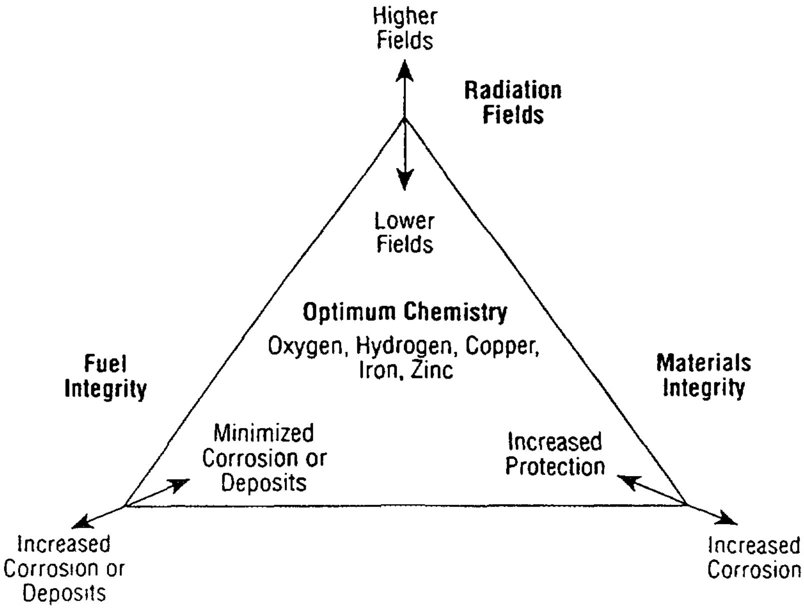

Fig. 3. Problems with optimizing BWR chemistry (EPRI TR-100265 1992).

3 ENVIRONMENTALLY ASSISTED CRACKING (EAC) IN LIGHT WATER REACTORS (LWR'S)

Environmentally assisted cracking of LWR pressure boundary components has caused significant outages with occasional safety hazards. Most of the components of nuclear power plants have been affected by corrosion damage. Environment sensitive cracking incidents in pressure vessels, piping and heat exchanger have led to replacements of these major components often after only a small fraction of their design life. Numerous reviews of environment sensitive cracking in light water reactor components have been presented (e.g. Stahlkopf 1982, Berry 1984, Scott 1985, Norring & Rosborg 1984, Hänninen & Aho-Mantila 1985, Hänninen & Aho-Mantila 1986).

The complex interplay of metallurgical, mechanical and environmental factors in environment sensitive cracking is shown in Fig. 4. The number of variables that affect environment sensitive cracking in light water reactor conditions is large and they possibly have a number of synergistic interactions.

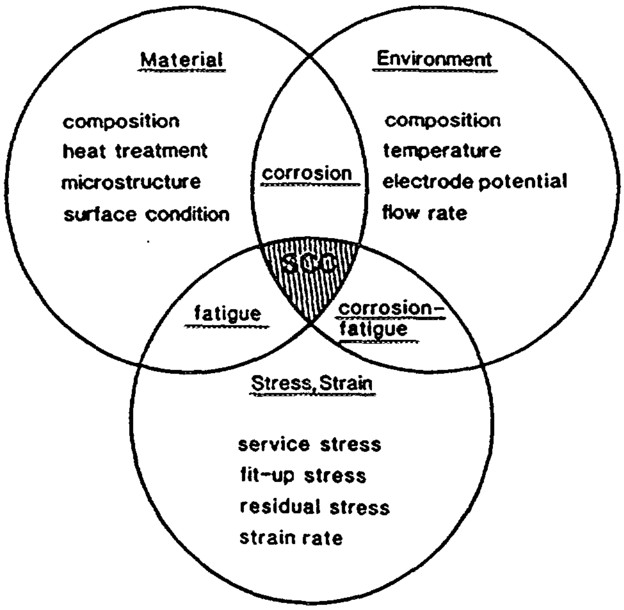

Fig. 4. Factors affecting environment sensitive cracking. Note that specific conditions are required for cracking to occur.

3.1 CARBON STEELS

Corrosion fatigue is the main problem associated with carbon steels in LWR's. The environment sensitive cracking properties of pressure vessel steels such as A533B and A508 have been studied to a large extent. Appendix A of Section XI of the ASME Boiler and Pressure Vessel Code presents a procedure for estimating the remaining useful life of a cracked reactor pressure vessel or nozzle. This procedure combines a fatigue crack growth analysis with a maximum allowable flaw size. Results obtained in PWR conditions lie usually below the ASME curves (see e.g. Cullen, 1985 & 1986). However, results obtained in pure water containing high dissolved oxygen contents indicate considerably higher crack growth rates than the ASME reference curves, Fig. 5.

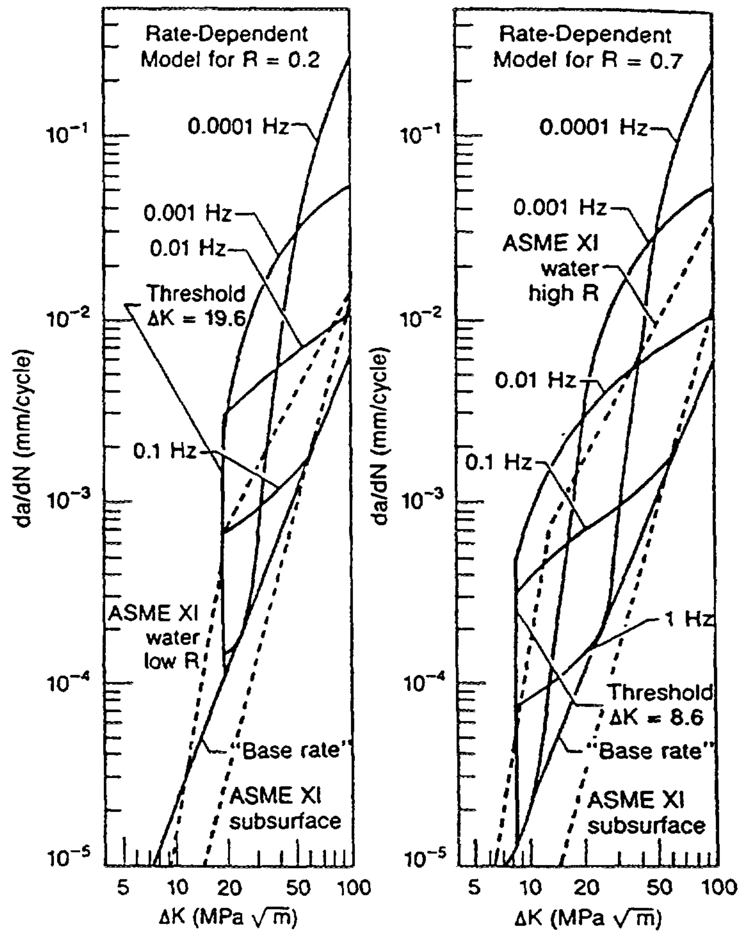

Fig. 5. Predictive curves based on time-dependent corrosion fatigue crack growth model for reactor pressure vessel steels in reactor grade water at 288°C (Gilman 1986).

The importance of metallurgical variables of steels is now clear, based on the laboratory test results. The sulphur content and, especially, the MnS inclusion size, shape and distribution in the steel seem to be responsible for material-to-material and heat-to-heat variability. Large elongated MnS inclusions generally contribute to rapid crack growth rates, whereas materials containing small spherical MnS inclusions are less susceptible. The sulphur species produced from the dissolution (H_2S, HS^- etc.) are known both to enhance hydrogen absorption and to increase anodic dissolution of the steel.

Besides the reactor vessels there has also been cracking in uncladded steam generator shells of PWR's, in steam generator feedwater piping of PWR's, in carbon steel piping of BWR's and in feedwater tanks of nuclear reactors, which are mechanistically relevant to pressure vessel steel problems.

Thermal fatigue is the primary cause of steam generator feedwater pipe cracking incidents induced by thermal stratification during low flow conditions during plant start-up and low power operation. Cracks are oriented circumferentially and located in the base metal outside the weld heat-affected zone. Pitting was associated with the initiation of the cracks. Temperature differences of the order of 120 °C have been measured from feedwater lines between the top and the bottom under low flow conditions. The main factor affecting this cracking is thought to be the large number of thermal stratification cycles (0.1-10 Hz); the feedwater chemistry, particularly the oxygen level, was of secondary concern. However, they both contribute significantly to crack initiation.

Strain-induced corrosion cracking (SICC) has caused cracks in the medium-strength, low alloy steel 17MnMoV64 and in the relatively high-strength, fine-grained, structural steel 22NiMoCr37 used in Germany for BWR piping and reactor vessel nozzles. This kind of cracking has caused circumferential cracking in the region of feedwater nozzles and at welds and axial cracking in pipe bends, but also cracking in straight sections of thin-walled piping in German BWRs. SICC refers to those corrosion situations in which the presence of localized, dynamic straining is essential for crack formation to occur, but in which cyclic loading is either absent or is restricted to a very low number of infrequent events. The high content of dissolved oxygen seems to be an important factor. Oxygen leads to the formation of mixed magnetite/hematite oxide films on low-alloy steel surfaces.

Erosion-corrosion has reduced pipe wall thickness in several PWR's. An oxide dissolution mechanism is believed to be the mechanism of erosion-corrosion. The interaction of piping design, flow velocity (5 ms^{-1}), temperature (Fig. 6) and pressure (193°C/25 bar), pH (8.8-9.2), unusually low amounts of alloying elements of the steel, particularly chromium (less than 0.02 %) (Fig. 7) as well as extremely low oxygen content (less than 4 ppb) in the water (Fig. 8), contribute to this type of degradation. Generally, this type of corrosion has been a problem on the secondary loop materials. Erosion-corrosion may become an increasing problem in BWR's, when the water chemistry is changed to a hydrogen water chemistry (HWC). Depending on the type of water chemistry and materials selected for NHR's, the efforts to avoid erosion corrosion should be included in the design.

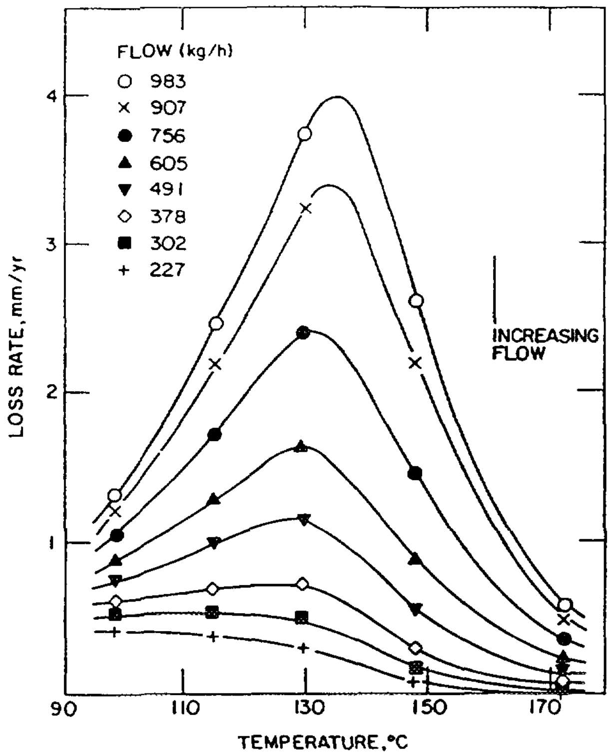

Fig. 6. Temperature dependence of post-orifice erosion-corrosion rates for mild steel in deoxygenated water (pH = 9.05) (Bignold et al. 1980).

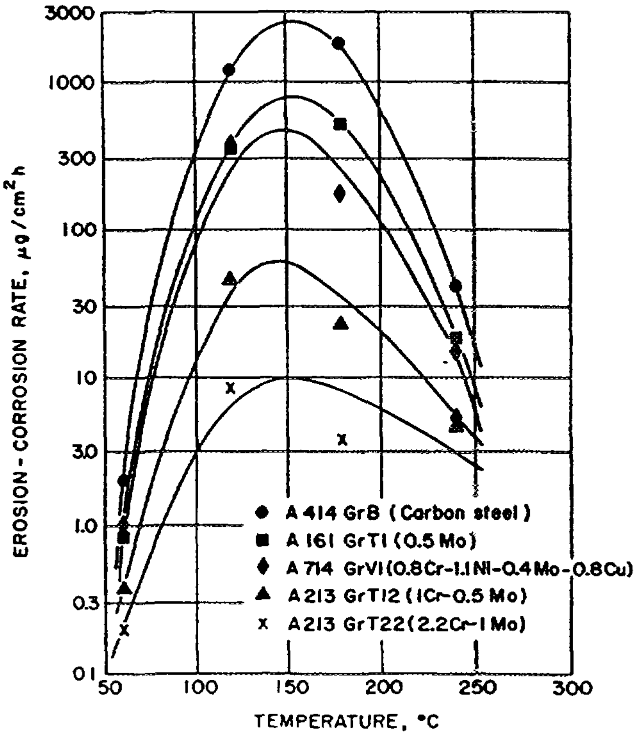

Fig. 7. Erosion-corrosion rate vs. temperature for various steels in water. pH = 7.0; p = 40 bar, flow rate = 35 m/s; CO_2 < 40 ppb (Heitmann and Schub 1983).

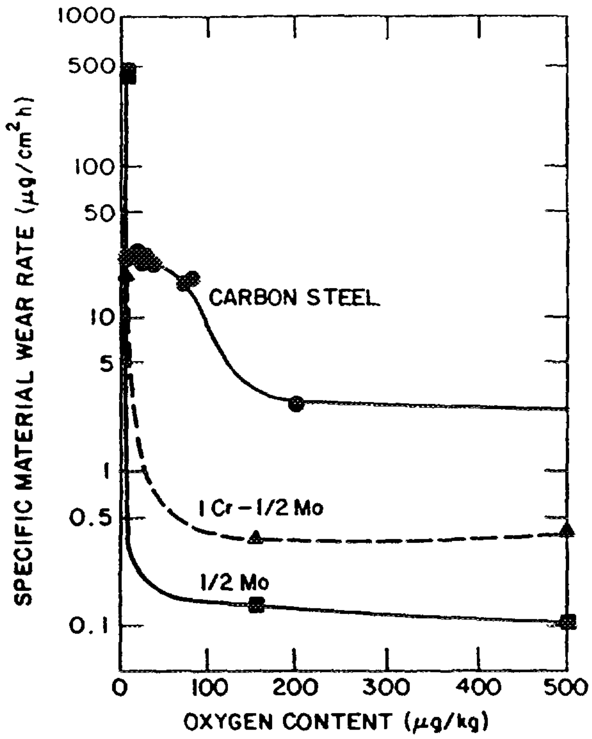

Fig. 8. Effect of oxygen content on material wear rate due to erosion-corrosion for various steels in neutral water for steel denominations (Heitmann and Schub 1983).

3.2 STAINLESS STEELS AND NICKEL BASE ALLOYS

Intergranular stress corrosion cracking (IGSCC) in the weld heat affected zones (HAZ) of AISI 304 and 316 stainless steel reactor pressure vessel nozzle safe-ends and piping have occurred in several BWR's. The BWR pipe cracking has occurred in the sensitized zones of AISI 304 and 316 type stainless steel weldments. After welding the degree of sensitization is generally low, but it can increase during operation due to low temperature sensitization (LTS). The development of a sensitized microstructure in LTS as a function of time is shown in Fig. 9. It can be seen that in about 10 years a marked change in the weld sensitized structure can be expected at reactor operating temperatures.

In addition to the sensitized microstructure and stresses the presence of a certain amount of oxygen in the coolant is necessary for BWR IGSCC. In general, by keeping the amount of oxygen in the coolant low enough IGSCC is inhibited, but the exact level depends on the conductivity of the water. To prevent crack formation in BWR stainless steel piping the electrochemical potential in the cooling water has to be kept more negative than about -250 mV_{SHE}. Under normal conditions of 100 - 300 ppb dissolved oxygen the electrochemical potential in BWR water varies from plant to plant between -100 and +100 mV_{SHE} which supports stress corrosion cracking. If the conductivity of the water is below 0.3 $\mu S\ cm^{-1}$, the 20 ppb of oxygen is sufficient low to keep the potential on the right level and to inhibit IGSCC. Therefore, hydrogen addition in conjunction with impurity control has become into widespread use in BWR practice for preventing and mitigating cracking in piping.

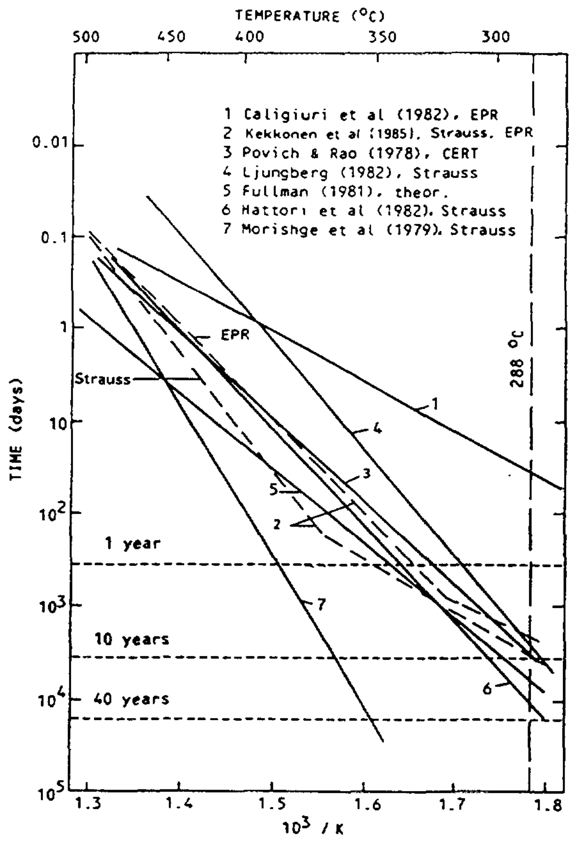

Fig. 9. Several (T, t) dependencies for LTS of AISI 304 steel showing wide scatter of various tests. The scatter is based on the various starting conditions used as well as test methods (Kekkonen et al. 1985).

Cold work and residual stresses has been attributed to a number of AISI 304 steel pipe cracking incidents in BWR's. Cracking in these cases can be mainly intergranular. An example of this type of cracking has occurred in AISI 304 elbows in the shut-down cooling and clean-up system of the Oskarshamn BWR plant. In cold bending the inner surface deformation (15-20 %) produced some α'-martensite, which initiated axial cracking in the pipe bend in the absence of sensitization. Extensive cold work should therefore be avoided in austenitic stainless steel primary circuit components.

IGSCC of Inconel 600 has been detected e.g. in a recirculation inlet nozzle safe-end weld at the Duane Arnold plant. The safe-ends were made of Inconel 600 forging and all safe-ends of the plant showed cracking essentially completely around the circumference. Cracking had initiated in the weld HAZ and propagated through the safe-end weldment (Inconel 182). This plant had marked resin intrusions which lowered the pH and increased the conductivity. A large number of recent cracking cases of Inconel 182 weld metal in BWR's emphasizes especially the need for good water quality control. In PWR's similar cracking incidents concerning Inconel 600 and Inconel 182 used in the pressure vessel head penetrations have been observed in many plants.

Thermal fatigue cracks accounts for only a small percentage of the total pipe cracks in stainless steel piping if compared to IGSCC. These cracks are transgranular and can be prevented by modifying the pipe system design and installing mixers to minimize the thermal gradients. These cracks may be a problem in areas which do not belong to current in-service inspection programs and, thus, for early identification of leaks a good leak detection system is important.

In stagnant borated water of PWR's a number of pipe cracking incidents have been reported. Cracking has occurred in low pressure systems in the heat-affected zones of AISI 304 stainless steel pipe welds. Investigations of the borated water system pipe cracks have shown that cracking is intergranular and occurs in a similar pattern to pipe cracking in BWRs. The major difference between BWR pipe cracking and PWR spent fuel pool pipe cracking is in the environment. The steady-state BWR environment consists of high-purity water at 288 °C containing dissolved oxygen while the PWR spent fuel pool environment normally consists of approximately 13000 ppm boric acid at about 65 °C.

Thiosulphate and tetrathionate anions lead to cracking of sensitized AISI 304 stainless steel and the potential range over which IGSCC occurs corresponds to a region for the metastable sulphur oxyanions in which thiosulphate and tetrathionate are capable of being reduced to elemental sulphur. A strong synergistic effect exists between thiosulphate and chloride. When mixed together the thiosulphate and chloride produced more pronounced IGSCC than either thiosulphate or chloride separately. This problem was solved by maintaining high-purity water chemistry and using AISI 304L steel to prevent sensitization.

Irradiation assisted stress corrosion cracking (IASCC) is a time-dependent phenomenon, which needs a minimum residence time or threshold fast neutron fluence to occur, Fig. 10. IASCC can occur at very low stresses, but there is an apparent fluence threshold > 5 x 10^{20} n cm^{-2} (E > 1 MeV) as perceived today. Because the fluence and time dependencies of IASCC are not known precisely at low temperatures, this phenomenon may have great significance in the case of NHR's on in-core component life times.

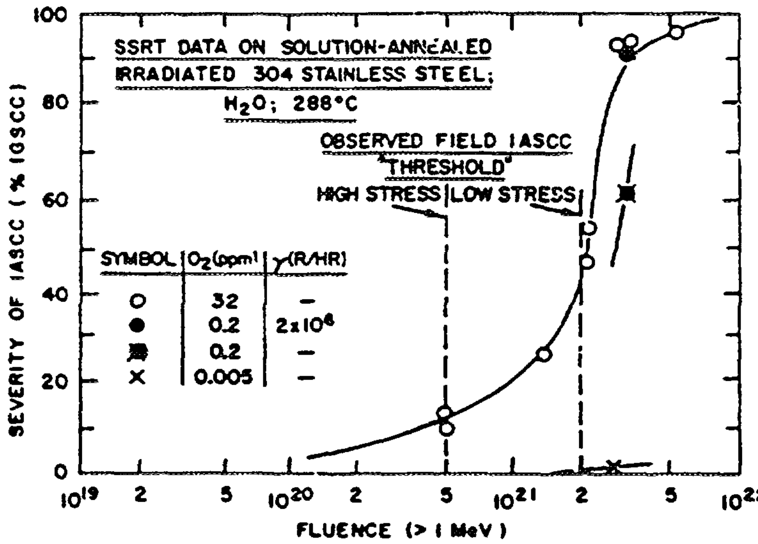

Fig. 10. Relationship between the severity of intergranular stress corrosion cracking and fluence for irradiated type 304 stainless steel under slow strain rate conditions in water at 288°C containing different amounts of oxygen (Andresen, 1989).

3.3 STEAM GENERATOR MATERIALS PROBLEMS

Primary water stress corrosion cracking (PWSCC) has been confined to inner row U-bends, severely dented tubes in tube support plate crevices, and roll-expanded areas of the tubesheet region. In general, multiple cracks are observed, suggesting the pattern of residual stresses, e.g., from the rolling operation at the roll transition zone. Dissolved hydrogen in primary side water accelerates cracking.

In general it is known that there is a great heat-to-heat variability to SCC in Inconel 600 tubing. Analyses of tubes taken from steam generators show that carbide distribution is of great importance in tube cracking: cracking is found in tubes with intragranular carbides while tubes with grain boundary carbides are free of cracking. The positive effects of intergranular carbides on cracking have been explained on the basis of their effects on local deformation and dislocation structure; grain boundary carbides decrease strain localization near grain boundaries. The factors which have often been assumed to contribute to SCC of Inconel 600 in operating steam generators have been summarized (Paine 1982):

- high hardness, high strength, low carbon content tubing;
- high degree of cold work in the bending or roll expansion;
- ovalization of tubes during bending, i.e., multiaxial stress state;
- residual and operating stresses approaching the yield stress;
- heat-to-heat factors (e.g., metal composition, grain boundary condition, heat treatment, etc.).

The above five factors must be taken into consideration together with the local environment in order to assess cracking susceptibility. PWSCC is a thermally activated process with a high activation energy. Therefore, the reduced operation temperature may be the remedy for reducing failures.

The tubes and tube supports in steam generators of VVER type reactors are made of titanium-stabilized austenitic stainless steel (AISI 321). The tubes are expanded by rolling and tightness-welded to the primary collectors. So far, experience with the austenitic tubes has been excellent. No denting stress corrosion or wastage corrosion have been observed yet. This is mainly due to the horizontal position of the steam generators, low operation temperature, the austenitic stainless steel tube supports, condensate polishing, effective deaeration and all volatile treatment of the secondary water.

3.4 HIGH STRENGTH MATERIALS DEGRADATION

Stress corrosion cracking (SCC) of age-hardenable nickel-base alloys such as Inconel X-750 and Inconel 718 and stainless steel A-286 were developed originally for high temperature applications, with good mechanical properties and excellent resistance to high temperature oxidation. For Inconel X-750 numerous heat treatments developed for the special high temperature applications have been used for the different structural parts in LWRs. The experience with Inconel X-750 in reactors has been contradictory. The known cases of SCC failures in LWRs are listed in Table 5.

Table 5. Reactor components of age-hardenable austenitic alloys which have experienced cracking.

Alloy	Component	Reactor type	Initiation mode
X-750	Bolts:		
	Core baffle	PWR	IGSCC
	Fuel assembly	BWR	IGSCC
	Pins:		
	Guide tube support	PWR	IGSCC
	Beams:		
	Jet pump	BWR	IGSCC
	Springs:		
	Control rod drive seal	BWR	IGSCC
	Fuel assembly holddown	PWR	Fatigue
	Fuel assembly finger	BWR	IGSCC
718	Springs:		
	Control component holddown	PWR	Fatigue
A-286	Bolts:		
	Thermal shield	PWR	IGSCC/Fatigue
	Fuel assembly	BWR	IGSCC
	Core barrel	PWR	IGSCC
	Beams:		
	Steam separator/dryer/holddown	BWR	IGSCC

The most important applications of the threaded fasteners are those constituting an integral part of the reactor coolant pressure boundary, such as pressure retaining closures in reactor vessels, pressurizers, reactor coolant pumps, and steam generators. Many of these failures have been caused by erosion-corrosion in PWR primary water leaks. Boric acid is significantly acidic at low temperatures and together with a high velocity jet of leaking primary water severe wastage of ferritic materials can occur.

5 CONCLUDING REMARKS

Of various types of corrosion problems in the nuclear industry, stress corrosion and corrosion fatigue have been the most important causes of failures in reactor pressure boundary materials. These environment sensitive cracking incidents in pressure vessels, piping and heat exchanger have given rise to safety concerns and have led to the replacements of some major components after only a small fraction of their design lifetime. Improvements in the material technology may not help any more the existing power plants with their problems. However, improvements in the field of water chemistry control, more stringent guidelines and improvements in the on-line monitoring technology due to new high temperature sensors have excellent changes to prevent cracking incidents in the future.

Many of the above failure modes are time-dependent and, thus, are expected to become more prevalent with ageing power plants. For safe operation in the future various ageing assessment techniques, such as NDE, statistical methods, transient data collection, water chemistry control and operational strategies and predictive models for various forms of failures have to be developed.

NHR's can be designed using the present available failure experiences of existing power plants. In this case also the material selection can be optimized in order to avoid the same problems encountered in existing power plants. Furthermore, the designed low operational temperature provides some additional safety if the corrosion properties of the materials are reviewed and solved in a proper way.

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AST-500 SAFETY ANALYSIS EXPERIENCE

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Abstract

Characteristic AST-type NHR safety features and requirements are described briefly. The main approaches and results of design and beyond-design accidents analyses for the AST-500 NHR, and the results of probabilistic safety assessments are considered. It is concluded that the AST-500 possesses a high safety level in virtue of the development and realization in the design of self-protection, passivity and defence-in-depth principles.

1. INTRODUCTION

The AST-500 reactor plant is intended to generate low grade heat for district heating and hot water supply to cities. The NHR specific features, as well as the economic-dictated necessity to locate a nuclear district heating plant (NDHP) close to a city caused strict requirements for the safety of these plants. Therefore, the AST-500 NHR has been developed proceeding from the necessity to ensure a qualitatively higher level of safety compared with NPPs.

The AST-500 safety philosophy is based primarily on the application of self-protection and passive safety principles in the design and on the development of the defence-in-depth concept. The AST-500 reactor is a PWR with an integral layout of the primary components, and with natural convection of the coolant with reduced working parameters. The main characteristics and design decisions for the AST-500, as well as the basic information concerning its safety are presented in [1-4].

The AST-500 design has been considered by the State Regulatory Body. An independent safety review of the twin AST-500 Gorky NDHP has been carried out by an experienced international team of experts under the aegis of IAEA (Pre-OSART mission). It has been given a positive estimate of the safety and the validity of the design bases and the main engineering solutions.

2. REQUIREMENTS FOR NDHP SAFETY

The design of the AST-500 NHR, taking into account its specific purpose and site location, was carried out under the imperative necessity to ensure a qualitatively higher level of safety compared with NPPs. Priority has been given to engineering decisions which provide a reliable solution of safety issues and, first of all, to a reactor plant intrinsic safety by virtue of natural processes and the application of passive safeguard systems.

The NHR specific features necessitated to develop additional requirements for safety compared with that for NPPs, which were then included into the top-level national regulatory document on nuclear plant safety [5,6]. The requirements were incorporated to account for external impacts such as aircrash and shock wave. In addition, more strict operational safety limits for fuel damage were adopted.

For the AST-500 safety analysis the following safety criteria for design accidents have been established:

1) The value of DNBR should exceed 1.0 taking into account the most unfavourable deviation of parameters, maximum non-uniformity of heat release, maximum inaccuracy in calculations of local energy release and critical heat flux.
2) Maximum fuel temperature should be below the melting point.
3) Maximum pressure in the primary circuit should not exceed 1.15 of the design value.
4) Maximum pressure in the guard vessel (GV) at primary coolant leaks should be below the design value.
5) The core should remain covered with coolant.

3. BASIC DECISIONS FOR ENSURING SAFETY

The AST-500 reactor is based on well-developed and proven types of LWRs and is characterized by a safety-important power self-control property in virtue of its negative reactivity feedbacks: temperature, power and void reactivity.

The AST-500 NHR uses natural convection of the primary coolant in all operating modes and an integral arrangement of the primary components which allowed to simplify the primary circuit, to eliminate the reactor coolant pumps and the primary pipeline system. It also excludes large and medium break loss-of-coolant accidents and limits a leak size in case of a rupture of a pipeline of less than 50 mm diameter (equivalent diameter). Natural convection of the primary coolant results in a coolant flowrate increase following a thermal power rise, and in flow self-profiling over the core fuel assemblies which results in significant margins against boiling crisis both in normal and emergency modes of reactor operation.

The passive ERHR system is made of three channels and operates under natural convection of the coolant. The water inventory in the ERHR tanks provides continuous system operation during several days. One ERHR channel is sufficient for cooling down the scrammed reactor. Secondary coolant evaporation with steam blow-off through the pilot-operated safety valves on the secondary circuit pressurizer is used as a back-up ERHR system.

The reactor is enclosed by the guard vessel. It serves for keeping the core covered with coolant after LOCAs, and for radioactivity confinement in case of a primary circuit loss-of-integrity accident. Double isolation valves are installed in all auxiliary pipelines of the primary circuit extending beyond the guard vessel boundaries.

To protect heat consumers against radioactivity, a three-circuit heat transport scheme with a pressure barrier between the intermediate and the heating grid circuits is used.

The reactor plant and the auxiliary pipelines of the primary circuit are housed in a leak-tight containment or in leak-tight compartments equipped with radioactivity suppression devices (filters, bubblers).

The following characteristics are essential for the reactor plant safety:

- low core power density;
- low reactor parameters (P=2.0 MPa, Tav.=170°C);
- large water inventory in the reactor which increases the reactor inertia and causes a relatively slow variation of parameters in accidents. The duration of emergency processes in the reactor plant amounts to tens of minutes and even hours. The reactor inertia and its slow dynamics are the important features of the reactor plant which give an ample time margin for making sound decisions on corrective measures and, if necessary, for correction of plant personnel errors.

4. DESIGN ACCIDENTS ANALYSIS

4.1. Initiating events

To carry out the analysis of design accidents, a list of initiating events was compiled on the basis of analysis of possible failures of equipment items and personnel errors, and of internal and external impacts. All initial events, similar to those presented in the IAEA recommendations [9] on the example of power reactors, were considered.

Naturally, the AST-500 specific engineering decisions were taken into account in the analysis. As an example of the AST-500 specific features one can mention the absence of reactor coolant pumps, turbine plant, safety valves in the primary circuit and, respectively, the impossibility of initial events caused by their failures. Elimination of soluble boron reactivity control excludes reactivity events caused by decrease of absorber content in the primary coolant, and so on.

A wide range of various emergency situations caused by heat removal disturbances, reactivity variations, primary system loss of integrity, failures at fuel handling and external impacts (aircrash, shock wave, earthquake) were considered in the safety analysis.

4.2. Analysis technique

In the analysis of reactivity and heat-removal accidents the DKAST-8 computer code was used. Loss-of-coolant accidents were calculated with the UROVEN/MB-3 code. These codes have been developed at OKBM specifically for the design analysis of integral reactor (AST-type) thermal-hydraulics. They take into account such AST-specific features as natural convection of the primary coolant, availability of built-in steam-gas pressurizer, guard vessel and passive ERHR systems. They are based on experimentally verified calculation models and correlations. Verification of the codes was performed on the basis of experimental results on separate effects and phenomena, and on integrated data obtained on integral reactor models which confirmed the reliability of calculational prediction of the reactor plant thermohydraulic characteristics in emergency modes.

The consideration of design accidents has been carried out taking into account the reactor protection system operation and the imposition of failures in accordance with requirements of the regulatory document proceeding from the single failure principle [5,6]. A conservative design approach was used according to which the accident analyses have been performed for the most unfavourable initial state, proceeding from the most pessimistic assumptions concerning the efficiency of the safety systems and the equipment. Possible deviations of parameters in the unfavourable direction within the limits of their maintaining and measuring accuracy were taken into account. For LOCAs and loss-of-heat removal accidents, a nominal power operation mode with increased parameters corresponding to the preventive protection setpoints was taken as an initial state.

4.3. Main results of safety analysis

The safety analysis have shown that for design accidents the prevention of reactor parameters to deviate beyond the limits allowable for normal operation is ensured by the reactor control system or by the emergency protection actuation and the available protective systems. At primary circuit loss-of-integrity accidents the core is kept under coolant during not less than 3 days because of the available localizing means (guard vessel, localizing valves) without connecting the make-up system to the reactor. The maximum pressure in the guard vessel amounts to 1.0 MPa. Reliable core cooling is ensured for all design accidents.

The consequences of design accidents are not radiologically dangerous, so it was concluded that the reactor plant meets the safety requirements for NDHPs located close to large cities.

5. BEYOND-DESIGN ACCIDENTS ANALYSIS

A wide spectrum of beyond-design accidents, characterized by the superposition of failures in the safety systems and personnel errors additionally to the single failure principle, has been considered in the design analysis. The most severe sequences, as concerned reactor core overheating or damage hazard and radioactivity releases beyond the protective barriers, were selected for the analysis.

A class of the most severe events - socalled hypothetical accidents, was singled out among the beyond-design accidents. Failure of entire safety systems, such as the reactor emergency protection, residual heat removal and confinement systems was postulated when considering the hypothetical accidents. For these accidents the reactor plant inherent safety properties play the role of the first protective barrier, taking into account the postulated failure of the engineered safeguard systems. To control these accidents, the use of additional back-up safety systems is envisaged in the design according to their technical capabilities and characteristics.

5.1. Unscrammed loss of heat removal to the grid with failure of three ERHR channels

This accident is considered as an ultimate event among the loss-of-heat removal accidents. The accident is accompanied by the heating up of the primary and secondary circuits and by rising pressure in the reactor and in the secondary circuits (Fig.1). As coolant temperature rises, the core power reduces due to neutron power self-control effects. When the pressure in the secondary circuit reaches 1.6 MPa the safety valves open and heat removal from the reactor is determined by the evaporation of secondary coolant. In 1000s from the beginning of the accident the reactor power falls down to the level of heat removal to the secondary circuit, and the pressure in the reactor reaches its maximum value of 3.5 MPa. Then the Xenon-135 poisoning effect causes the reduction of core power and primary system pressure.

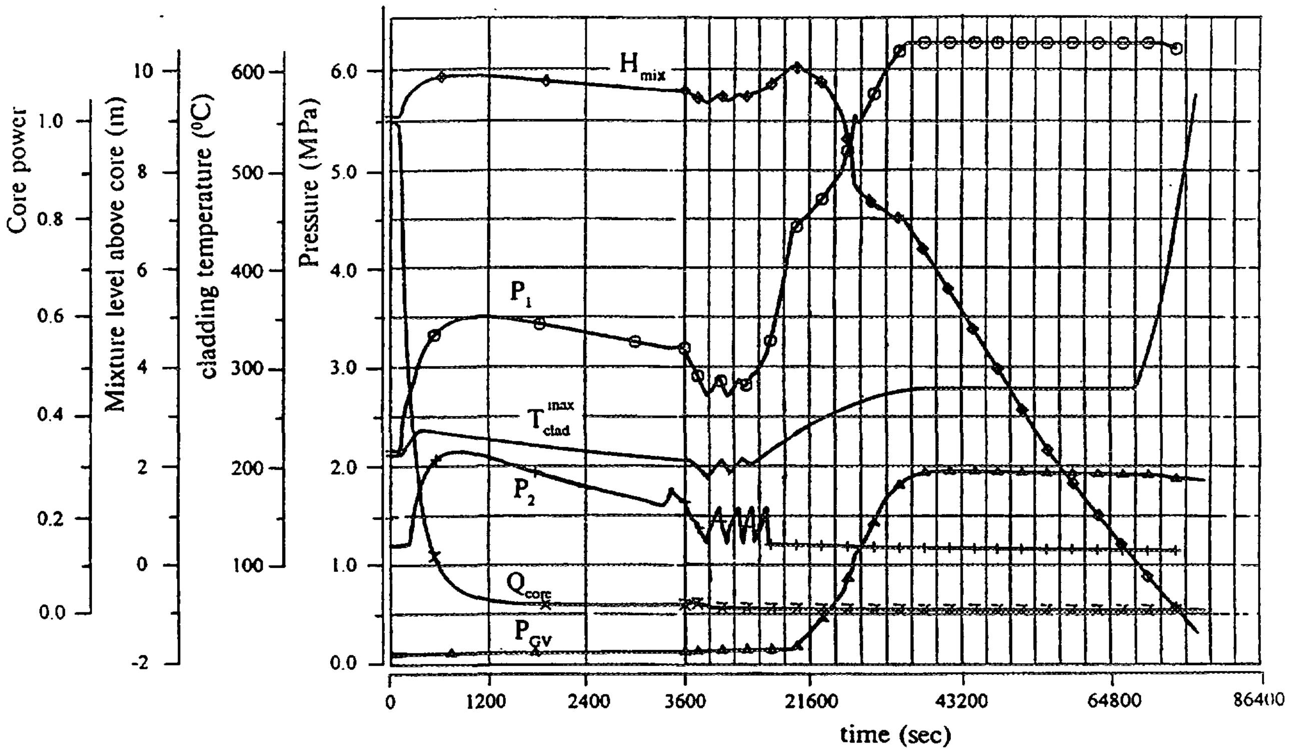

Fig.1. "Stop-grid" ATWS with failure of ERHRS and secondary make up system

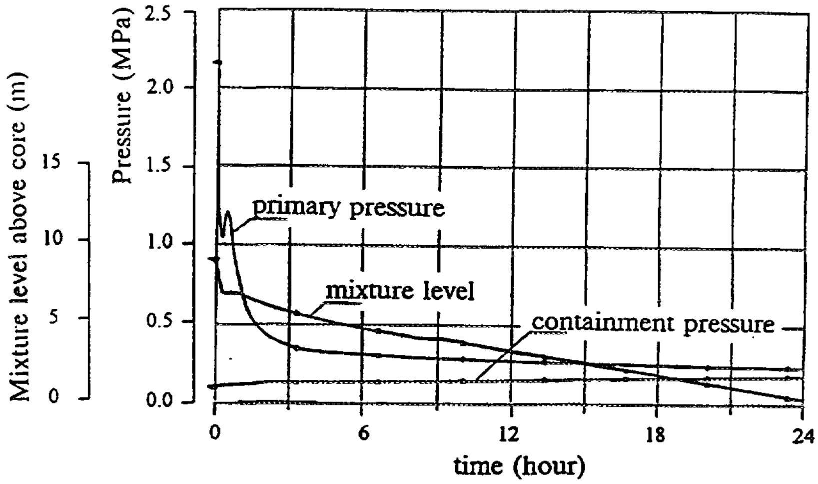

Fig.2. **AST-500. Rupture of cleanup system pipeline beyond GV with failure of localizing valves and of two ERHR channels**

When considering the accident, an additional failure of the secondary circuit make-up system was adopted which resulted in a complete loss of heat removal from the reactor following secondary coolant evaporation. The primary pressure is building up, the main joints of the reactor and guard vessels open, and primary coolant evaporation begins. The maximum value of the primary pressure amounts to 6.3 MPa and to 1.9 MPa in the GV. Both vessels remain intact. To prevent core dry-out it is sufficient to resume heat removal from the reactor after the time of 19 hrs.

5.2. The coolant purification system intake pipeline rupture beyond GV with failure to actuate two localizing valves and two ERHR channels

For this accident the rupture of the maximum diameter pipeline is considered, followed by a primary coolant outflow beyond the GV boundaries. Only one of three ERHR channels is operated. Failure of two independent ERHR channels means failure of at least four connecting valves and represents an extremely improbable event.

In this accident, a long-term steam outflow occurs, accompanied by a slow lowering of the coolant level following the steam-water outflow (**Fig.2**). The reactor core is kept under coolant during 24 hrs without initiation of the reactor make-up system. Provisions for accident termination include the closure of valves and making up the reactor with water.

6. PROBABILISTIC SAFETY ANALYSIS

A level-1 probabilistic safety analysis has been performed for the AST-500 regarding internal initial events associated with a loss of heat removal from the reactor, inadvertent positive reactivity insertion and primary circuit loss of integrity. A fuel cladding temperature of 1200°C was taken as a criterion for core damage.

The analysis is based on the event tree procedure for possible accident sequences consideration, on sequential - parallel logics (failure tree analogue) and on the minimum cut section method for analyzing safety systems reliability [7].

Various operational states of the reactor plant were considered for the analysis: reactor start-up, power operation, core refueling. Erroneous actions of personnel during plant emergency control at remote or manual actuation of the safety systems were considered. The problems of emergency situation identification, diagnostics of accidents and of appropriate actions were analyzed as well.

Dependent-on-initiating event failures of individual items of the systems, and structural-functional dependencies existing between systems were taken into account in the analysis. It is assumed that equipment of the same design is subjected to common cause failures both on a system and an intersystem level.

Simulation and quantitative analysis of the reliability of the safety systems and of the accident sequences were carried out using the computer code package explained in [8].

The analysis has shown that the point estimation of the core damage frequency is less than 10^{-8} per reactor-year. The most significant kinds of accidents, affected safety systems and devices, and the types of equipment failures were determined. The respective uncertainties were also analyzed.

7. APPROACH TO SEVERE ACCIDENTS

To prevent severe accidents, the following principal decisions were realized in the design of the AST-500 reactor plant:

- integral layout of the primary components which eliminates large and medium size loss-of-coolant accidents;

- guard vessel, which serves as an additional passive confinement system providing for keeping the reactor core under coolant and for localizing radioactivity at LOCAs;

- passive system for emergency heat removal through the heat exchangers built into the reactor.

The analysis was performed for a wide spectrum of the beyond-design accidents with postulated failures of the highly reliable safety systems (emergency protection one, emergency heat removal channels, localizing systems). It showed that inadmissible overheating and damage of the core does not arise in most severe beyond-design accidents. The maximum consequences of beyond-design accidents resulted from the loss of integrity of the most heat-rated fuel rod cladding. In virtue of the reactor self-protective properties an ample time margin (not less than 19 hrs) is available in beyond-design accidents until conditions appear at which core overheating and severe damage would become possible. Taking into account the ample time available to undertake appropriate corrective actions, means are provided in the reactor plant to control most unforeseen accidents and to prevent severe damage and melting of the reactor core.

The scenarios of the most severe beyond-design accidents in the AST-500 and the time margins available until core dry-out begins, and the appropriate accident control provisions are given below.

BEYOND-DESIGN ACCIDENTS AND ACCIDENT CONTROL MEANS CHARACTERISTIC FOR AST-500

Accident	Safety system failures	Time margin, hrs	Means for accident control
Loss of heat removal to grid	Emergency protection failure to actuate Three ERHR channels failure Secondary circuit make-up system failure	19	Connection of primary and secondary make-up systems, ERHR and boron injection system
Primary pipeline rupture inside the guard vessel	GV loss of integrity Two ERHR channels failure	>24	Connection of ERHR, primary and secondary make-up system
Primary pipeline rupture beyond the guard vessel	Localizing valves failure Two ERHR channels failure	24	Connection of ERHR, primary and secondary make-up system. Valves closure
Rupture of primary/secondary HX header	PORV failure to close. Isolation valves failure to close One ERHR channel failure	24	Isolation valve closure Connection of ERHR, primary and secondary make-up system
Reactor bottom loss of integrity	Two ERHR channel failure	24	Connection of ERHR, primary and secondary make-up system

8. CONCLUSION

The principal decisions adopted for the AST-500 design, such as the use of an integral PWR with natural convection of the primary coolant, an emergency heat removal system operating on the passive principle, use of a guard vessel, reduced working parameters and core power density provide a high level of reactor plant safety.

For the design basis accidents the safety criteria are met with some margin for at least 3 days without power supply and without plant personnel intervention due to the passive devices and systems available.

For beyond-design accidents with postulated multiple failures in safety systems, long-term core cooling is provided. Thus, fuel cladding overheating and loss of integrity are eliminated.

The radiological consequences of the accidents are considerably lower than the limits specified by the regulatory codes and correspond to the respective design requirements.

A considerable time margin is available before core overheating and melt conditions occur, and wide possibilities exist for accident control allowing to prevent a severe core damage. Evaluation of the frequency of postulated scenarios resulting in severe core damage gives a value of less than 10^{-8} per reactor-year.

The design decisions adopted for AST-500 safety provision and its in-depth protection in beyond-design accidents allow to refer the AST-500 reactor plant to the category of enhanced safety systems.

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SAFETY ANALYSES FOR NHR-200

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Abstract

The NHR-200, is a commercial 200-MW District Heating Reactor developed in China. It is designed on the basis of design, construction and four-year operating experience of the 5MW Experimental Heating Reactor (NHR-5). It has special safety features which are briefly described in this paper. Accident classification and safety criteria are also explained. Some typical and serious accidents are studied theoretically, and their results are detailed in this paper. They demonstrate the excellent safety characteristics of HR-200.

1. DESIGN FEATURES OF NHR-200

The 200-MW district heating reactor NHR-200, is designed on the basis of former experience of design, construction, and 4-year operation of the NHR-5.

The main design parameters are listed in Table 1. The primary coolant circuit of the NHR-200 is a self-pressurized one with natural circulation. All its components are integrated in a pressure vessel. The natural circulation of the primary coolant is built up by the buoyancy from the coolant density difference between the riser and the downcomer. There are two intermediate coolant circuits transferring the heat from the primary coolant circuit to the heat grid. The intermediate circuit can effectively separate the heat grid from the primary coolant circuit. There are two independent, natural circulation residual heat removal systems. Each one can assure long-term cooling of the core after reactor scram.

The patented technique of hydraulically driven control rods assures that no rod ejection accident can happen, and that the rods can automatically fall down into the core by gravity after a trip of electrical power. The redundant boron injection system, driven by gravity, can shut down the reactor in a hypothetical control rod failure.

The NHR-200 is designed as a double pressure vessel reactor (the pressure vessel and the containment which houses the pressure vessel). The gap between the two vessels is so narrow that the reactor core can always be kept submerged in water, even during hypothetical pressure vessel cracking accidents.

All penetrating pipes of the reactor pressure vessel (RPV) are concentrated in the upper part of the RPV. Limited to less than 50 mm, their diameters are designed as small as possible, and their location is far above the reactor core to limit potential coolant discharges.

Based on the above passive safety features, the NHR-200 is a highly inherent safe reactor. Even in beyond design accidents, serious radiation release can be avoided.

TABLE 1: MAIN DESIGN PARAMETERS OF NHR-200

Name	Unit	Value
Rated thermal power	MW	200
Pressure of the primary coolant circuit	MPa	2.5
Core inlet/outlet temperature	°C	145/210
Core coolant flow rate	t/h	2376
Intermediate circuit pressure	MPa	3.0
Intermediate circuit temperature	°C	95/145
Intermediate circuit flow rate	t/h	3400
Net work temperature	°C	130/80
Fuel assembly type		12x12-2
Fuel assembly number (initial core)		96
Enrichment of fuel (initial core)	%	1.8/2.4/3.0
Average core power density	kW/l	36.23
Average fuel linear power density	W/cm	77.
Max. fuel cladding temperature	°C	240
Max. fuel pellet temperature	°C	1366
Min. DNBR		2.65
Initial core equivalent diameter	M	1.9
Control rod number		32

2. ACCIDENT CLASSIFICATION AND SAFETY CRITERIA

2.1. Accident classification of NHR-200

According to the nuclear safety code HAF0200(91), the conditions of a Nuclear Heating Station can be classified into two types: operation and accident conditions. Class I (normal operation) and Class II (anticipated operational transients) belong to operational conditions. Class III (infrequent faults), Class IV (limiting faults) and additional conditions are included under accident conditions.

2.2. Accident types of initiating events

1) Increase of heat removal from RPV
2) Decrease of heat removal from RPV
3) Anomalous reactivity and power distribution
4) Increase of primary coolant in RPV
5) Loss of coolant accidents (LOCA)
6) Anticipated transient without scram (ATWS)

2.3. Safety criteria

1) Loss of primary circuit coolant should be less than 138t, it means that the core is always covered by water.
2) To avoid fuel burnout, the minimum DNBR should be above 1.35 for all accidents.
3) The Maximum fuel temperature should be below 2590°C.
4) The maximum fuel specific enthalpy must be limited to avoid fuel pellet break down. The maximum pellet radial average specific enthalpy must be less than 837 kJ/kg UO_2 for a Class III or Class IV accident.

3. Analytical method and initial assumptions

The RETRAN-02 code is used to analyze the accident transients of NHR-200. The code was developed by EPRI and qualified by NRC of USA. It is used to analyze accidents and operational transients of PWRs and BWRs.

It is important to select conservative initial conditions and parameters for different accidents, including fuel temperature coefficient and moderator temperature coefficient. Trip signals taken in the analyses for reactor scram are:

- overpower protection, short reactor period protection;
- high primary coolant pressure protection;
- low primary coolant pressure protection;
- high primary coolant level protection;
- low primary coolant level protection;
- low intermediate circuit pressure protection;
- low intermediate circuit flow rate protection;
- high containment pressure protection.

Appropriate delays of signal transmission and rod dropping time are conservatively considered.

There are safety valves set on the pressure vessel to protect the integrated pressure boundary of the primary circuit. The opening pressure is 3.0 MPa, and the closing pressure is 2.6 MPa.

4. ACCIDENT TRANSIENT ANALYSES

Five important accident transients were analyzed and the results are presented in this paper. Some of them are typical accidents, the others are limiting accidents. Results of limiting accidents are given only to demonstrate the passive safety characteristics of NHR-200 and its outstanding robustness in extremely serious accidents.

4.1. Rupture of boron injection tube

It is a typical accident of decrease in primary coolant and is a Class III accident.

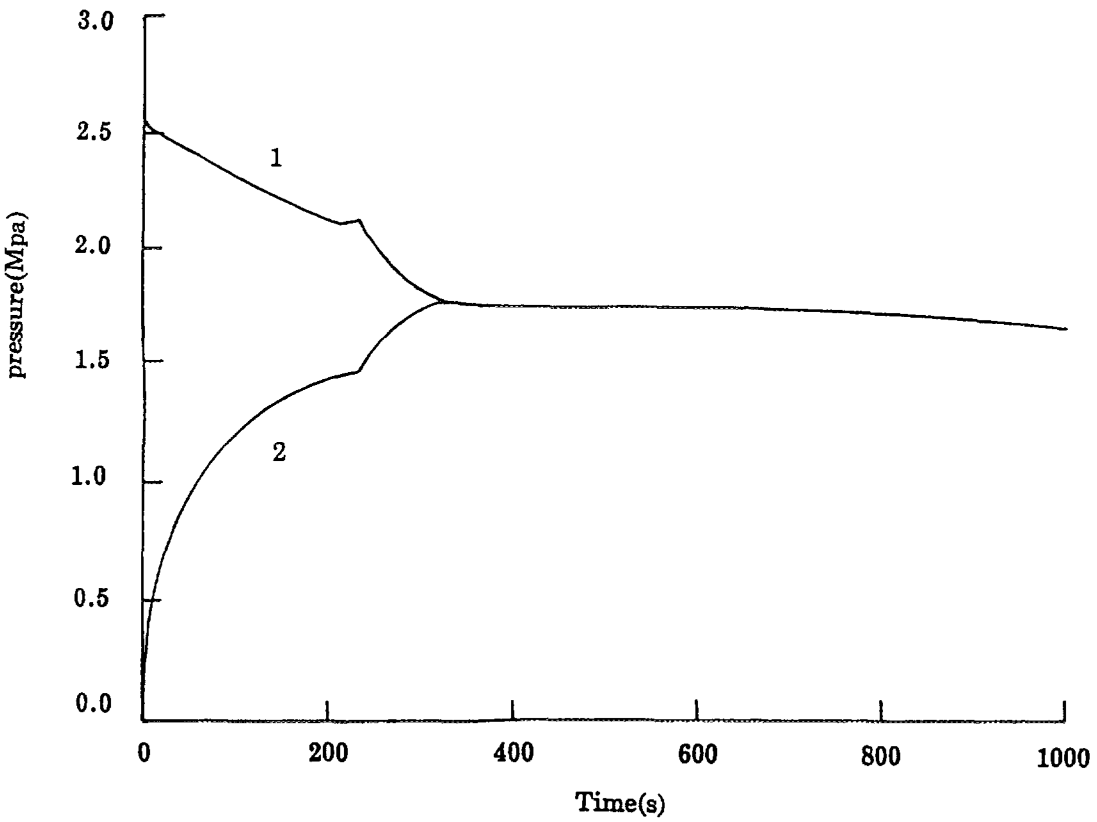

Fig.1 Rupture of coolant penetrating tube
1. in the pressure vessel
2. in the containment

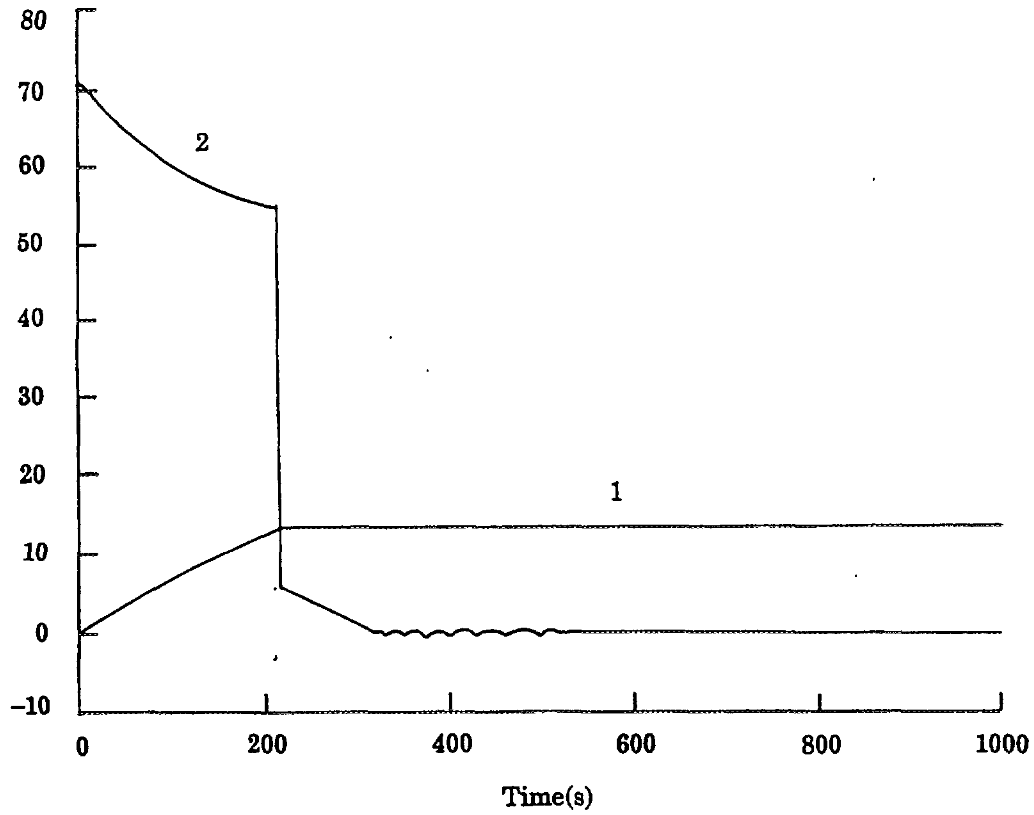

Fig.2 Rupture of coolant penetrating tube
1. total loss of water (t)
2. leakage flow rate (Kg/s)

4.1.1. Assumed conditions

1) Boron injection tube (50mm diameter) ruptures immediately between the pressure vessel and the containment.
2) Reactor scram.
3) Isolation of intermediate circuits after scram.
4) One residual heat removal system is in operation.

4.1.2 Accident transient and sequence

After the tube rupture, primary coolant is discharged to the containment, the liquid level in the pressure vessel drops, and the pressure in the containment rises. 9.2 seconds after the tube rupture, a high containment pressure reaches the trip point. 48.8 seconds after the rupture, the low pressure in the pressure vessel gives the second trip signal, then control rods fall down into the core, and the reactor shuts down. 218 seconds after the rupture, the primary coolant level drops to the level of the boron injection pipe, then the liquid discharge turns to steam discharge, so the discharge mass flow rate decreases conspicuously. The pressure in the pressure vessel goes down markedly, and the pressure of the containment rises rapidly. After 340 seconds of steam discharge, the pressures in the pressure vessel and in the containment become balanced at 1.74 MPa, and the coolant discharge ends. At the end of the accident the primary coolant level is still higher than the top of the core, above which there is still 86.1 tons of water. Figs. 1 and 2 show the changes of some parameters during the accident.

4.2. Unexpected opening of a safety valve

It is a typical loss of coolant discharging into a special blow-down tank. It is a Class III accident.

4.2.1. Assumed accident conditions

1) Safety valve stays in the keeps opening position,
2) Reactor scram,
3) Isolation of all intermediate circuits after scram,
4) One residual heat removal system in operation.

4.2.2. Accident transient and sequence

After the failure, primary coolant discharges to the discharge tank in steam phase, with an initial discharge flow rate of 2.58 kg/s, causing the pressure in the pressure vessel to go down.

After 500 seconds, the high vessel pressure reaches the trip point. At 887 seconds, the low primary liquid level signal triggers the reactor protection system to shutdown the reactor. Then the intermediate circuits are isolated, and one residual heat removal system starts to operate. Primary coolant pressure and water level fall continuously. After 10800 seconds, the pressure in the two vessel reaches 0.53 MPa, and the accident can be considered terminated. During the accident transient, the total coolant loss from the pressure vessel is 10.4t, the core is still covered by 89.6t of water.

Fig. 3 demonstrates the transient of leakage flow rate and the total loss of water.

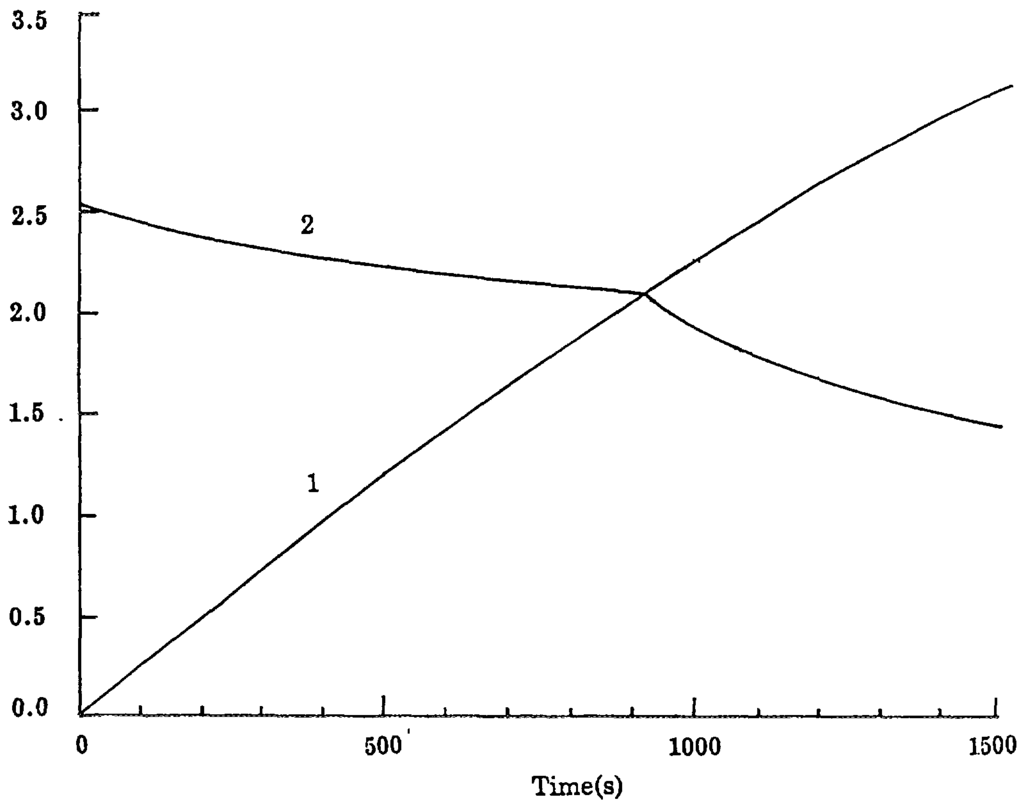

Fig.3 Unexpected opening of a valve
1. total loss of water (t)
2. leakage flow rate (Kg/s)

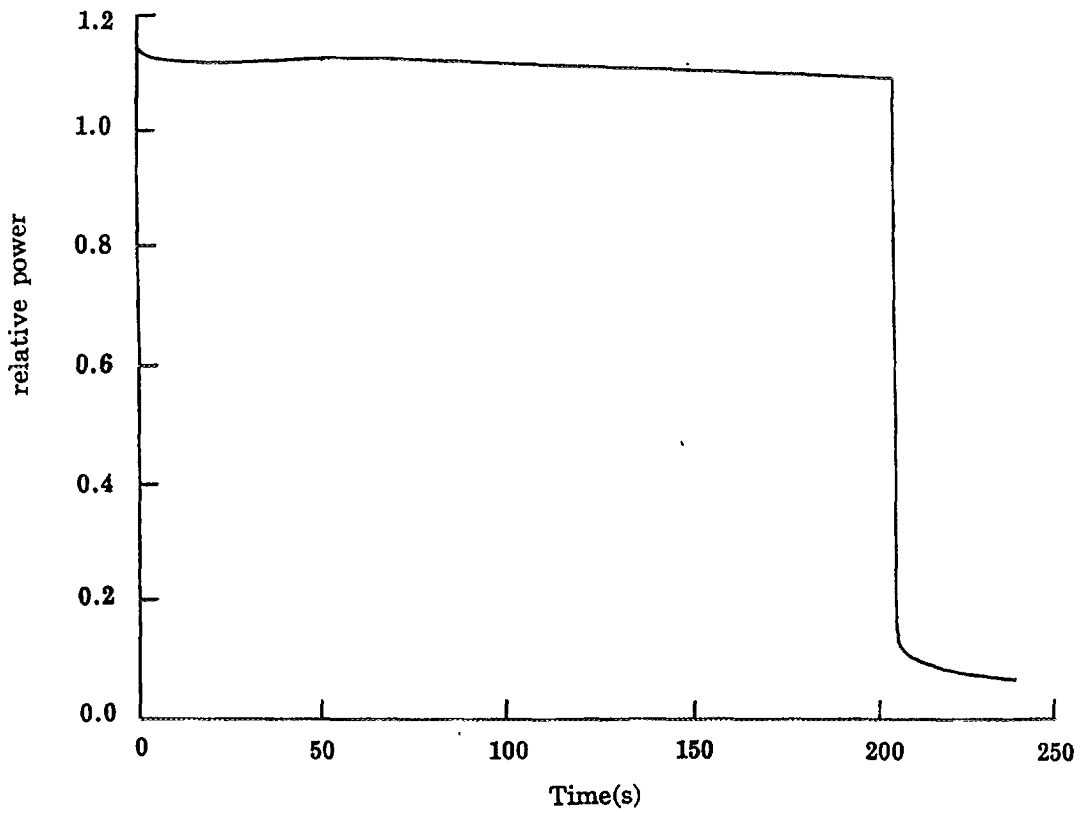

Fig.4 A compensate control rod rising for two steps at once

4.3. A compensate control rod rising two steps at once

Assuming that a compensating control rod with the maximum reactivity worth rises by two steps at once by error, the reactivity of 0.724×10^{-3} would be inserted in 0.2 seconds. It is a typical anomalous reactivity and power distribution accident, and it is considered as a Class II accident.

4.3.1. Assumed accident conditions

1) A compensate control rod with the maximum worth rises by two steps at once,
2) Reactor scram,
3) Isolation of all intermediate circuits after scram,
4) One residual heat removal system in operation.

4.3.2. Accident transient and sequence

At 1.0 second after the accident the reactor power reaches its peak, 238.6 MW, but it is still lower than the high power protection set point of 258MW. At about 202 seconds after the accident the high primary coolant pressure signal triggers the reactor protection system and the reactor is shutdown. A two seconds signal transition delay is taken into account in the accident analysis. In the accident transient the maximum nuclear power is 119% of rated power. The minimum DNBR is 1.98 and the maximum pellet temperature is 1627°C. All parameters satisfy the criteria of the safety limits. Fig 4 shows the nuclear power transient of the accident.

4.4. Loss of external power ATWS accident without boron injection

4.4.1. Assumed accident conditions

1) Loss of external power supply. It leads all circulation pumps to trip, and the primary system loses its heat sink,
2) All control rods are assumed stuck (ATWS),
3) No boron injection takes place,
4) One residual heat removal system is in operation.

The above assumptions are extremely conservative.

4.4.2. Accident transient and sequence

After loss of power, all circulation pumps (intermediate and tertiary circuits) trip at once. The reactor loses its main heat sink, so heat accumulates first in both the main heat exchanger and the downcomer. This leads the coolant temperature in the downcomer to rise, and the natural driving head goes down. Consequently, primary coolant flow rate goes down, core temperature and primary coolant pressure rise. Reactor power goes down because of the negative moderator and fuel temperature coefficient.

At 143 seconds, the primary coolant pressure reaches the set point of the safety valve which opens in two seconds. Steam passes through the valve to the discharge tank at a flow rate of 3.04 kg/s and removes the heat accumulated in the reactor. The primary coolant pressure falls, and at 500 second it reaches 2.6 MPa; the valve closes. Then the heat accumulates again. Finally the heat generated in the core becomes as much as that removed

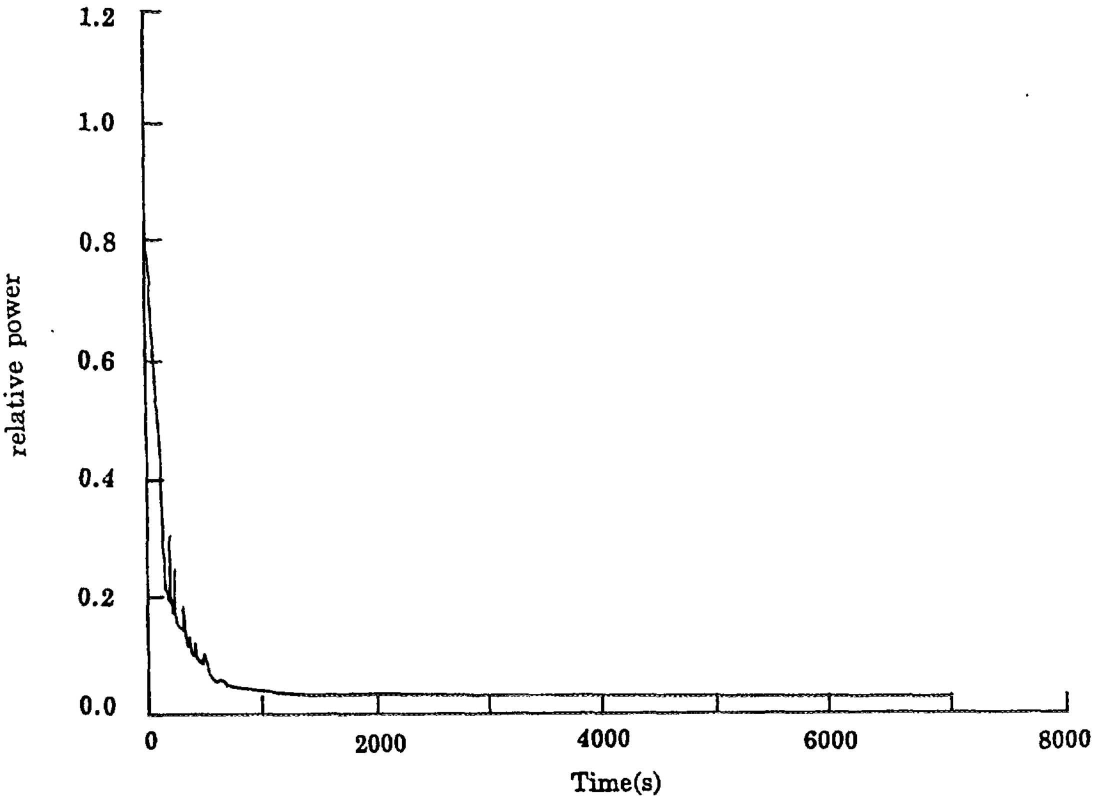

Fig.5 Loss of external power ATWS without boron injection

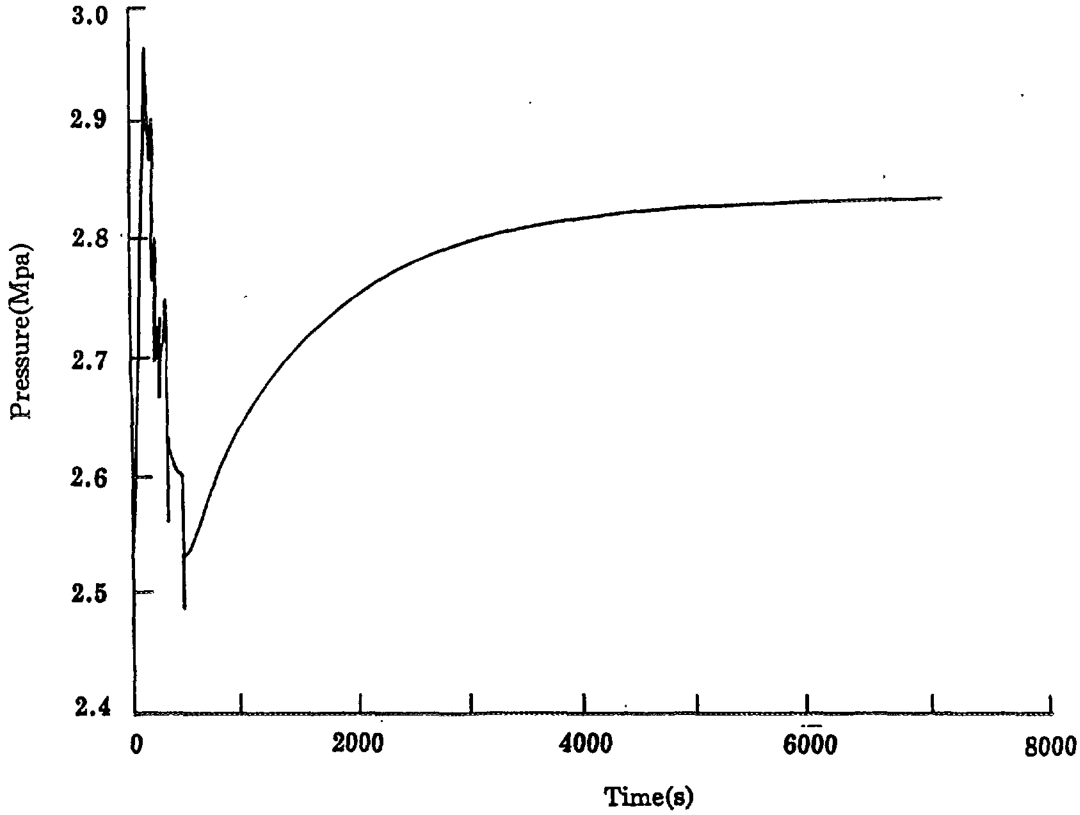

Fig.6 Loss of external power ATWS without boron injection
pressure in the RPV

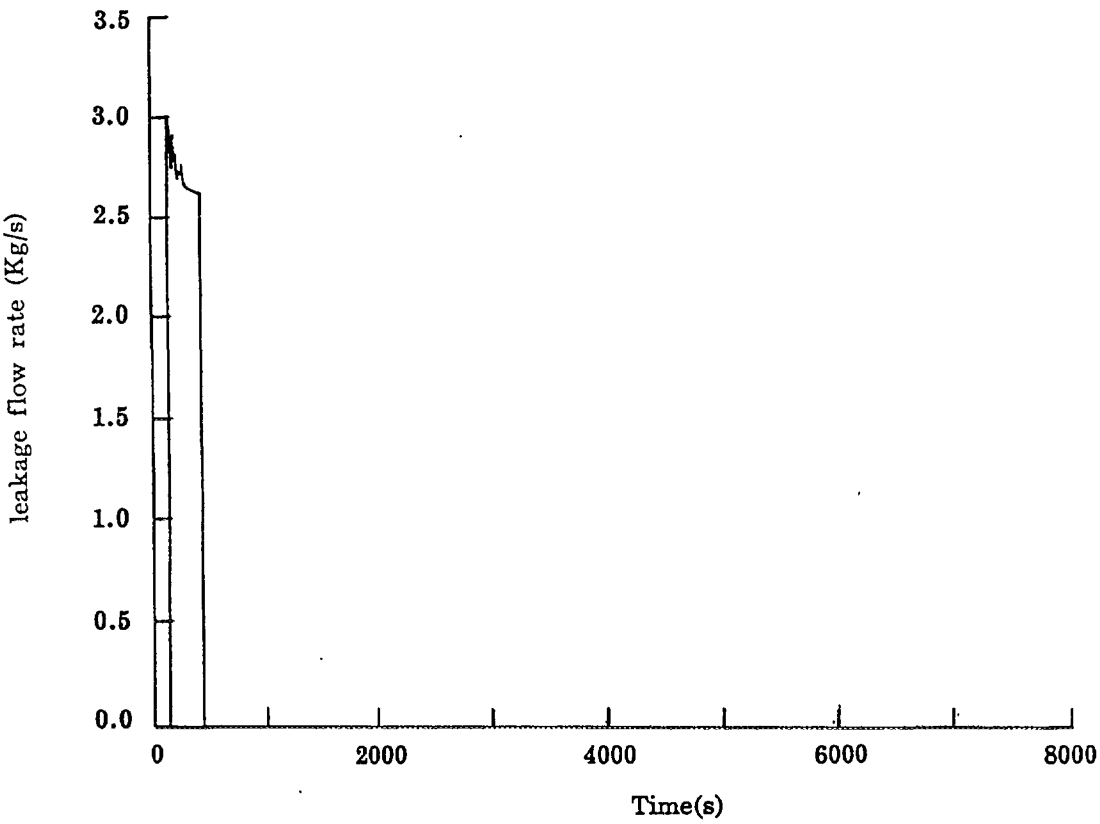

Fig.7 Loss of external power ATWS without boron injection

by the residual heat removal system. The primary temperature and pressure remain constant, and the safety valve opens no more. The system reaches a new stable state with a low power of 5.8 MW, 2.9% of rated power. In the accident only 942 kg of coolant is lost. The minimum DNBR is larger than 2.4 and the maximum fuel pellet temperature is lower than 1483°C.

Figs. 5 and 7 give the main transient parameters in the accident. The analysis results show that there is no fuel element burnout even under such serious accident conditions.

4.5. Pressure vessel cracking at bottom

Pressure vessel cracking is considered as a very low probability accident. But if it took place, coolant would rapidly discharge though the break in liquid phase, causing a great amount of coolant loss, so it is potentially an extremely serious accident. Conventionally the accident is arbitrarily considered to be impossible for nuclear power plants. In this paper, results are given only to demonstrate the safety characteristics of the NHR-200 under such an extremely serious accident.

4.5.1. Assumed accident conditions

1) Pressure vessel cracking at bottom, equivalent break size is of 1 cm^2;
2) Reactor scram;
3) Intermediate circuit isolation (two intermediate circuits are isolated at once);
4) Only one residual heat removal system in operation.

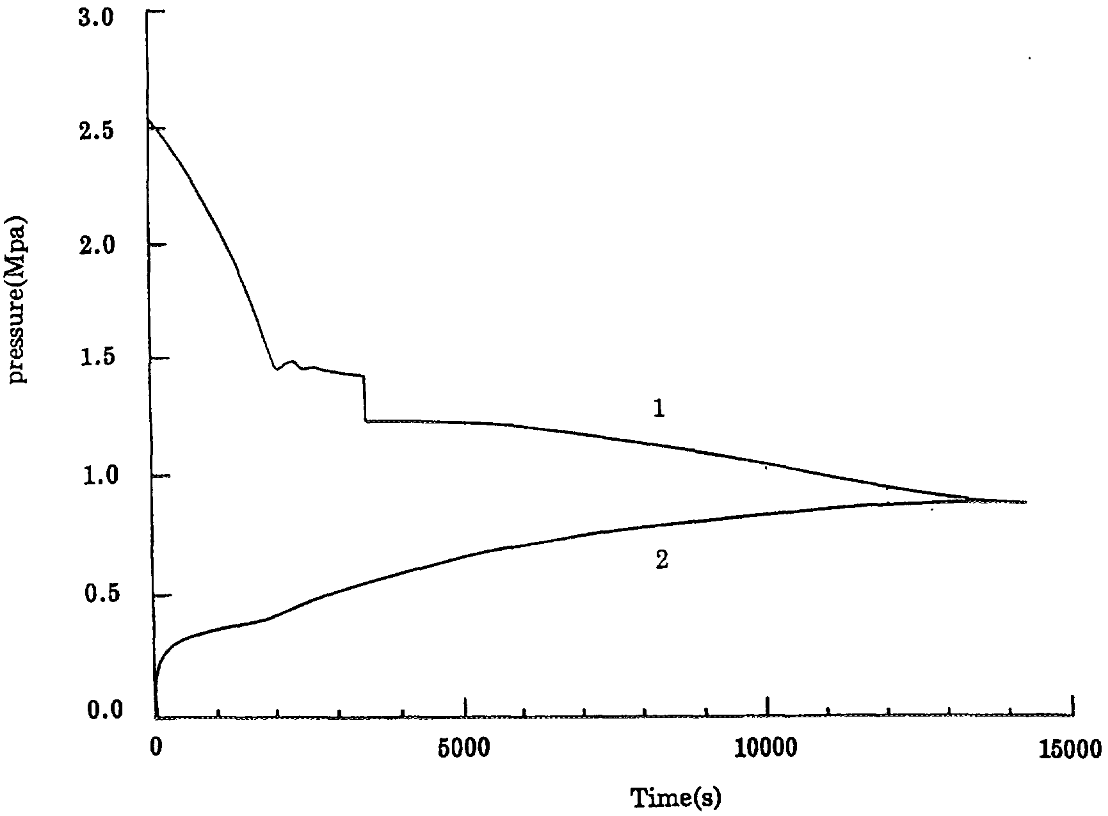

Fig.8 pressure vessel cracking at bottom
1. pressure in the RPV
2. pressure in the containment

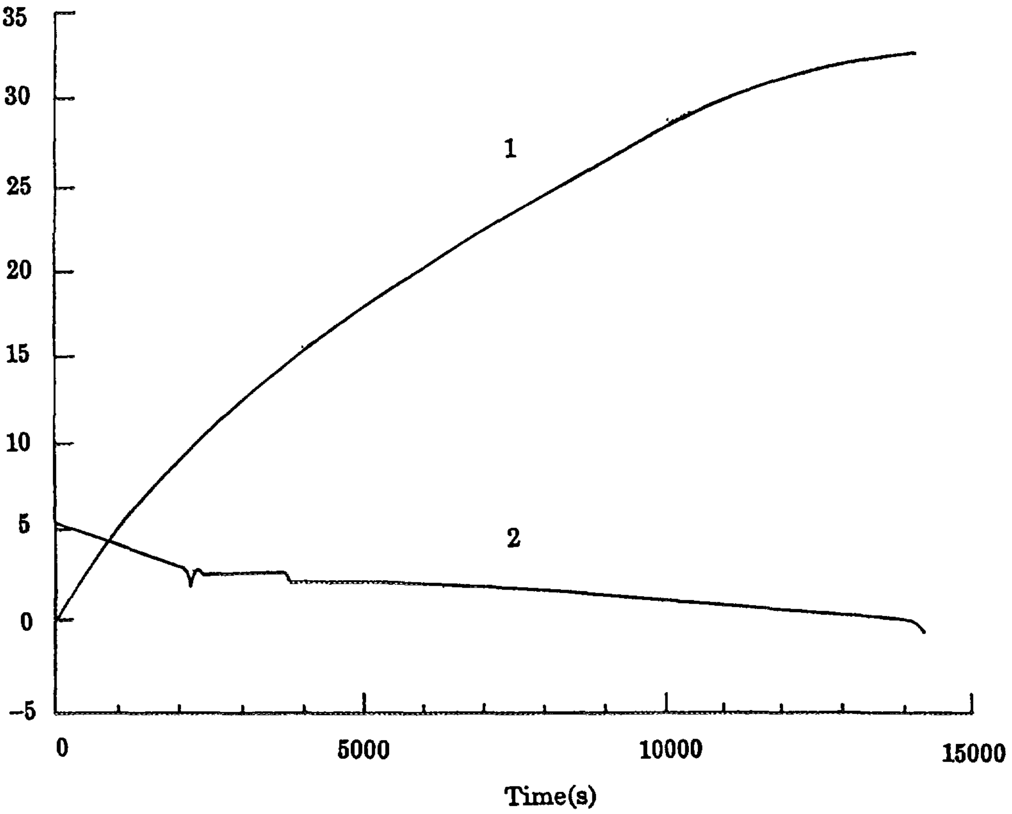

Fig.9 pressure vessel cracking at bottom
1. total loss of water (t)
2. leakage flow rate (Kg/s)

4.5.2. Accident transient and sequence

After cracking, primary coolant discharge flow is about 5.67 kg/s. Because of the double vessel configuration feature, the discharged coolant is contained by the containment which nestles the pressure vessel inside. Following the coolant discharge, the primary coolant pressure goes down and the pressure in the containment rises. At 560 second after the accident, the low liquid level reaches its protection value. At 757.8 second the high pressure in the containment trips the protection system, 2 seconds later, the control rods insert into the core and cause the reactor to shut down. The residual heat removal system operates 20 seconds later. At about 10600 seconds after the accident the coolant discharge ends, the primary coolant level is still above the top of the reactor core. The pressures in the pressure vessel and in the containment are 0.9 MPa. At the end of the accident the total amount of coolant discharged from the pressure vessel is 32.9t. There is still 67.1t of water that covers the reactor core.

During the accident, the minimum DNBR is 2.2, the maximum central temperature of the fuel is 1467°C. Both values are lower than the limits. Figs. 8 and 9 show the main parameters during the accident.

5. CONCLUSION

Based on the design, construction and operation experience of the 5 MW Experimental Heating Reactor NHR-5, the design features of the NHR-200 guarantee its safety. The accident analyses have predicted the excellent safety features of the NHR-200. No fuel element burn-out could take place even in the extremely serious accidents of very low probability.

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DISTRICT HEATING GRID OF THE DAQING NUCLEAR HEATING PLANT

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Abstract

The Daqing Nuclear Heating Plant is the first commercial heating plant to be built in China. The plant is planned to be used as the main heat resource of one residential quarter of Daqing city. The main parameters of the heating plant are summarized in the paper. The load curve shows that the capacity of the NHP is about 69% of total capacity of the grid. The 12 existing boilers can be used as reserve and peak load heat resources. Two patterns of load following have been considered and tested on the 5MW Test Heating Reactor. Experiment shows load of heat grid is changed slowly, so automatic load following is not necessary.

1. GRID GENERAL DESCRIPTION

The Daqing Nuclear Heating Plant is planned to be used as the main heat resource for one residential quarter of Daqing city. The distance between the plant and the boundary of the residential quarter is about 2.5 km. Thermal power of the plant is 200 MW. In winter, the plant will supply heat for 3.8 million m^2 houses and buildings. In summer it will provide some hot water. The heating grid (main heating piping) is shown in Fig.1.

The total heat power of the grid is about 350MW. The temperature of hot water going into the houses is ~ 95°C. The principal systems of the plant are shown in Fig.2. The heating system consists of 4 water loops. The first loop is the reactor coolant circulating loop with the temperatures of primary coolant being 210/145°C and the pressure 2.5 MPa. The second loop is an intermediate loop with temperatures of 145/85°C and a pressure of 3.0 MPa. The third loop is the main heat grid with temperatures of 130/80°C and a pressure of 1.0 MPa. The fourth loop is a branch heating grid with temperatures of 95/70°C. The main parameters of these loops are shown in Table 1.

2. HEAT GRID CAPACITY AND LOAD

There are 12 existing boilers. After the nuclear heating plant has been built they can be used as reserve and peak load heat resources. The total capacity of these boilers is 350 MW. Some of them may be dismantled or removed to other regions. According to Chinese rules, the capacity of the maximum heat plant should not be more than 70% of the capacity of heat grid. The maximum load of the grid is 312 MW. The peak boiler capacity is more than 123 MW.

Load curves of the grid are shown in Fig.3 and Fig.4. The shaded part on the Fig.3 shows the peak load taken by the peak boilers.

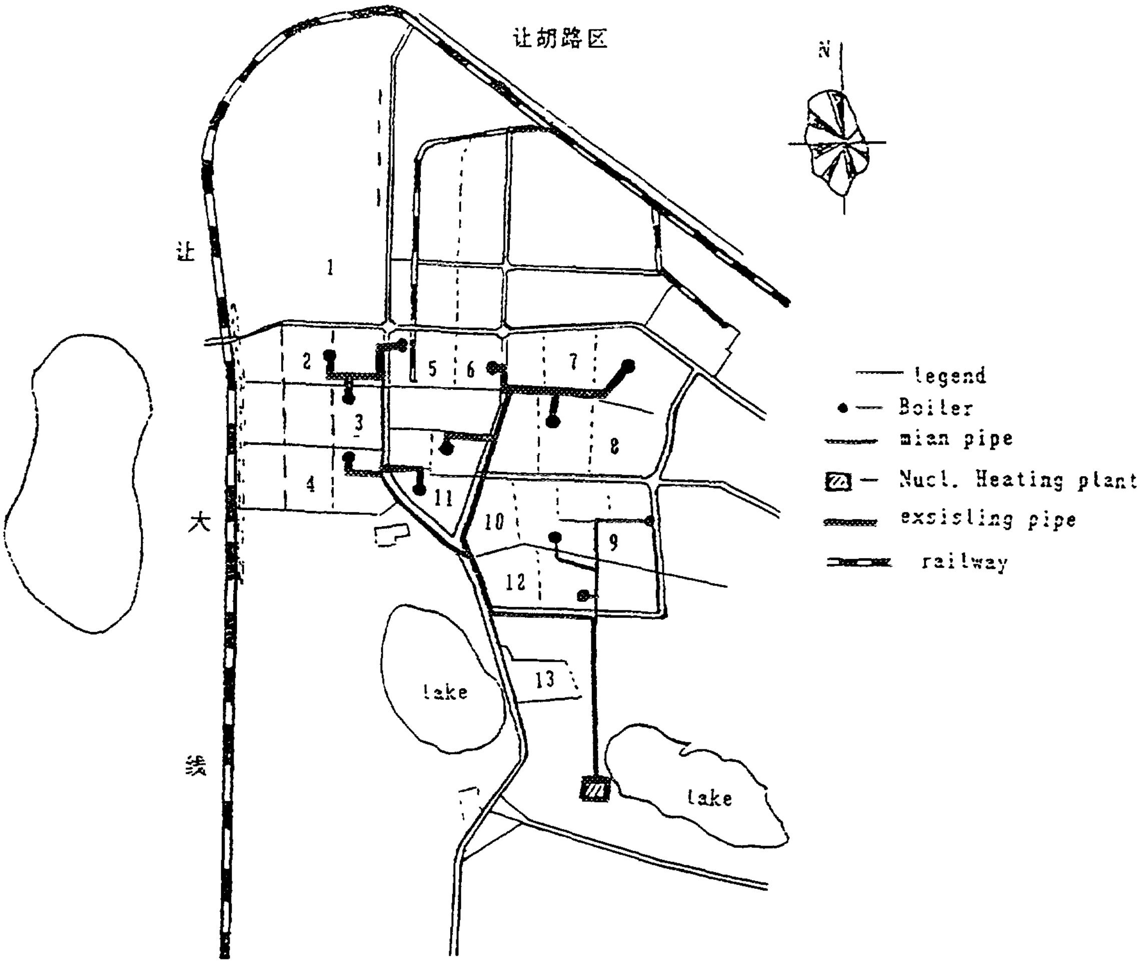
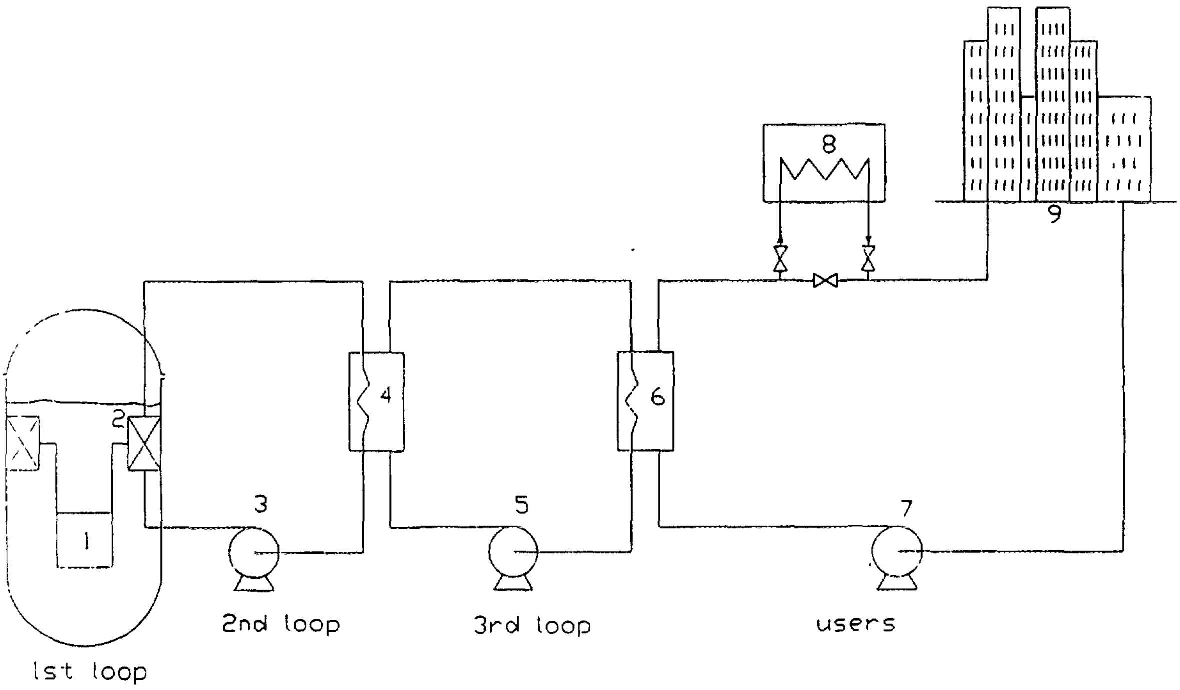


Fig.1 Daqing heat grid



1 reactor core 2 main heat exchanger 3,5,7 pumps
4,6 heat exchangers 8 peak load heater source 9 user

Fig.2 Principal system of Daqing heat grid

TABLE 1: MAIN PARAMETERS OF THE HEATING SYSTEM

	Temperature max.	Temperature min.	Pressure MPa
First loop (reactor coolant)	210	145	2.5
Second loop (intermediate loop)	145	85	3.0
3rd loop (main grid)	130	80	~1.0
4th loop (branch grid)	95	70	~0.3

3. LOAD FOLLOWING AND REACTOR CONTROL

3.1. Control pattern

As far as reactor control is concerned, the target is to keep the outlet water temperature from the core constant. Consequently, the pressure of the reactor will essentially be constant also.

As to heat grid control, in China two types of regulation are used: quantity regulation and quality regulation. In the case of quantity regulation, heating water distribution is unchanged during the whole heating season. It does not need control flow rate for every user. So the original investment of the grid is smaller. But the heating quality is poor. It is used for some small heat grids.

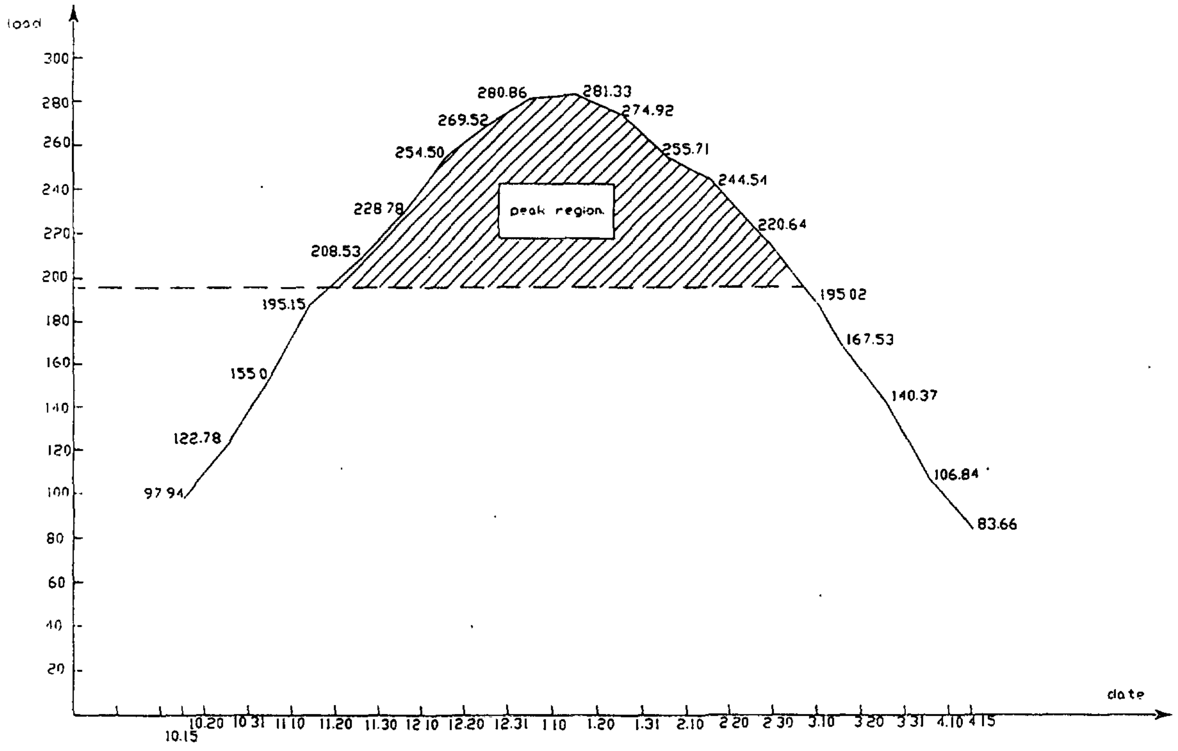


Fig.3 Average load curve

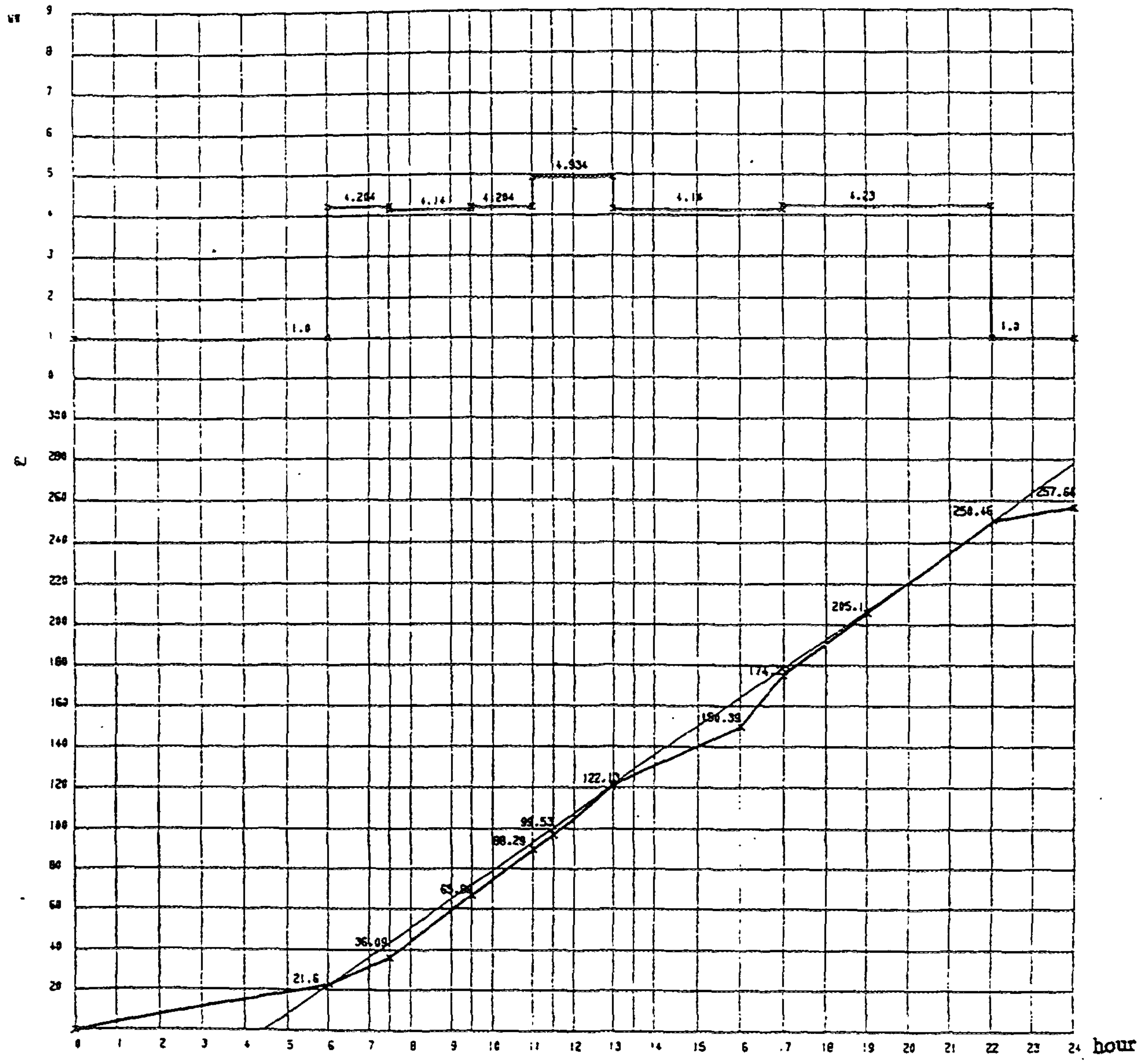


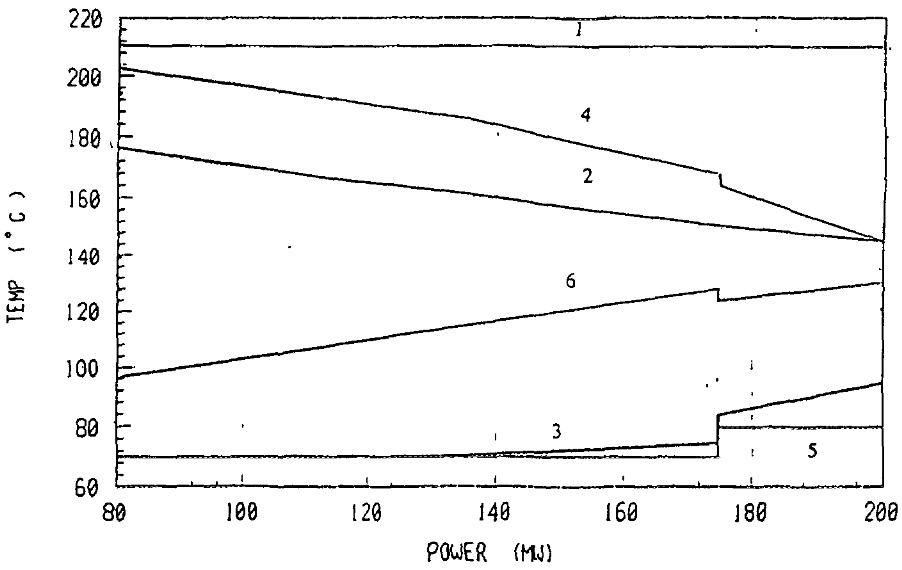
Fig.4 Typical daily heat load curve (in winter)

In the case of quality regulation, heating water supply is controlled by automatical instruments, so the flow rate of the heat grid is changed continually. Because of high control quality, it is widely used in big heat grids. For the Daqing city, heat grid is not a big one, so a combination of quantity and quality regulation (stepped quality regulation) is adopted. That means when the heat load is less than 75% of the maximum load, the heating water flow rate will be adjusted to 75% of maximum flow rate step by step (Fig.5 and 6).

For the intermediate loop a quantity regulation is adopted. The circulation pump is driven by a motor with frequence converter, allowing the necessary flow changes.

3.2. Loading following

Because that the grid has a great heat capacity, room temperature changes slowly. Heat load can't be followed with the room temperature. There are two patterns of load



1 core outlet T 2 core inlet T 3 heat exchanger inlet T
4 heat exchanger outlet T 5, 6 heat grid T

Fig.5 Dependence of temperature on the heat load

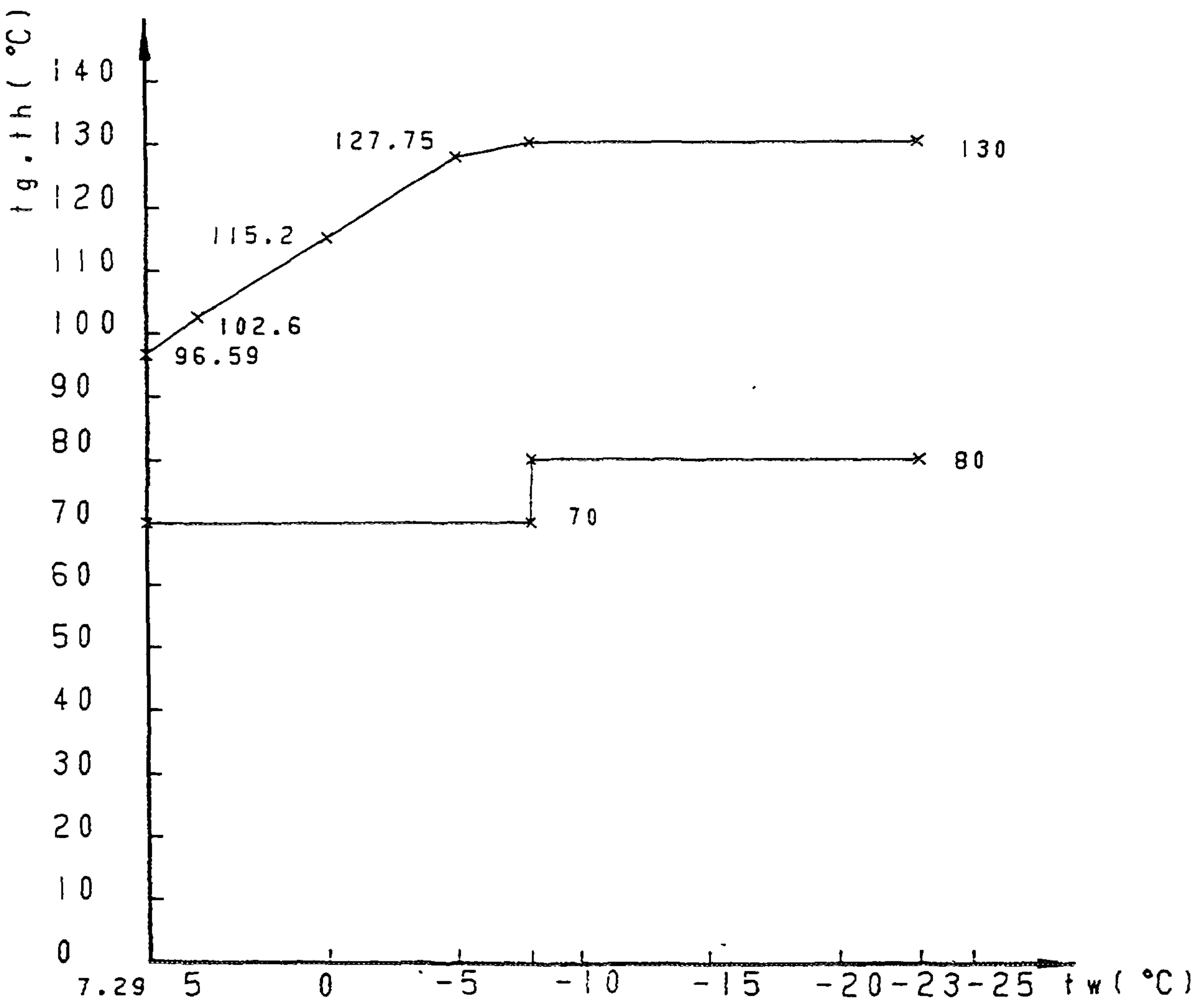


Fig.6 Dependence of heat grid water temp. on the heat load

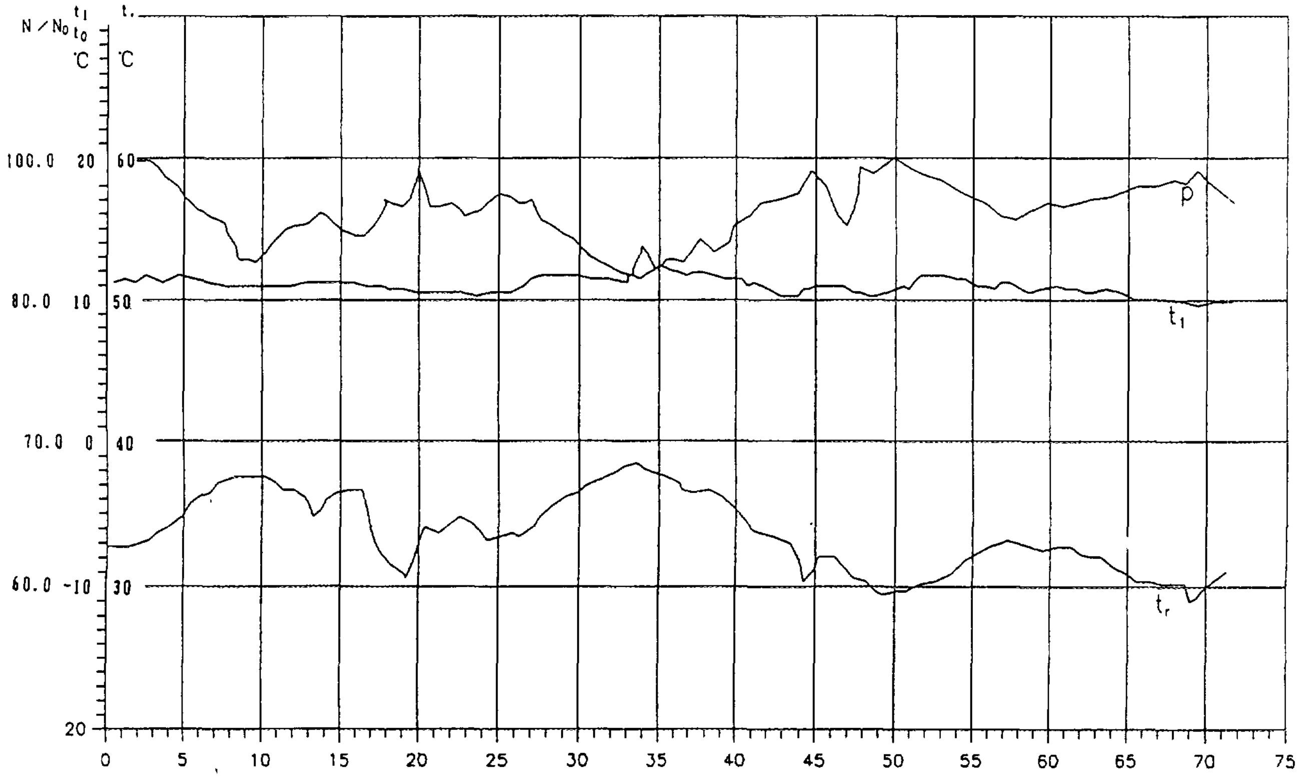


Fig.7 Load following character 1

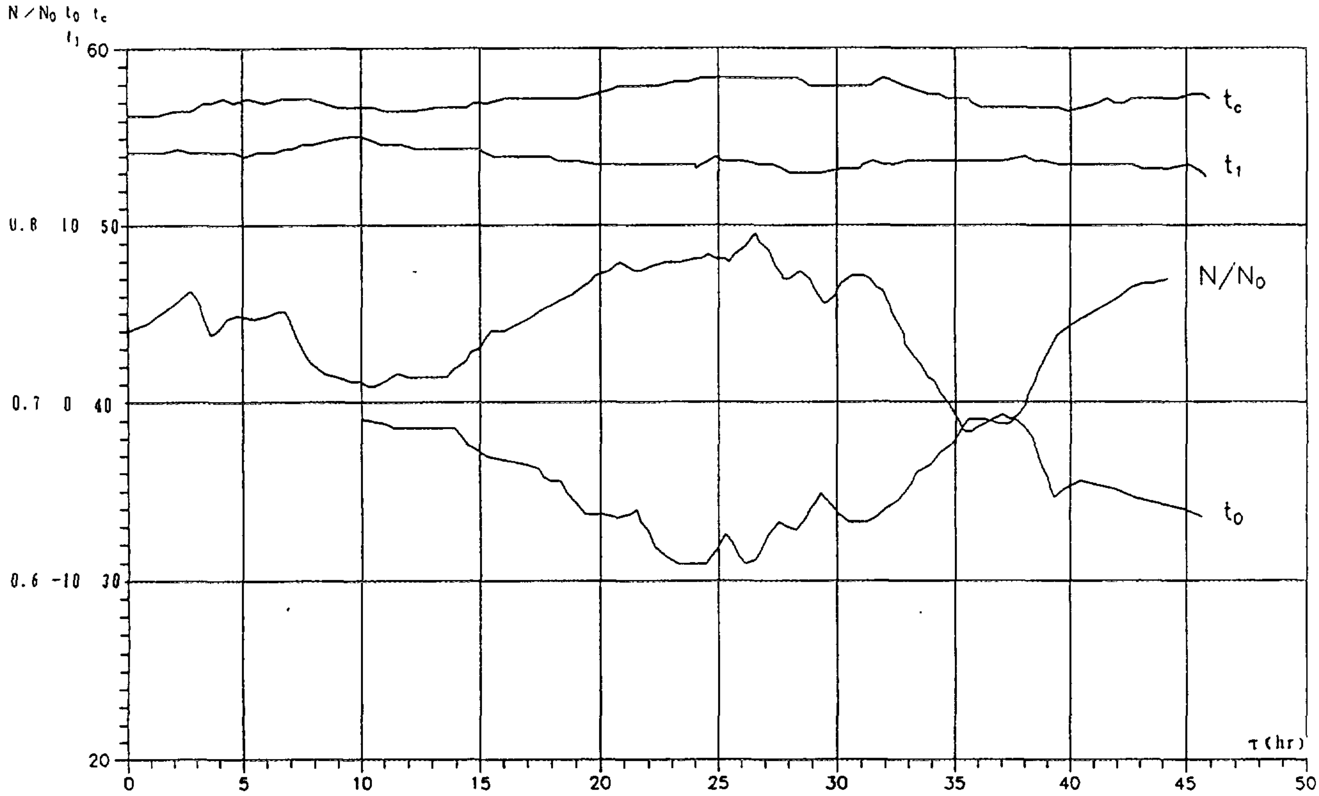


Fig.8 Load following character 2

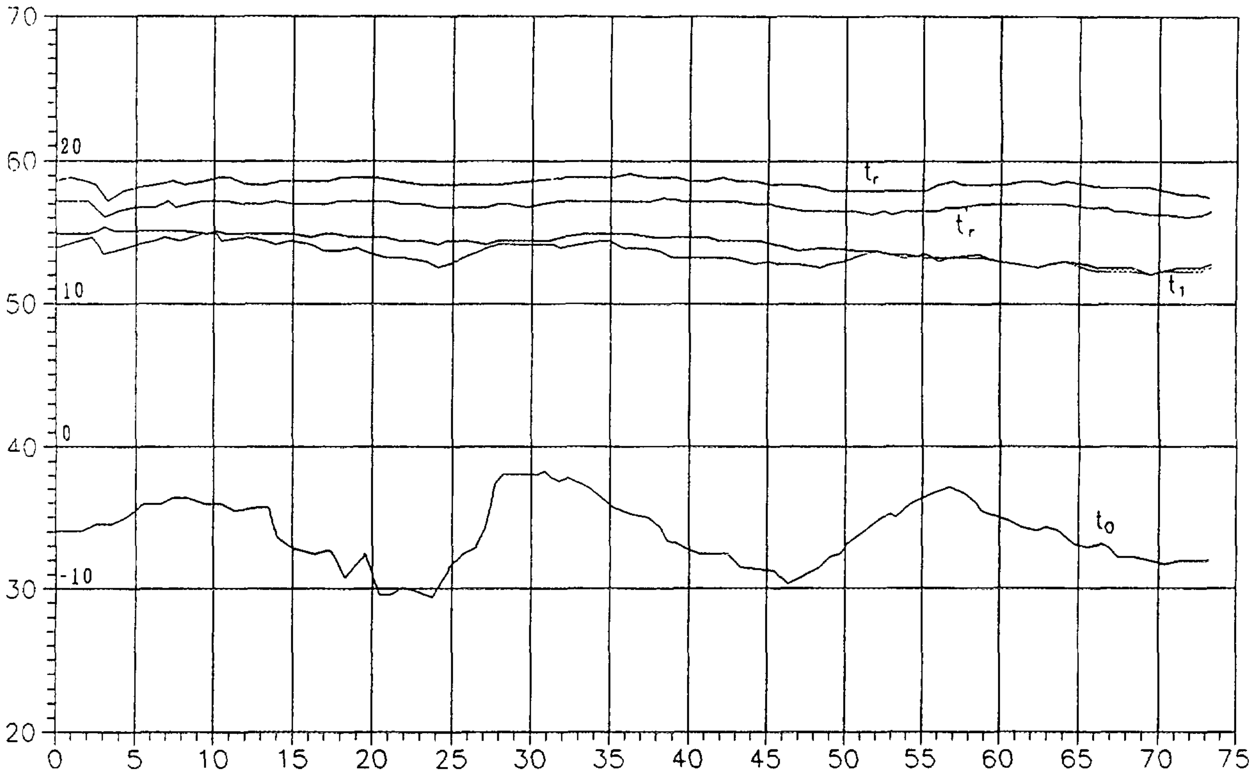


Fig.9 Load following characte 3

following that have been considered. First, automatic load following keeping outlet temperature of the reactor core constant. In this case the control rods must be regulated continuously to fit the change of the environmental temperature. The other mode is load following by using the temperature coefficient of the reactor. In this case the outlet temperature of the reactor core cannot be constant. It will change with the environmental temperature.

Several experiments have been done on the 5MW Test Nuclear Heating Reactor to model these two patterns of load following.

In the first group of experiments the control rods were regulated to follow the environmental temperature. That means when the environmental temperature decreased, the reactor power will be increased. Typical experimental results are shown in the Fig.7 and Fig.8. It shows that the room temperature can be kept very well. The changes of the room temperature were negligible. But the control rods moved very frequently. From the view point of operation frequent moving control rods needs to be avoided.

In the second group of experiments position of the control rods was not changed during a day. When the environmental temperature changed, due to the negative temperature reaction coefficient of the reactor, the reactor power is changed automatically to fit the change of the environmental temperature. The experiment shows that in this case room temperature and outlet temperature of the core had some changes, but these changes are acceptable. Typical results are shown in Fig.9.

Based on above experiments, it is recommended to adopt the temperature coefficient automatic regulation to follow the load of the heat grid. Moving the control rods is needed only when the change of core outlet temperature exceeded a operation acceptable range.

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COUPLING OF AST-500 HEATING REACTORS WITH DESALINATION FACILITIES

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Abstract

The general issues regarding the joint operation of a NHR and a desalination facility for potable water production are briefly considered. The AST-500 reactor plant and the DOUGTPA-type evaporating desalination facilities, both relying on proven technology and solid experience of construction and operation, are taken as a basis for the design of a large-output nuclear desalination complex. Its main design characteristics are given. The similarity of NHR operation for heating grid and desalination facility in respect of reactor plant operating conditions and power regulation principles is pointed out. The issues of nuclear desalination complexes composition are discussed briefly as well.

1. INTRODUCTION

A low-parameter nuclear heating reactor is capable to be used effectively as a part of water desalination complexes taking into account their distinctive features such as:

- high reliability of the heat generation;
- extended refuelling interval (2 years);
- identity of NHR operating conditions both for a heating grid and for a desalination facility;
- enhanced radiological safety which allows:
- to construct the nuclear power plant close to a sea shore or any other sources of salted water used for a desalination process;
- to create an integrated complex incorporating both desalination facilities with desalted water stocks and a nuclear power plant;
- to deploy such complexes close to fresh water consumers, e.g. industrial-residential centers [1].

2. DESALINATION COMPLEX COMPOSITION

The AST-reactor is designed to produce low parameter heat, which defines the type of an appropriate desalination facility. Direct usage of the low-parameter heat is most effective in evaporating facilities. Large operation experience is available in Russia for desalination facilities with horizontal film evaporators. Particularly, there are more than 20 years of experience with such plants in operation at the industrial desalination complex with the BN-350 fast nuclear reactor (also developed by OKBM) on the Caspian sea shore. Desalination plants of such type are characterized by the good quality of distilled water produced, by a low power consumption and by a long-term (more than 25 years) service life of the basic equipment.

According to IAEA data, the rate of expansion of desalination facilities with horizontal-tube film evaporators was the highest for the last 10 years, compared with desalination plants of other types. This can be explained by their mass production mastering, better economics, e.g. relatively low (in comparison with the reverse osmosis facilities) operational cost and by a lower price of desalinated water, as well as by higher quality of potable water (data of IDE Technology LTD., Israel).

The creation of desalination complexes incorporating AST-500 power units of some 200,000 m^3 per day output in one unit (Fig.1), and even more, presumes availability of large consumers of desalinated water in the region, as well as of a developed system for water supply and distribution (pipelines, pump stations, etc.) and developed power systems. Under these conditions there are grounds to energize the desalination complex from an external power system, while the reactor is operated only for heat supply to the desalination facilities. This simplifies the operation of the complex and improves the efficiency of the usage of the generated heat.

However, for regions with weakly developed power grids and high tariffs for electricity, there is a possibility to realize a desalination complex on the basis of an AST-500 reactor powered from its own auxiliary turbogenerator. A turbine building, related electrical equipment etc. should be additionally included in the plant. In this case the desalination complex output is decreased by appr. 20% due to the redistribution of thermal power generated by the reactor between the turbine plant and the desalination facility.

The number of power units in the desalination complex is determined by the specified output of desalinated water with account of its maximum consumption and load variation over the seasons, and by the availability of other sources of fresh water in the given region. The water output should not be less than a constant portion of the annual load curve determined by the residential and industrial water consumption with account of planned outage of one power unit for refuelling. The reactor refuelling should be carried out in the period of minimum load in water consumption, if this factor is to be considered (e.g. regions with irrigated agriculture).

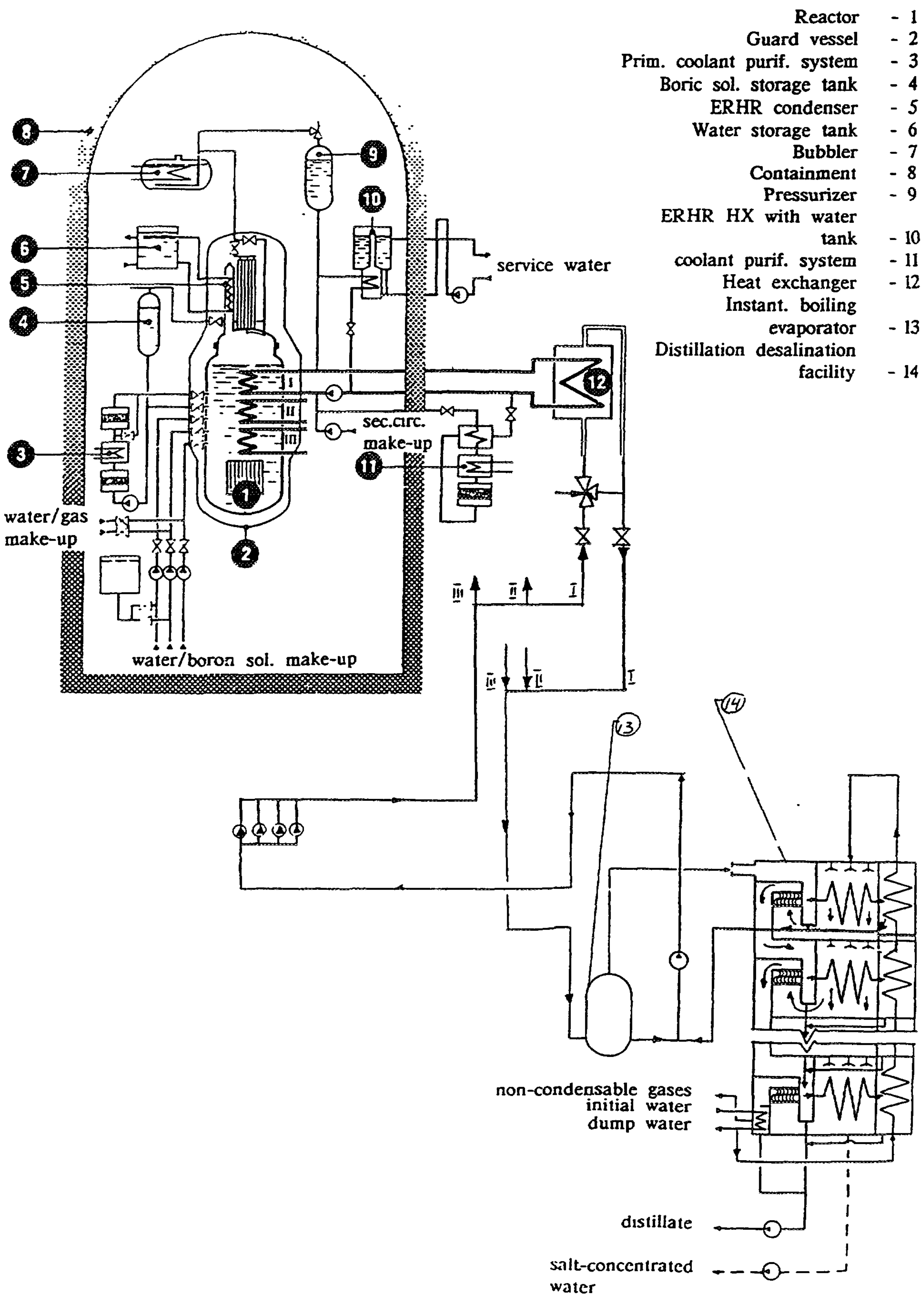
3. PLANT OPERATION CONTROL CONCEPT

When considering nuclear desalination plant control principles, the similarity should be noted with an NHR operating for a heating grid (load variation over seasons) and for hot water supply (daytime load variation). But in a given case there is a wide possibility to flatten the influence of load variations onto the nuclear reactor because of the capability to accumulate or consume desalinated water stored in water tanks following a daytime reduction, or a rise in water consumption, respectively. Besides, there is a possibility to divide the peaks of industrial and agricultural loads (e.g. fields irrigation during night hours, if possible).

Thanks to the availability of desalinated water stocks, the continuous supply of fresh water to consumers can be provided also in case of a short-term unplanned outage of the desalination unit. Taking account the heat rate limit specified for the AST-500 reactor start-up mode, the minimum period of interruption in the desalinated water production process for this unit amounts to appr. 24 hours.

In general, the choice of the method for heat output regulation to either the base-load or load-following principle (or their combination) is dictated by the specific conditions and

AST-500 NHR-based Desalination Facility Principal Flow Diagram



requirements of the user. The actual and prospective loads of industrial, household and agricultural consumers of water are taken into account as well as their load variation curves, availability of other sources of fresh water and heat. The range of heating water temperature variation at the desalination facility inlet, etc. have also to be taken into account. The AST-500 reactor ensures the reliable operation of the complex both in base-load and load-following operation modes with a reactor power variation rate limit of 25% N_{nom} per hour.

4. DESIGN STUDY FOR AST-500 INTEGRATION INTO DESALINATION COMPLEX

A design study was carried out by the OKBM together with other design Institutions (SverdNIIChimMash, NIAEP, Kurchatov Institute, etc.) aiming at the development of a desalination complex which is composed of several (more than two) autonomous units, each comprising a AST-500 reactor plant and a desalination facility [2]. The number of desalination apparatus in the facility which are working in parallel is defined, taking into account the possibility to disconnect a part of them for preventive maintenance and repair because the maintenance interval may not coincide with the reactor refuelling one. The level of redundancy accounts also for the necessity of the annual maintenance to be performed for the desalination apparatus (up to 20 days), as well as a probability of an additional failure of one of them.

The main technical characteristics of a nuclear desalination unit based on the AST-500 reactor (without turbogenerator plant) are given below.

TABLE 1: DESALINATION PLANT MAIN TECHNICAL DATA

	Nuclear reactor plant	
1.	Reactor thermal power, MW	400
2.	Primary circuit pressure, MPa	2
3.	Primary circuit temperature, °C	208/141
4.	Secondary circuit pressure, MPa	1.2
5.	Secondary circuit temperature, °C	160/102
6.	Tertiary circuit pressure, MPa	2
7.	Tertiary circuit temperature, °C	130/98
8.	Auxiliary power (off-site power source), MW	appr. 7
9.	Auxiliary heat consumption, MW	appr. 15
	Desalination facility	evaporation of the DOUGTPA-700 type apparatus with horizontal tube-film evaporators
10.	Number of evaporation apparatus	
	total	16
	operating	13
	stand-by	3
11.	Auxiliary power (off-site power source), MW	appr. 12

12.	Nominal output, m^3/day	appr. 220,000 (16,800x13)
13.	Sea water max. boiling temperature in the first stage of each apparatus, °C	<115
14.	Salt concentration in desalinated water, mg/l	<20

The uprating potential available for the advanced AST-500M reactor allows to additionally rise the desalination facility output by appr. 15%.

The nuclear desalination complex principal flow diagram is given in Fig.1.

The three-loop flow diagram being traditional for the AST-500 secondary and tertiary circuits is retained here along with a possibility for disconnecting one loop from the consumer.

In the desalination complex, as well as in the AST-500, the tertiary circuit loops are joined by "hot" and "cold" headers. Circulating pumps similar to the grid ones in the AST-500 provide heat transport from the reactor plant to the consumers and the desalination apparatus with water recirculation through a closed loop. The heating medium can be supplied to the first stage of the desalination apparatus as water or steam produced from this water in an instantaneous boiling facility. Both methods are verified in the operating nuclear desalination complex with the BN-350 nuclear reactor.

At plant operation under automatic control, the reactor power and the parameters of the circuits are established and maintained by the reactor control rods and by three-way valves in the loops of the tertiary (heating) circuit according to a control algorithm similar that adopted for the AST-500.

The reactor power variation range (10-100% N_{nom}) overlaps with a margin the range of stable operation of the desalination apparatus (30-100% G_{nom}, where G_{nom} is the nominal flow of desalinated water).

When determining the volume of the stand-by desalinated water storage tanks it is expedient to take account of the power variation rate admissible for the reactor at planned variations of desalinated water consumption.

Transients associated with a complete loss of load cause the reactor to shutdown by emergency protection signals, and its cooling down by the normal or emergency heat removal systems. Reactor operation with an incomplete number of heat transport loops is permitted in the range of 10-50% N_{nom} without time limitation if the design limits and safety conditions specified for the AST-500 are satisfied. The reactor transition to partial operation means a proportional decrease in desalination facility output by disconnection of several desalination apparatus or by a proportional decrease of each apparatus output.

5. CONCLUSION

The AST-500 NHR has become a reference reactor plant for the whole series of integral enhanced safety reactors that have been developed recently at OKBM. The basic principles and the engineering decisions realized in the reactor plant allow to use it effectively as a heat source for an evaporating desalination facility on the basis of the well proven apparatus of the DOUGTPA type with horizontal tube-film evaporators.

The rated output of this facility amounts to 220,000 m^3/day of desalinated water, with the potential for a 15% increase in the output.

The modes of AST-500 operation in combination with the desalination facility are similar to those for operation of a heating grid.

The high level of radiological safety intrinsic for the AST-500 reactor allows to site it close to water sources, to fresh water consumers and in the immediate proximity of a desalination facility, thus forming an integrated power-desalination complex.

The positive practical experience of Russia in the creation and operation of the sea water nuclear desalination system based on the BN-350 reactor and on the desalination apparatus of the DOUGTPA type, together with the proven technology of the AST-500 NHR allow to facilitate substantially the licensing and implementation of AST-based desalination complexes.

Their excellent safety and economic characteristics give grounds to consider them as rather prospective for the deployment in many regions worldwide suffering from shortage of potable water.

REFERENCES

[1] IAEA draft report on nuclear heating reactors utilization for sea water desalination, 1990.

[2] Report on investigation for OKBM-designed reactor plants for desalination complexes, - OKBM, 1992, (in Russian).

FIVE MW NUCLEAR HEATING REACTOR

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Abstract

The 5 MW Nuclear Heating Reactor (NHR-5) developed and designed by the Institute of Nuclear Energy Technology (INET) has been operated for four winter seasons since 1989. During the time of commissioning and operation a number of experiments including self-stability, self-regulation, and simulation of ATWS etc. were carried out. Some operating experiences such as water chemistry, radiation protection and environmental impacts and so on were also obtained at the same time. All of these results demonstrate the design of the NHR-5 is successful.

1. INTRODUCTION

The 5MW Nuclear Heating Reactor (NHR-5) developed and designed by the Institute of Nuclear Energy Technology (INET) has been in operation for four winter seasons. The construction of NHR-5 began in March 1986, the civil engineering was completed in September 1987, and the erection of the NHR-5 was finished in April 1989. The initial criticality of the NHR-5 was reached in Nov. 1989 and full power operation began in Dec. of the same year.

In order to expand the utilization of the NHR and to improve its economic competitiveness, the operational experiments of cogeneration- heat and electricity- and refrigeration for air condition using nuclear steam from NHR-5 were carried out in 1992. The milestones of NHR-5 are listed in table 1.

TABLE 1: THE MILESTONES OF NHR-5

Beginning of construction	Mar. 1986
Completion of civil engineering	Sept. 1987
Completion of erection of reactor	Apr. 1989
Beginning of commissioning	May 7, 1989
Initial fuel loading	Oct. 9, 1989
Initial criticality	Nov. 3, 1989
Full power operation	Dec. 16, 1989

The operational practice shows that the NHR - 5 has excellent operating and safety features, and a high availability of 99%. The practice also shows that the NHR-5 is easy to start up and to be operated. The operating results demonstrated that the NHR-5 has fully reached the design requirements and meets the main design parameters. Table 2 gives the main operating parameters in comparison with the design values. In the table the operation temperature of the reactor inlet is higher than the design value, which shows that the reactor has a larger than estimated natural-circulation capability.

2. DESCRIPTION OF NHR-5

The NHR-5 is the first heating reactor in operation in the world. It is an integrated vessel type light water reactor cooled by natural circulation with self-pressurized performance.

TABLE 2: MAIN OPERATING PARAMETERS OF THE NHR-5

	Design value	Operation value
Reactor thermal power	5MW	5MW
Reactor		
Outlet temperature	186°C	186°C
Inlet temperature	146.6°C	151°C
Pressure	1.37MPa	1.37MPa
Intermediate circuit		
Primary heat exchanger		
Outlet temperature	142°C	144°C
Inlet temperature	102°C	100°C
Flow rate	107t/hr	97 t/hr
Intermediate heat exchanger		
Outlet temperature	75.2°C	80°C
Inlet temperature	142°C	144°C
Flow rate	64 t/hr	67 t/hr
Pressure	1.7MPa	1.7MPa
Heating grid		
Outlet temperature	90°C	84°C
Inlet temperature	60°C	56°C
Flow rate	143t/hr	152t/hr

2.1. Structures of NHR-5

Integral design and natural circulation

The core and main components of the primary circuit are housed within a reactor pressure vessel (RPV). The reactor core is located at the bottom of a hanging barrel; underneath the hanging barrel a secondary support is placed in the bottom of the vessel. There is a long riser above the core outlet to enhance the natural circulation capability. There are four primary heat exchangers in the downcomer between the riser and the vessel wall. The reactor core is cooled by natural circulation and the carried heat is transferred to the intermediate circuit via primary heat exchangers.

Dual pressure vessel

A dual pressure vessel is adopted in the design of the NHR-5. The reactor pressure vessel is designed for an operating pressure of 1.5MPa. Outside the RPV, a second metallic vessel containment is mounted. The design pressure is 1.5MPa at a temperature of 177°C. The gap between the RPV and the containment is very small. The location of all RPV penetrations are at a height of 2m above the core outlet and there are no large-bore piping.

All of these measures can avoid and mitigate serious consequences which result from loss of coolant accidents. If the RPV would develop a leak at its bottom the core can also be kept covered with water. Fig. 1 shows the reactor structure with dual vessel. The main technical parameters of the dual pressure vessel arrangement are listed in table 3.

TABLE 3: THE MAIN PARAMETERS OF DUAL PRESSURE VESSEL

Pressure vessel		
ID	m	1.8
Total height	m	6.5
Working pressure	MPa	1.5
Working temperature	°C	198
Lining thickness (Braze welding)	mm	~6
Thickness of wall	mm	90
Total weight	t	35
Containment		
ID	m	2.8
Total height	m	9.5
Thickness of wall	mm	20
Design temperature	°C	177
Design pressure	MPa	1.5
Material		16MnR
Weight	t	29

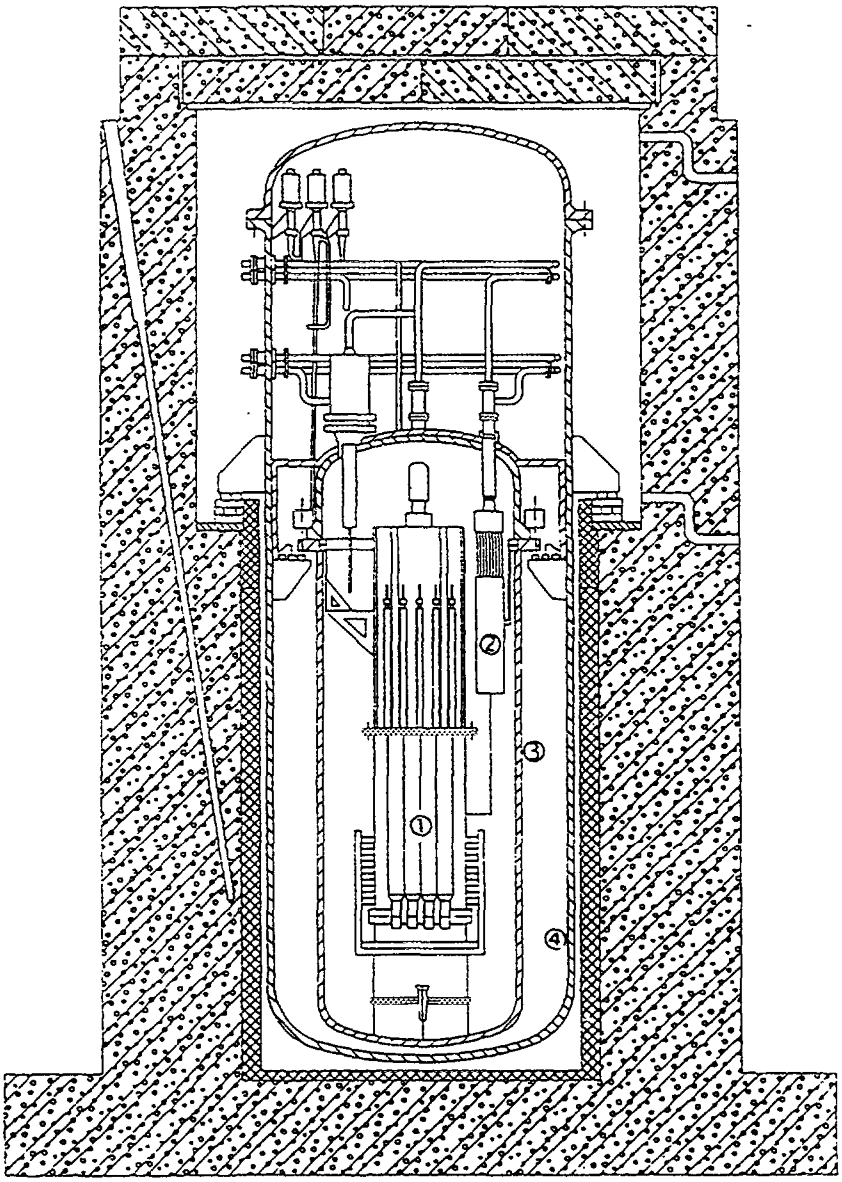
Self-pressurized system

A space above the coolant level inside the RPV acts as self-pressurizer. The pressure inside the RPV is depends on initial partial pressure of nitrogen and saturate vapor pressure corresponds to the core outlet temperature in the pressurized water operation mode. Due to the nitrogen partial pressure existing, the coolant can be kept subcooling in the core outlet. This is called pressurized water operation mode.

2.2. Overall arrangement of reactor core

Reactor core

The core cross section of NHR-5 is shown in Fig.2. In the core there are 12 fuel assemblies with 96 fuel rods, and 4 with 35 fuel rods. The fuel rod with a cladding of Zircaloy-4 has an active length of 690mm and a diameter of 10mm. The nuclear fuel is uranium dioxide with an enrichment of 3%. The total amount of UO_2 loaded in the core is 0.508 tons.

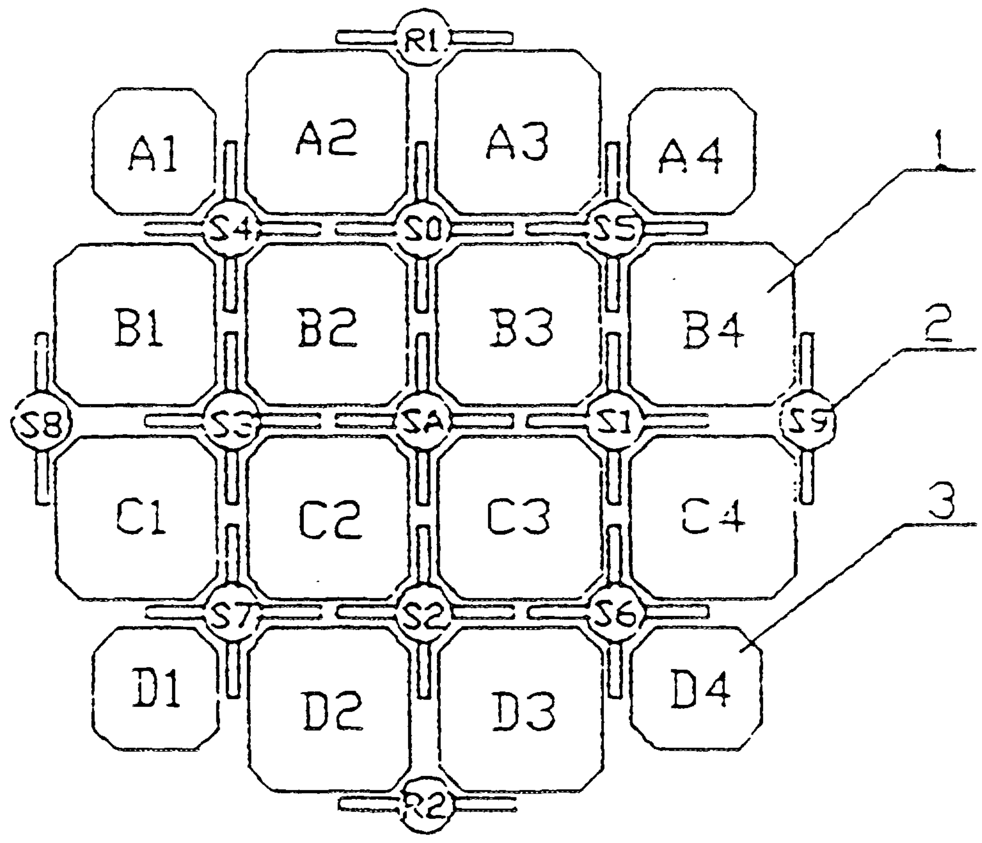


1. core 2. primary heat exchanger
3. reactor pressure vessel 4. containment

Fig.1 The NHR-5 structure with dual vessel

Control of reactivity

The reactivity is controlled by a combination of fuel rods containing fixed burnable poison of 1.5% Gd_2O_3, movable absorption rods (boron carbide) and the negative reactivity coefficient of the coolant. In the core there are 13 control rods which are all driven by a hydraulic driving system. The control rod drive system consists of three parts: an actuating loop outside the containment, 13 hydraulic step cylinders in the core and two control units (combined valves). The control rods can be dropped into the core by gravity when reactor shutdown is needed. Boron injection as a standby shutdown system is initiated by pumps or pressurized nitrogen during ATWS events.



1. assembly with 96 fuel rods
2. control rod
3. assembly with 35 fuel rods

Fig.2 The core cross section of NHR-5

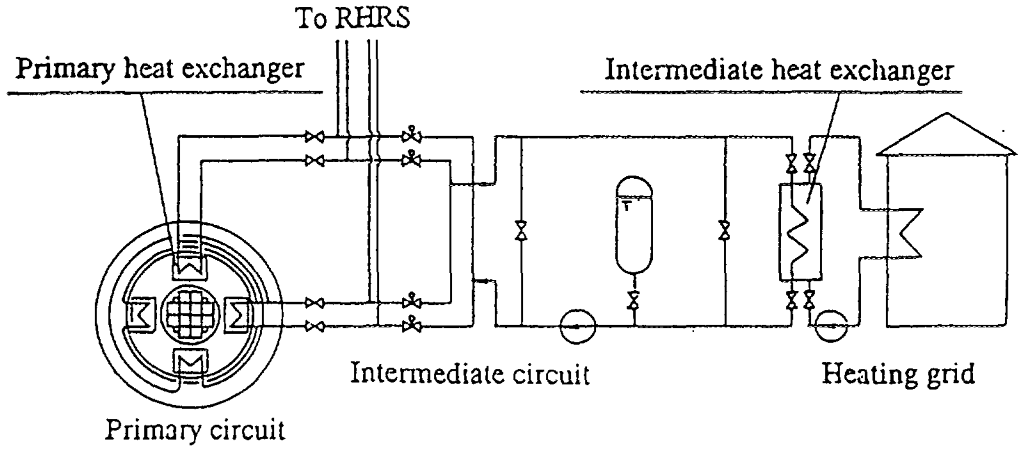


Fig.3 The main heat transfer system of NHR-5

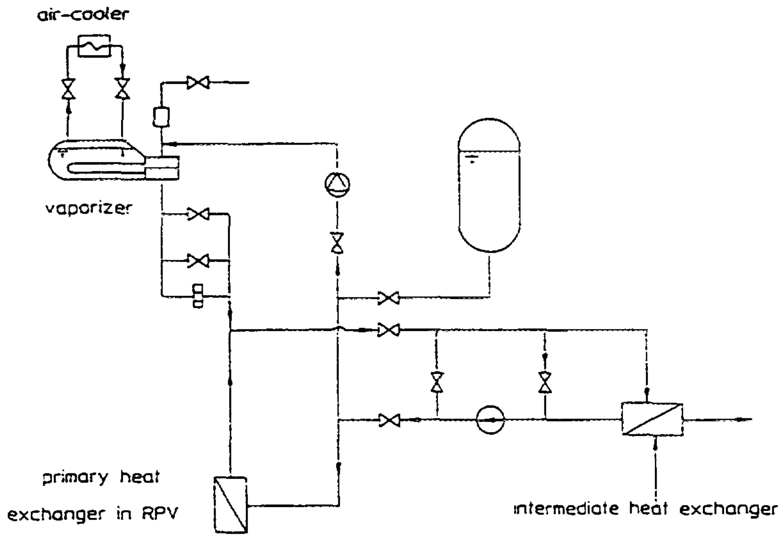


Fig.4 The schematic system diagram of RHRS

2.3. Main heat transfer system

The main heat transfer system is composed of three circuits, i.e. the primary circuit, the intermediate circuit and the heat grid. The intermediate circuit is a single loop which connects with the primary circuit and the heat grid via the four primary heat exchangers and two intermediate heat exchangers. The four primary heat exchangers are divided into two groups in parallel operation, which are merged into a single loop through isolating valves. The operating pressure in the intermediate circuit is higher than in the primary circuit. This choice can keep the heat grid free of radioactivity. Heat generated in the core is transferred to the heat grid via the intermediate circuit. The main heat transfer system is shown in Fig.3.

2.4. Residual heat removal system

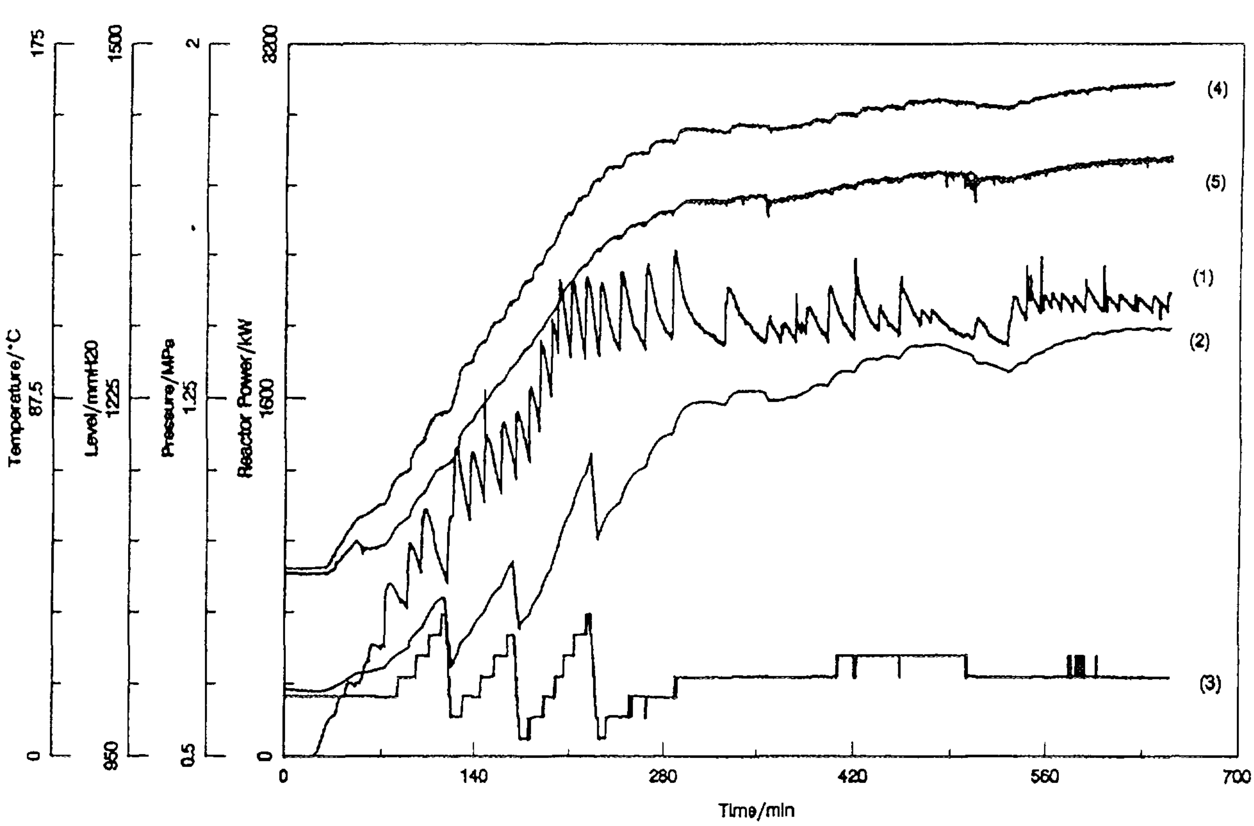
The residual heat removal system (RHRS) of the NHR-5 consists of two independent trains which are assigned to two groups of primary heat exchangers. In each train there are three natural circulation paths. Figure 4 shows the schematic system diagram of the RHRS. After reactor shut-down the decay heat will be transferred to the intermediate circuit via the primary heat exchangers. Then the heat carried is going to a vaporizer located at a high local position in the reactor hall. This is the first natural circulation path. The second natural circulation path consists of the vaporizer, air cooler and related piping and valves. Finally, the decay heat can be discharged to the atmosphere via the air cooler on the floor of the building by natural convection of air.

3. OPERATIONAL EXPERIENCE OF NHR-5

3.1. Reactor operating conditions

Start up

Start up of the NHR-5 is the process from cold condition to the expected operation state by means of nuclear heating. During the start up process three things have to be done,



1 reactor power 2 pressure in RPV 3 coolant level
4 core outlet temperature 5 core outlet temperature

Fig.5 The start up process of NHR-5

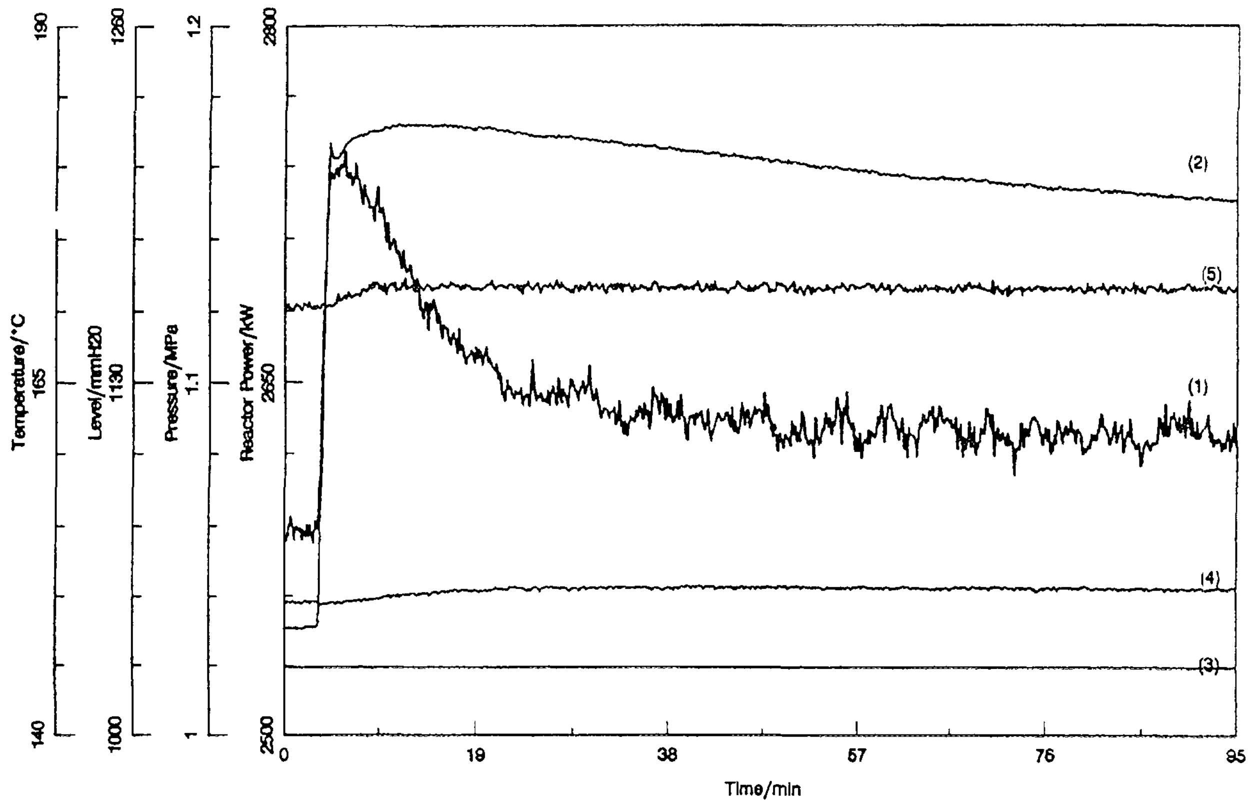
i.e. to set up the initial partial pressure of nitrogen in the RPV, to limit the rising temperature rate to less than 50°C/hr in primary circuit and to keep the coolant level in the RPV in a certain range. Fig.5 shows the start up process to full external load.

Feeding nitrogen and water into RPV

Feeding nitrogen and water into the RPV to compensate their loss caused by various reasons (mainly sampling) is needed for maintaining the normal operating conditions of the NHR-5.

As a result of feeding gas into the RPV, the reactor power increases with the pressure increases and comes to a peak. After that it begins to decrease and finally reaches a new steady state. The result is given in Fig. 6. In the process of this experiment, the reactor power increased by 5.7%, the core inlet and outlet temperature rose 1.1°C and 1.4°C, respectively. The variation of reactor power indicates that there is a certain void content in the core at operation condition.

The reactor has a similar behavior when the water enters the downcomer and then into the core. Due to the coolant level rising, and the feed water temperature being less than the coolant temperature, the core inlet temperature slightly decreases and the pressure increases. For both reasons of pressure rising and temperature decreasing the reactor power increases. The experimental results are given in Fig. 7.



1 reactor power 2 pressure in RPV 3 coolant level
4 core outlet temperature 5 core outlet temperature

Fig.6 Feeding nitrogen into the RPV

Self-pressurized performance

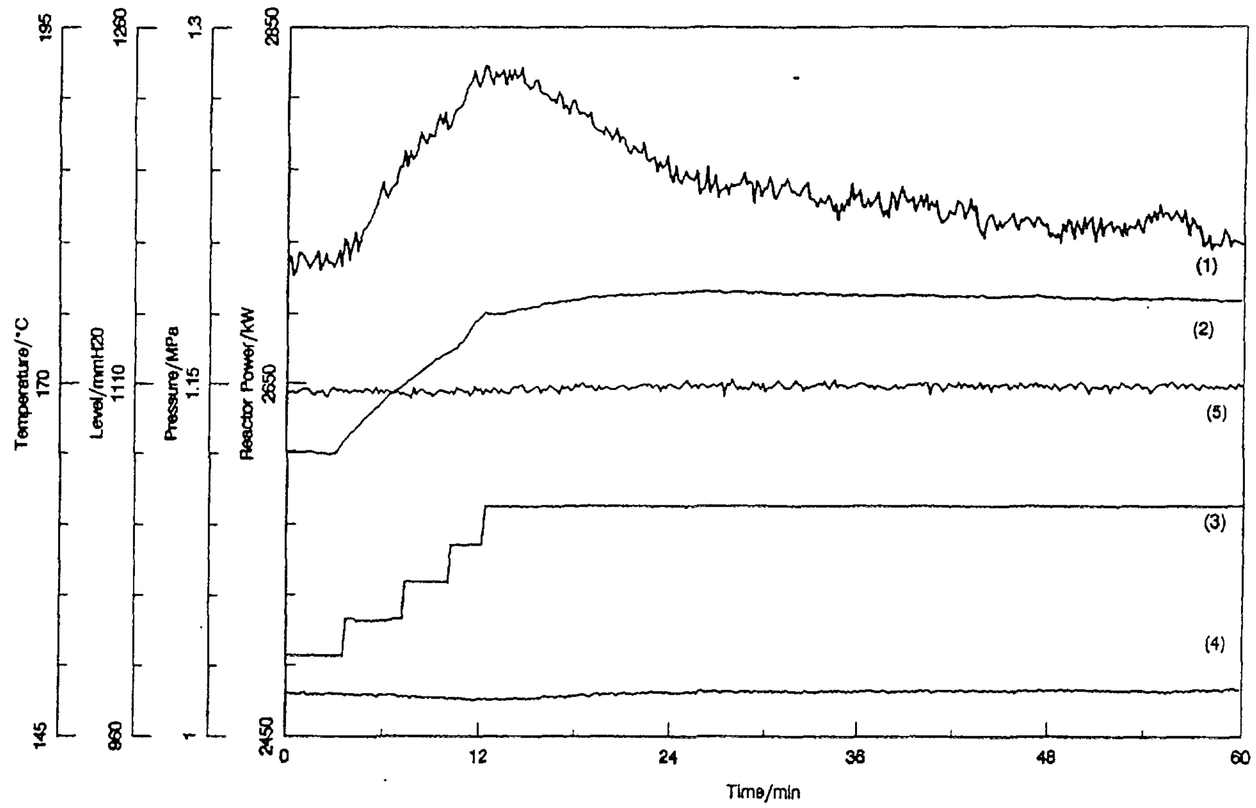
A space in the upper part of the vessel is used for self-pressurizing. The total pressure in the RPV is formed by both a nitrogen partial pressure of 0.3MPa and a saturated steam partial pressure of 1.17MPa which corresponds to the core outlet temperature of 186°C.

The change of the total pressure in the RPV caused by various transient conditions is smooth and small, which results from the large coolant inventory and the large self-pressurized space. For example, when the external load changed to 60% the total pressure in RPV only changes 5%.

The changes in total pressure is by reason of both the changes of core outlet temperature and coolant level.

High operational availability

As a heating reactor, the NHR-5 is only operated in winter. The operational availability of the NHR-5 is evaluated by comparing the actually operated days with the planned operation days. From December 1989 to March 1993, the NHR-5 has been operated for more than 9330 hours. The average availability of the heating operation was about 99%.



1. reactor power 2. pressure in RPV 3. coolant level
4. core outlet temperature 5 core outlet temperature

Fig.7 Feeding coolant water into the RPV

During the four winters operation, there were four unexpected reactor shutdowns caused mainly by failures of electric power supply and faults in the auxiliary systems. Each duration of reactor shutdown was less than 4 hours, so space heating was not affected very much due to the great heat capacity of the heat grid. In spite of the fact that the NHR-5 is the first kind of vessel type heating reactor, it has reached a high availability of heating operation.

3.2. Radiation protection and environmental impacts

Specific radioactivity of water in three loops

During operation the water radioactivity level in the primary circuit, the intermediate circuit and the heating grid have been regularly monitored. The radioactive back-ground of potable water of the site area is about 0.10 Bq/l. The radioactivity level in the water of the intermediate circuit and the heating grid are as low as that of potable water. In the primary circuit the specific radioactivity of coolant is at the level of 2.5E2 to 2.7E3 Bq/l. The nuclide analysis showed that there were no fission products in the coolant.

From the point of view of radioactivity, the isolating the intermediate circuit performs a perfect function in keeping the heating grid free of radioactivity.

Radiation exposure rate

The distribution of radiation exposure in the NHR-5 building is reasonable. A large part of the building have a very low exposure rate near the background level. A higher exposure rate is found outside the biological shielding where the reheater of the primary purification system is placed. A local shielding with lead had to be added to reduce the radiation exposure rate.

Effluents

During normal operation, the gaseous effluent radioactivity level is at the same level as that of the background. The nuclides analysis indicated that there was no artificial nuclide in the effluent. The nuclides in the effluent are natural ^{40}K and Radon daughters. The amount of waste water produced from operation and maintenance is about 10.2 m^3 in four years.

Collective dose

The collective doses for all operators in each heating period are also very low, and are indicated in Table 4. The data demonstrate that the radiation protection design of the NHR-5 was successful.

TABLE 4: COLLECTIVE DOSE FOR ALL OPERATORS IN EACH HEATING PERIOD

Period	Collective dose (mSv-man)
11.1989 - 3.1990	2.4
11.1990 - 2.1991	3.2
11.1991 - 3.1992	11.4

In addition, there are many items to regularly monitor onsite and offsite, such as gamma exposure, gross beta-radioactivity level of aerosol and service drains, offsite samples of water, soil, air and plants, etc.

All measured data indicate that the NHR-5 operation does not cause any changes in radioactivity levels in this area.

3.3. Water chemistry of NHR-5

In consideration of the features of NHR-5: low temperature, low power density, and refueling period being longer than for a PWR, and by reference to the operating experience of nuclear powered ship "Otto Hahn", a water chemistry differing from PWR and BWR is adopted in the operation of the NHR-5. This water chemistry is based on neutral water, not to contain boron and not to add hydrogen to the primary coolant; oxygen is removed by chemical additive (C_2H_4).

The results of monitoring and analyses show that the dissolved oxygen can maintain the level of 40 ppb and a pH value of 6-7. Table 5 lists the analysis results and the specification of primary coolant.

TABLE 5: SPECIFICATION AND MONITORED RESULTS FOR PRIMARY COOLANT

Item	Specification	Analysis results
Dissolved oxygen	<50 ppb	30-40 ppb
pH (25°C)	6-10	6-7
F	< 100 ppb	< 50 ppb
Cl	< 100 ppb	< 50 ppb
Cr	< 10 ppb	< 0.1 ppb
Fe	< 10 ppb	< 0.05 ppb
Na	< 5 ppb	< 5 ppb
Cu	----	< 0.2 ppb
NO_3^-	----	< 5 ppb
NO_2^-	----	< 5 ppb
Total solids	< 100 ppb	< 0.5-1ppm

The nitrate and nitrite contents are less than 5 ppb in the coolant at any operating condition. This concentration is too low to cause metal structures to corrode. So, nitrogen used as cover gas is feasible for the NHR-5.

In order to effectively decrease the dissolved oxygen level in the primary coolant, three things have to be done in the future. The first is to remove the oxygen from the makeup water, the second is to add an additive to the primary circuit continuously, and the third is to exhaust the air from the nitrogen supply lines, especially the nitrogen cylinder.

3.4. Operation of intermediate circuit

Function of intermediate circuit

Besides the function of isolating radioactivity, the intermediate circuit also has other important functions. The heat generated in the core has to pass through it on its transfer to the heat grid. That means a change in operation mode (for example to change the flow rate) will affect the heat transfer to the heat grid. The operation practice demonstrated that the present operation mode of intermediate circuit to be changed is favorable over increasing its average temperature, especially when the reactor is operated at partial load.

To maintain isolating function

To keep the pressure in the intermediate circuit higher than in the primary circuit is the important condition to keep the isolating function. The RHRS is a part of the intermediate circuit during normal reactor operation. In the normal operation condition the pressure in the intermediate circuit depends on the pressure of the pressurizer tank (A) in this circuit. When the intermediate circuit is isolated the isolating condition depends on the pressure of the pressurized tanks (B and C) installed in the RHRS. The two pressurized tanks are connected to the RHRS by small bore valves and piping. Its advantage is that the isolating function can be kept beyond a large loss of water from the intermediate circuit.

Detection of leakage rate for the intermediate circuit

The changes in water level in the three pressurized tanks (A, B and C) is used for detecting the leakage rate. This method is applicable for steady state operation. The operation practice indicates that the leakage rate is more than 2 l/hr (normal leakage rate). This abnormal leakage must occur somewhere else in this circuit.

3.5. Operation of RHRS

The reactor residual heat is removed by a passive residual heat removal system which connects to the intermediate circuit. There are two independent trains of the RHRS which is composed of three natural circulation loops (see Fig. 4).

Hot standby condition

When the reactor is operated in normal condition, the RHRS is working in a hot standby condition. In this case the vaporizer of RHRS and primary heat exchangers work in parallel, and a very small part of the flow rate in the intermediate circuit passes through the vaporizer to prevent freezing in the air-cooler. In order to set up the second circulation-vaporization- and condensation loop, the air on the shell side of the vaporizer has to be discharged at high temperature.

The outlet and inlet temperature of the air-cooler have the approach value, which is the main feature of hot standby condition.

Direction of natural circulation

When the RHRS is put into operation the primary heat exchanger and vaporizer will change from parallel mode to series mode. So the direction of water flow must change in either the vaporizer or the primary heat exchangers, depending on the temperature distribution in this system. In general, if the reactor is operated at a high power level the direction of natural circulation will be the same as in the primary heat exchanger. If the reactor is at a low power level, the direction will be the same as in vaporizer. The experimental results indicated that both circulation directions have the same capacity to remove the decay heat from the core.

From the experimental results it is indicated that the natural circulation of the RHRS can be reliably established, and that the direction of natural convection in the intermediate circuit did not affect the decay heat removal.

Capability test of RHRS

According to the principle of thermal energy balance, the heat removal capability of RHRS is measured at a steady operation state of NHR-5. The heat generated in the reactor core should be balanced by the heat loss, the heat drawn by the purification system and the heat removed by the RHRS. A heat removal capability of 116 KW was measured at the average temperature of 166°C in the primary circuit. This value is more than the design value of 75 KW for each train.

In addition, the RHRS can be operated at a lower temperature than 100°C, and still has a certain capability to remove the residual heat from the core. This shows that the reactor can be cooled down to the cold shutdown by the RHRS only.

3.6. Operational status of control rod driving system

The control rod driving system was satisfactory for starting up, regulating reactor power, and reactor shutdown during the past operation. The full travel time for dropping into the core was still less than 2 seconds.

Owing to the use of a temperature compensation device in the hydraulic drive system, it is not necessary to adjust the flow rate at high temperature.

The ultrasonic position indicators were also satisfactory for indicating the position of the control rods under the pressurized water operation mode. The ultrasonic indicator system can not work under two phase flow or the condition with an interface of gas and liquid. Therefore, the correct position of control rod would not be indicated in this system at the loss of pressure or fast pressure reduction inside RPV.

At the beginning of commissioning a special method has been used for eliminating the interference with ultrasonic sensors to the fission-chamber detecting system.

4. SAFETY FEATURES EXPERIMENT OF NHR-5

In the course of commissioning and operation, a number of experiments have been carried out to demonstrate the feasibility and safety of the vessel type heating reactor concept. In these experiments there were no any external interferences by the operators.

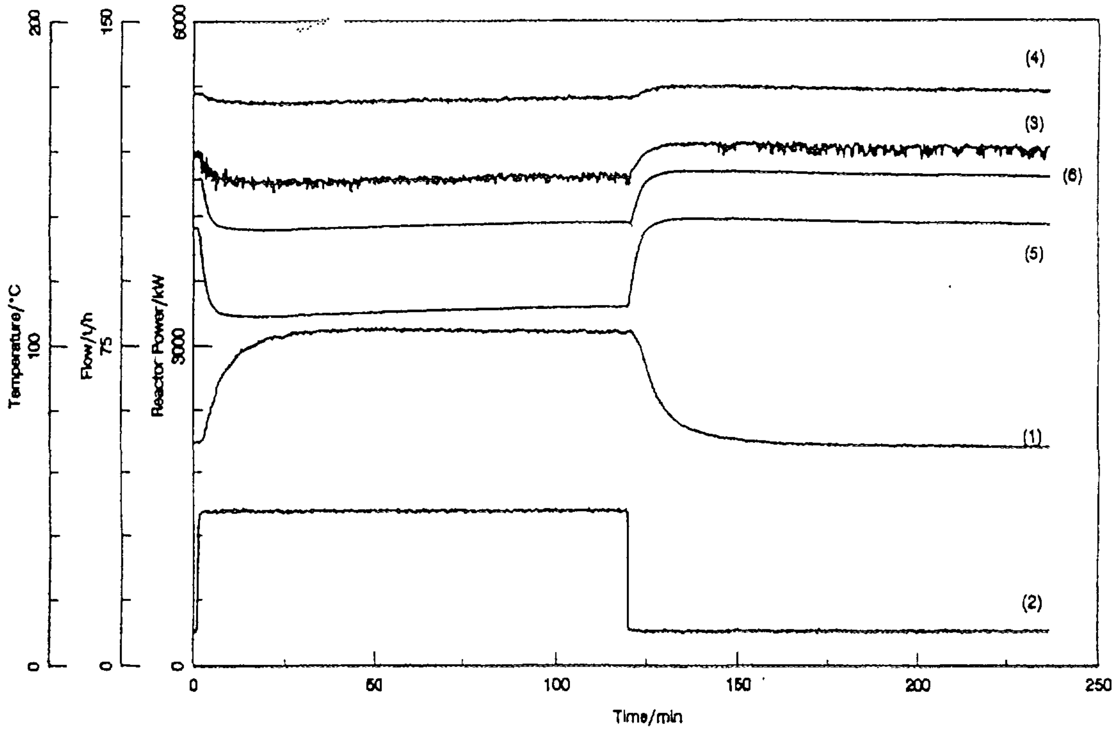
4.1. Self-regulation feature

The self-regulation experiment has been performed to investigate the reactor self-regulation ability to follow a change of the heating load. The heating load can be varied by means of changing the flow rate through the intermediate heat exchangers.

The flow rate through the intermediate heat exchangers was changed from 8 t/hr to 35 t/hr, then back to 8 t/hr. This value corresponds to a heating load change from 1.5 MW to 2.5 MW, a variation of about 66%. Figure 8 shows the behavior of the NHR-5 following the heating load change. The reactor power, caused by the self-regulating mechanism automatically to vary within 90 seconds, reached a new power level to match the heating load within 30 minutes. The moderator temperature coefficient plays a main role in this process. The experimental results show that the NHR-5 has a very good self-regulation ability to follow a load change without any operator action.

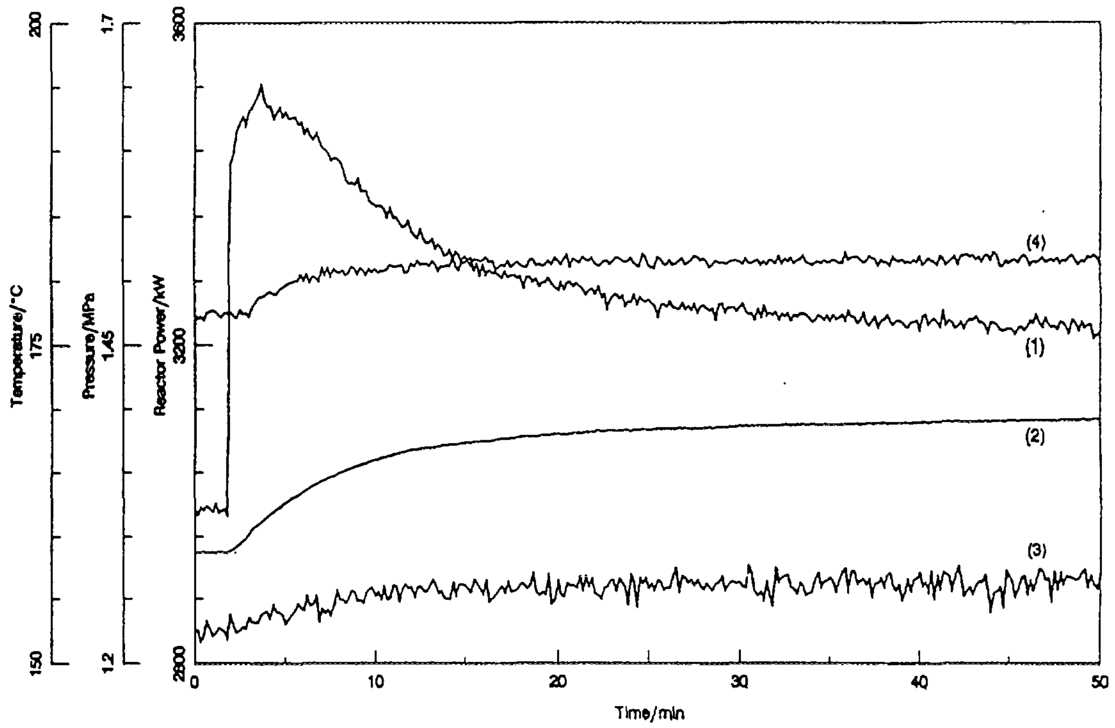
4.2. Self-stability feature

The self-stability experiment was performed in order to investigate the response to a reactivity insertion. In this experiment the reactor was operated at a power of 3.0 MW, then a step insertion of 2 mk reactivity was introduced. Figure 9 indicates the variation of the reactor parameters. At the beginning of the transient, the reactor power increased rapidly due to the extra reactivity and reached a maximum relative value of 1.18 in 100 seconds.



1. reactor power
2. flow rate through intermediate heat exchanger
3. core inlet temperature
4. core outlet temperature
5 inlet temperature in the 2nd of primary heat exchanger
6 outlet temperature in the 2nd of primary heat exchanger

Fig.8 The NHR-5 self-regulation feature



1. reactor power
2. pressure in RPV
3 core inlet temperature
4 core outlet temperature

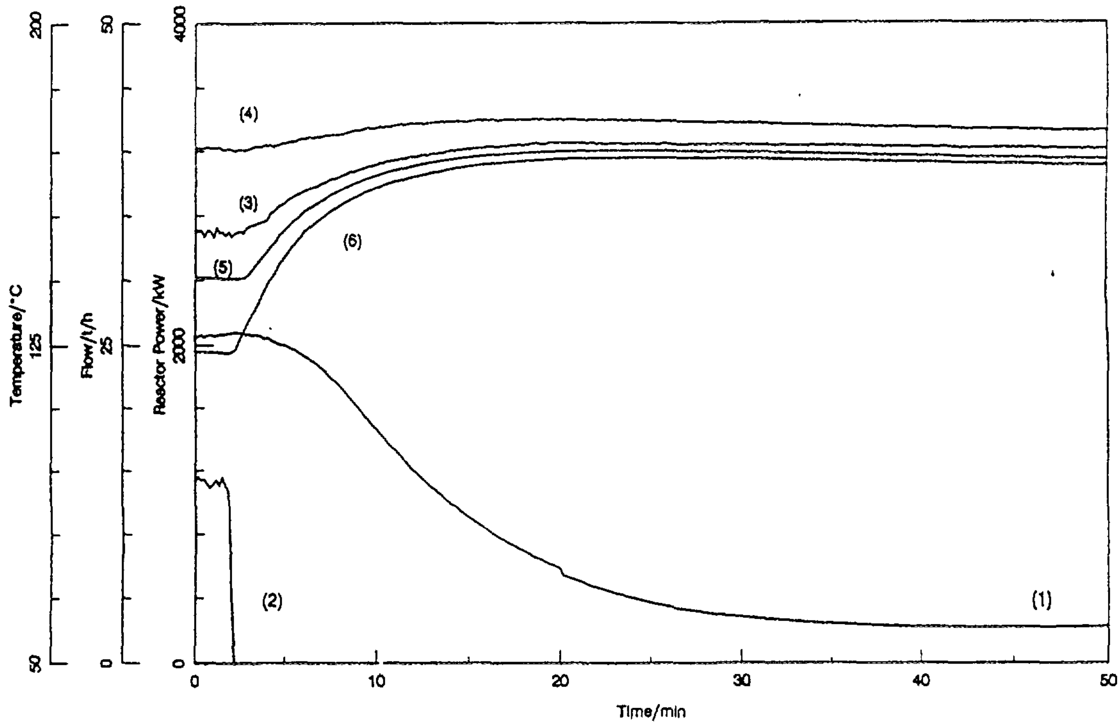
Fig.9 The NHR-5 self stability feature

Then the reactor power began to decrease due to the feedback of the negative reactivity coefficient, and came to a new relative power level of 1.08 in 30 minutes. The core inlet and outlet temperatures added an increment of 3.8°C and 4.2°C, respectively. The reactor pressure increased with a Δp of 0.102 MPa.

4.3. Experiment for ATWS

In order to study the safety behavior of the NHR-5, in 1990 an experiment has been carried out which simulated an ATWS, i.e. a loss of the main heat sink followed by the failure of all 13 shutdown rods.

In this experiment, the intermediate heat exchangers were isolated at a reactor power of 2 MW, and none of the shutdown rods was inserted. Figure 10 shows the power variation observed, together with the changes in temperature and pressure of the reactor. The power decreased as a consequence of the feedback due to the negative temperature coefficient to a stable value of about 0.2 MW in about 30 minutes. The inlet and outlet temperature of the reactor core rose by 20.4°C and 4.7°C respectively. The temperature variation is not serious at all. The primary system pressure rose by 0.23 MPa. The result of the experiment demonstrated that the NHR-5 has excellent inherent and passive safety features. The reactor will be shutdown passively even in the described ATWS case.



1. reactor power 2. flow rate through intermediate heat exchanger
3. core inlet temperature 4 core outlet temperature
5. outlet temperature in the 2nd of primary heat exchanger
6 inlet temperature in the 2nd of primary heat exchanger

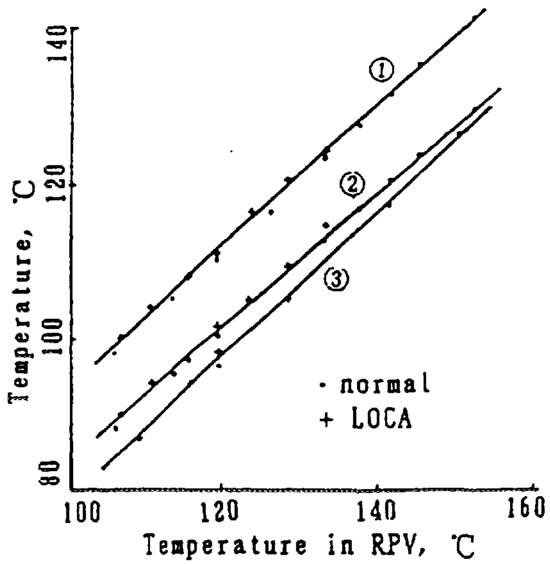
Fig.10 The transient behaviour of loss of main heat sink without scram

4.4. Residual heat removal after interruption of natural circulation in the primary circuit

When a loss of coolant accident (LOCA) occurs in the primary circuit, the water level inside the RPV will decrease. Due to the integrated arrangement of the primary circuit, and due to the feature that all penetrations of small pipes are located at the upper part of the vessel, the reactor core will never be uncovered. But as a result of the water level decrease the natural circulation of the water in the primary circuit might be interrupted. In this case the residual heat of the reactor will be transported by vapor condensed at the uncovered tubes of the primary heat exchangers.

To demonstrate the capability of residual heat removal under LOCA conditions a special experiment was made at the NHR-5 in March 1992. After reactor shut down the water in the reactor vessel was discharged by opening the valve to the blowdown tank. The water discharge rate was $1.6 m^3/hr$ and an amount of $2.4\ m^3$ water was drained off. The water level in the reactor vessel decreased below the entrance of the primary heat exchanger and the water-phase natural circulation was interrupted. In this case the residual heat removal was mainly realized by condensation of the vapor. Due to the discharge of $2.4\ m^3$ water the partial pressure of nitrogen reduced from 0.29 MPa to 0.022 MPa, so that the water subcooling of the reactor outlet temperature decreased from 12°C to 2°C.

The reduction of subcooling enhanced the vaporization - condensation process. Figure 11 shows the comparison of the residual heat removal capabilities during LOCA conditions and under the normal operation. From the results of the comparison it can be shown that the procedures of both LOCA and normal operation are almost the same. The decay heat can be reliably removed by means of vapor condensation on the primary heat exchanger under LOCA conditions.



1. inlet temperature of vaporizer 2. outlet temperature of vaporizer
3. inlet temperature of aie-cooler

Fig.11 The comparison of the capability of residual heat removal

5. SUMMARY

During four winters of NHR-5 heating operation, the reactor has been known as a valuable tool for a number of experiments on operational behavior and safety features. The operational and experimental results have successfully demonstrated the inherent and passive safety characteristics of the NHR-5. It was proven that the design concept and technical measures of NHR are suitable to meet the requirements for district heating in northern cities, cogeneration and air condition in the middle cities of China, as well as the requirements for seawater desalination.

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