Absorber materials, control rods and designs of shutdown systems for advanced liquid metal fast reactors

Proceeding of a Technical Committee meeting held in Obninsk, Russian Federation, 3–7 July 1995

INTERNATIONAL ATOMIC ENERGY AGENCY

June 1996
The IAEA does not normally maintain stocks of reports in this series. However, microfiche copies of these reports can be obtained from

INIS Clearinghouse
International Atomic Energy Agency
Wagramerstrasse 5
P.O. Box 100
A-1400 Vienna, Austria

Orders should be accompanied by prepayment of Austrian Schillings 100,— in the form of a cheque or in the form of IAEA microfiche service coupons which may be ordered separately from the INIS Clearinghouse.
Absorber materials, control rods and designs of shutdown systems for advanced liquid metal fast reactors

Proceeding of a Technical Committee meeting held in Obninsk, Russian Federation, 3–7 July 1995
PLEASE BE AWARE THAT
ALL OF THE MISSING PAGES IN THIS DOCUMENT
WERE ORIGINALLY BLANK
The safe, reliable and economic operation of nuclear reactors depends upon the reliable operation of the reactor shutdown and regulation systems. In liquid metal cooled fast reactors (LMFRs), these consist of control assemblies, drive mechanisms, guide tubes and other structural components. The main element of the control assembly is the control rod, which consists of a neutron absorbing material and is (usually) clad in stainless steel. Neutron absorbing materials in development include the ceramic boron carbide and europium oxide. The current control assembly designs have worked well, but recent developments have shown that their lifetime in the core and reliability could be increased by design improvements.

Fast reactor R&D programmes in some countries reflect the effort to enhance the flexibility to manage plutonium stockpiles. Studies performed recently have demonstrated that using the natural absorber $\text{B}_4\text{C}$ in combination with moderator $^{11}\text{B}_4\text{C}$ could improve the safety of fast reactors in the plutonium burner mode of operation.

Owing to improvements in the reliability of operation of reactivity shutdown and decay heat removal systems, core melting accidents are highly improbable. The first step to prevent core melting is to be able to shutdown the reactor under all probable circumstances. For advanced fast reactors there are two independent reactor shutdown systems: a primary conventional active system and a backup passive system. At present, there are many proposed passive systems for the insertion of control rods into the core.

The IWGFR (International Working Group on Fast Reactors) proposed that the IAEA organize a Technical Committee Meeting on absorber materials, control rods and designs of backup reactivity shutdown systems for breakeven cores and burner cores for reducing plutonium stockpiles, to review operational experience with absorber elements and control rods and the development of passive backup shutdown systems for advanced LMFRs.

This was the third IAEA meeting held on the subject of absorber materials and control rods for LMFRs. The other two meetings were held in Dimitrovgrad (1973) and Obninsk, (1983), in the Former Soviet Union.
EDITORIAL NOTE

In preparing this publication for press, staff of the IAEA have made up the pages from the original manuscripts as submitted by the authors. The views expressed do not necessarily reflect those of the governments of the nominating Member States or of the nominating organizations.

Throughout the text names of Member States are retained as they were when the text was compiled.

The use of particular designations of countries or territories does not imply any judgement by the publisher, the IAEA, as to the legal status of such countries or territories, of their authorities and institutions or of the delimitation of their boundaries.

The mention of names of specific companies or products (whether or not indicated as registered) does not imply any intention to infringe proprietary rights, nor should it be construed as an endorsement or recommendation on the part of the IAEA.

The authors are responsible for having obtained the necessary permission for the IAEA to reproduce, translate or use material from sources already protected by copyrights.
CONTENTS

Summary of the Technical Committee Meeting ........................................... 7

Review of technical approaches and solutions for LMFR control rods development . 11
  V.I. Matveev, A.P. Ivanov, R.M. Vożnjesniski, V.P. Evdokimov

Development and improvement of control rods for the BN-350 and BN-600 reactors . 19
  A.I. Efremov, V.I. Ryakhovskikh, V.B. Ponomarenko, V.M. Chernyshov

Experience of the BN-600 reactor control rods development ......................... 33
  Yu.K. Alexandrov, B.A. Vasilyev, S.A. Iskhakov, O.B. Mishin, V.A. Rogov,
  A.S. Shabalin

Control assembly to be used in CEFR ....................................................... 47
  Xie Guangshan, Zhang Ruxian, Wang Yongian, Li Shikun

The control rod modelling code REGAIN ................................................. 53
  J. Truffert

Experience with control rod drive mechanism of FBTR ................................ 61
  P.V. Ramalingam, M.A.K. Iyear, R. Veerasamy, S.K. Gupta, L. Soosainathan,
  V. Rajan Babu

Development of passive shut-down systems for the European Fast Reactor EFR ...... 69
  M. Edelmann, G. Kussmaul, W. Váth

Design philosophy of PFBR shutdown systems ......................................... 81
  V. Rajan Babu, R. Vijayashree, S. Govindarajan, G. Vaidyanathan,

Design of shutdown system for PFBR ..................................................... 89
  V. Rajan Babu, R. Vijayashree, R. Veerasamy, L. Soosainathan,
  S. Govindarajan, S.M. Lee, S.C. Chetal

Development of passive safety devices for sodium-cooled fast reactors .......... 97
  Yu.E. Bagdasaroy, Yu.K. Buksha, R.M. Vożnesniski, N.V. Vyunnikov,
  V.P. Korniyov, N.D. Krayev, L.I. Mamayev, A.G. Portyanoy, A.P. Sorokin

Main features of the BN-800 passive shutdown rods ................................ 107
  Yu.K. Alexandrov, V.A. Rogov, A.S. Shabalin

The design of a backup reactor shutdown system of DFR ......................... 113
  K. Okada, K. Tarutani, Y. Shibata, M. Ueta, T. Inagaki

Irradiation performances of the Superphenix type absorber element ............. 127
  B. Kryger, D. Gosset, J.M. Escleine

Operation experience of the BN-600 reactor control rods ........................ 141
  V.V. Malisev, V.F. Roslyakov, G.V. Babenko, A.N. Ogorodov

The experience of post irradiation investigations of the BN-600 control rods .. 153
  V.P. Tarasikov, R.M. Vożnesenski, V.A. Rudenko

Irradiation behavior of boron carbide neutron absorber ............................ 161
  T. Kaito, T. Maruyama, S. Onose, T. Horiuchi

Third shutdown level for EFR project .................................................. 173
  D. Favet, B. Carluec, S. Dechelette

Improvement of the performances of the "poisoned" CAPRA core by use of $^{11}$B$_{4}$C ..... 189
  G. Gastaldo, G. Vambenepe, J.C. Garnier

Experience in development, operating, and material investigation of the BOR-60
  reactor control and safety rods ....................................................... 195
  V.B. Ponomarenko, A.I. Efremov, G.I. Gadzhiev, V.D. Risovoy,
  A.V. Zakharov, T.M. Guseva
Production of gamma-sources, based on europium oxide in fast reactors ........ 205
  V.D. Risovany, A.V. Zakharov, E.P. Klochkov, T.M. Guseva,
  V.B. Ponomarenko, V.M. Chernyshov
Reprocessing of the irradiated boron carbide enriched by the boron-10 isotope
  and its reuse in the control rods of the fast breeder reactors .......... 219
  V.D. Risovany, A.V. Zakharov, E.P. Klochkov, A.G. Osipenko,
  N.S. Kosulin, G.I. Mikhailichenko
Experience in production of articles from boron carbide for fast reactor control rods  225
  I.A. Bairamashvili
Diving-bell and double-vented B₄C control rod pin .................... 231
  Li Shikun, Z. Changshan, Xu Yingxian

List of Participants ........................................ 245
SUMMARY

1. INTRODUCTION

The IAEA Technical Committee Meeting (TCM) on Absorber Materials, Control Rods and Designs of Backup Reactivity Shutdown Systems for Breakeven Cores and Burner Cores for Reducing Plutonium Stockpiles, was hosted by the Institute of Physics and Power Engineering (IPPE), Obninsk, Russian Federation, from 3 to 7 July 1995. This was the third IAEA meeting held on the subject of absorber materials and control rods for liquid metal fast reactors (LMFRs). The other two meetings were held in Dimitrovgrad (1973) and Obninsk (1983), in the Former Soviet Union.

Thirty-five specialists from France, Germany, India, Japan, the Republic of Kazakhsan, the Russian Federation and the Republic of Georgia (observer) attended the meeting.

The meeting had seven sessions. The sessions were chaired by B. Kryger (France), R. Babu (India), M. Edelman (Germany), K. Okada (Japan), T. Kaito (Japan), D. Favet (France), V. Risovany (Russian Federation), J.P. Truffert (France) and I. Bairamashvili (Republic of Georgia).

The main topics of discussions were:

- status of control rod designs for fast reactors and experience with operation,
- properties and behaviour of absorber materials for control rods; results of post-irradiation examination of absorber materials, and mechanisms affecting their properties and behaviour,
- design of a backup reactivity shutdown system utilizing passive mechanisms:
  - Curie point electromagnetic mechanism,
  - enhancement of thermal expansion of absorber rod drive lines,
  - hydraulically suspended control rods,
  - gas expansion modules in the core, and
  - the possibility of optimizing the reactivity coefficients and the efficiency of Pu burning by using absorber and moderator materials in the core.

A total of 23 papers were presented, and a technical tour of the IPPE also took place.
2. DISCUSSION AND CONCLUSIONS

2.1. Status of control rod designs for fast reactors and experience with operation

Significant experience, accumulated on absorber materials for fast reactor control rods, was presented at the meeting. Boron carbide of high enrichment (45-90% of $^{10}$B) and vented absorber pins (with sodium bonding) have been successfully used in the Phénix, BN-350, BN-600 and Superphénix (SPX) reactors. The control rod lifetime for the BN-600 reactor is 500 effective days (e.d.); the lifetime predicted for the SPX reactor (based on the tests in the Phénix reactor) is 640 e.d. Boron carbide is the favoured absorber material. Europium oxide ($\text{Eu}_2\text{O}_3$) appeared to be an alternative to boron carbide of natural enrichment.

Investigation of the properties and behaviour of B$_4$C have been carried out in the BN-350, BN-600, BOP-60, Phénix and JOYO reactors for a broad range of conditions affecting its endurance. Data from these studies defined the behaviour of irradiated absorbing materials (both pins and the whole rods) provide the basis for improved control rod designs and extended control rod lifetimes.

Swelling of irradiated boron carbide is one of the main factors limiting the lifetime of the rods. The dependence of the boron carbide swelling rate ($\Delta d/d\%/B\%$) on the burn up was determined from post-radiation studies of control rods from the BN-600 reactor. After about 500 e.d. the boron burn up was $\sim$18%. A minimum swelling rate of $\sim$0.1% for enriched boron carbide and 0.2-0.4% for natural boron carbide was observed at a temperature of 500°C. At a working temperature of about 800°C, the swelling rate is $\sim$1.0% per percent burnup for natural boron carbide and 0.6% per percent burnup of enriched boron. Some reserve remained since the gap between the absorber and cladding was not filled completely. Under these conditions mechanical interaction of the cladding was not observed, and the cladding material retained a sufficient reserve strength.

A series of tests were carried out in the Phénix reactor irradiating an enriched boron carbide rod for 240 e.d. These experiments showed that the in-pile residence time of the pin is limited by the mechanical interaction of boron carbide with the cladding, owing to absorber fragments entering the gap between the absorber and the cladding.

Irradiated boron carbide data (swelling rate and gas release on burnup) were also obtained in Japan in tests in the experimental fast reactor JOYO. Burn up of about 23% ($230\times10^{26}$ capt/m$^2$)$^1$ was achieved at a maximum temperature of 1400°C. The dependence of pellet swelling and gas release on burnup was measured. Japanese and French results of the absorber material behaviour under irradiation are in good agreement but differ in some cases from Russian results. It appears that the differences can be explained by differences in methods of obtaining enriched boron carbide. However, the results obtained are not contradictory, and they improve the understanding of the swelling and gas release processes.

New absorbing materials and new pin designs are required to achieve control rod lifetimes of 1000 e.d. Some measures to increase the control rods lifetime for existing reactors

---

$^1$ The abbreviations capt. or cap. stand for capture
were proposed by the French. The first proposal was to reduce the capture rate in boron carbide by lowering the $^{10}\text{B}$ enrichment of the $\text{B}_4\text{C}$ pellets in the lower part of the control rods pins. The second proposal was to prevent absorber fragment relocation by providing a thin stainless steel shroud to enclose the pellet stack.

Considerable progress has been made in the Russian Federation in the design and manufacture of rods containing absorbing (boron carbide) and moderating (zirconium) hybrid materials. Using this design with its lifetime of 450 e.d. would result in considerable saving of high cost enriched boron.

The participants agreed that the most efficient way to utilize high enrichment boron is to separate it from irradiated materials. The development of technology for reprocessing irradiated boron carbide, which provides complete removal of radionuclides from irradiated materials, was presented at the meeting. This permits repeated use of $^{10}\text{B}$ enriched $\text{B}_4\text{C}$ in fast reactors.

It became clear after discussion that the French control rod calculation code REGAIN can analyse both steady state and transient pin behaviour and covers thermal, mechanical and some chemical characteristics. For boron carbide materials the code can deal with both sodium bonded and helium bonded design concepts.

The meeting participants concluded that the problems of reliable and effective control rods for sodium cooled fast reactors are being solved.

2.2. Optimizing core safety parameters for a plutonium burner reactor using absorber and moderator materials

It was noted that two approaches to optimize a fast reactor core for plutonium (or minor actinide) burning have been considered in France: the "dilution approach" (about two thirds of the pins are fissile pins, the remaining one third are fuel free) and the "poisoning approach" (the absorber $\text{B}_4\text{C}$ is introduced in the diluent subassemblies). At present, the dilution approach is preferred since it allows better core safety parameters (mainly the Doppler constant). Data presented in the French paper indicated that the introduction of the moderator in the fuel bundle of the $\text{B}_4\text{C}$ poisoned core strongly increases the Doppler constant (+60%) and reduces the sodium void reactivity worth (-20%). As a result the poisoned core option with $^{11}\text{B}_4\text{C}$ has similar plutonium burning characteristics and better safety parameters than the dilution option. Its main advantages result from the large diameter fuel pins which increase fuel lifetime using a standard fuel bundle. The use of $^{11}\text{B}_4\text{C}$ as filling material does not introduce additional technical problems.

It was concluded that the poisoned reactor option deserves further investigation.

2.3. Advanced passive reactor shutdown systems

The meeting participants noted that in classic core disruptions (unprotected loss of flow (ULOFL) and unprotected transient overpower (UTOP), the reactor can be shut down only by inherent passive reactivity feedback mechanisms. Therefore, special measures to provide additional negative reactivity are needed.
The Russian Federation has developed a backup passive shutdown system based on hydraulically suspended absorber rods (HSAR) which drop into the core when the coolant flow rate in the core falls below 50%. Full scale HSARs for the BN-600 and BN-800 reactors have been tested. This system is capable of shutting the reactor down in a ULOF transient.

The German system for passive shutdown of the European Fast Reactor (EFR) uses enhanced thermal expansion of control rod drive lines (CRDL) to provide automatic scram. This system forces the control rods into the core when pre-set coolant temperatures are exceeded. This device uses a hydraulic expansion module and two individual (coaxially arranged) drive lines. The sodium has a larger thermal expansion coefficient than the container material. This system is capable of shutting the reactor down in ULOF and UTOP transients.

The demonstration fast breeder reactor (DFBR) in Japan includes the following passive backup shutdown systems:

- gas expansion modules (GEMS), and
- self-actuated shutdown systems (SASS) using a Curie point magnetic alloy.

In addition to GEMS and SASS, a feasibility study of the enhanced thermal expansion of CRDL is in progress.

The French presentation described and evaluated a third shutdown level for the EFR project which has been developed, implementing the following four new device systems:

- a system which terminates the power to the absorber electromagnets after a loss of primary pump electric power supply,
- a device for CRDL passively enhanced thermal expansion,
- a device which overcomes control rod jamming by motorized insertion of absorber rods, and
- a mechanical stroke limitation device which passively terminates the withdrawal of a faulted control rod.

The inclusion of these systems in the EFR project leads to a significant improvement of the shutdown function.

The shutdown system of the Indian prototype fast breeder reactor (PFBR) incorporates diverse and redundant features and includes the following systems:

- a system which terminates the power supply to the absorber electromagnets after a loss of pump power,
- a gas expansion module,
- a Curie point magnetic switch

It was concluded that with the introduction of these passive backup shutdown systems, future advanced LMFRs will have a very high degree of safety.
REVIEW OF TECHNICAL APPROACHES AND SOLUTIONS FOR LMFR CONTROL RODS DEVELOPMENT

V.I. MATVEEV, A.P. IVANOV, R.M. VOZNESENSKI,
V.P. EVDOKIMOV
IPPE, Obninsk,
Russian Federation

Abstract

This paper reviews and gives a retrospective analysis of technical approaches and solutions used during control rods development for sodium cooled fast reactors. General principles of fast reactor control rod design development and absorber material selection are considered, as well as the main results obtained in the field of advanced control rods containing absorber and moderator materials.

INTRODUCTION

In the process of BN-350 and BN-600 fast power reactors development and operation a problem of creation of highly efficient and economical control rods was solved. This problem arose due to some peculiar features of sodium cooled fast reactor physics and technology which, as is known, reduce to as follows:

- because of low neutron - absorption cross-sections at high energies the range of materials suitable for their use in the control rods is substantially limited;
- because of comparatively small fast reactor cores dimensions connected with high thermal loads there arise certain difficulties with accommodation of a large number of control rod drive mechanisms;
- high neutron fluxes in the core (~16 n/cm²·sec) determine high rates of nuclear reactions in control rod materials resulting in high energy release and burn-up values, as well as high fluence upon structural materials.

In the course of fast reactor control rods development and validation, primarily for the BN-350, BN-600 reactors and for the BN-800 reactor project, extensive studies in the field of physics, technology and material research were carried out, many engineering solutions and approaches were considered and verified. The aim of this paper is a brief presentation of these solutions and their retrospective analysis.

1. Main Principles of the Choice of Control Rods for BN - Type Reactors.

The BN-type reactors control system has a number of features the main of which are as follows:

- general arrangement of the control system is performed on a basis of the separated- functions, principle owing to which the rods containing high- enrichment boron carbide are outside the core during reactor on-power operation;
- in all types of rods with boron carbide vented absorption elements are used.

Besides, for the advanced BN-800 and BN-1600 reactor projects for a long time there were developed fuel compensators for reactivity compensation due to fuel burn-up. However, such type of rods was excluded from subsequent developments of the above reactors.

The principle of control and safety rods separation according to their functional duty was used still at the design stage of the first power reactor BN-350 and is continued to be used in all further control and safety system developments for BN-type reactors. It should be noted that therewith the functions of some rods can be combined.
For example, there are combined safety and temperature compensation functions. However, the main aim of using this principle is fulfilled: the high enrichment boron rods are under facilitated conditions of operation. The most important problem at the development of the control and safety system rods is the choice of absorption material. Already at the very early stage of the BN-350 reactor design the enriched boron carbide was chosen for safety rods and automatic control rods. Later on, however, at various laboratories there were started a search for and studies of other absorbing materials for their use in fast reactors. The main reason for these studies was a lack of confidence in boron carbide irradiation resistance prevailing at that time. As one of alternative absorbers there was chosen and validated CrB₂ - based material (CrB₂ alloyed by Ta). This material was used in the BOR-60 reactor in 1972-1973 and showed good radiation stability. However, because of lower (by about 20%) efficiency as compared to boron carbide and more complex technology it was withdrawn from production and replaced by boron carbide. (n,γ) absorbers seemed to be highly attractive for their use in fast reactors. These materials still could not compete with enriched boron by their efficiency but in those cases where enriched boron carbide was used they seemed to be a reasonable alternative. One of such materials was tantalum. For the BN-350 reactor a tantalum control rod was developed and fabricated which, however, was not used. The most serious alternative to boron carbide was europium oxide (Eu₂O₃). By its main characteristics this material is substantially superior to boron carbide: high stability of its efficiency in during reactor a run, absence of gas release and swelling. This material was used in burn-up compensation rods of the BN-600 reactor which were successfully operated till 1988. Only one disadvantage, i.e. their high residual activity, substantially complicating spent rods handling, predetermined in the end passing over to boron carbide.

In the course of the BN-type reactors design the development work on some other absorption materials (on the base of EuB₆, ReB₃, etc.) was carried out which, however, was not brought up to the introduction stage.

The general conclusion from the investigations conducted is that boron carbide despite a number of its disadvantages proved to be the most optimum absorption material for its use in fast reactors.

The most important element of the control rod design primarily determining the life time of the rod is the absorber pin. In the control rods of the first BN-350 loading the sealed absorber rods with helium sub-layer were used. A disadvantage of this type absorber rods is their short life time determined by a permissible pressure on cladding of helium released from absorber during operation. The life time of such rods was 2-4 eff. months.

The use of vented absorber pins with a sodium sub-layer has radically influenced upon the improvement of technical and economical characteristics of the control and safety system rods. A vented absorber pin is made in form of a cladding with two end plugs. In the upper plug an item of unsealing is made in form of a system of holes. Through them filling up of absorber pins with sodium in the reactor and gas release into the circuit take place. In this type absorber pins no problem of gas pressure on the cladding occurs. To a considerable degree, a problem of absorber pressure upon cladding is withdrawn as the presence of a sodium sublayer allows to change the size of the absorber - cladding gap at a constant absorber temperature.

Therewith an established level of the absorber temperature determines, in accordance with its temperature dependence, the swelling rates, the minimum swelling rate being 5-6 times less than in a sealed absorber pin.

With a sodium sub-layer the absorber is more resistant to fragmentation; in spite of the presence of cracks it remains as if intact, no fall of small fragments into the gap between absorber and cladding occurs. The above fact allowed to abandon the use of small-diameter absorber pins used with the aim to exclude radial cracking of pellets due
to temperature stresses. Passing over to "thick absorber pins" has allowed by the present time to unify the rods by the absorber pin design, to improve economical characteristics indexes.

It should be noted that the introduction of the vented absorber pin design was preceded by testing of various type items of unsealing (hydraulic valve, discs with a discontinuous weld, compacted metallic wire, etc.) as a part of experimental absorber pins and rods of the BR-5, BOR-60, BN-350 reactors. Now the vented absorber pins are used in all types of control and safety boron carbide-based rods. No remark upon their operation were made. A maximum life time of rods with vented absorber pins was achieved in the BN-600 reactor and was ~ 500 eff. days (a burn-up of ~ 19% at. of boron).

Specific design of the rods is determined by their functional duty. The control and safety rods are made in form of a cylindrical multimembered structure connected by hinges. For a free passage of rods in the guide tubes a sufficient gap between the rods and the tube, as well as an optimum length of members should be assured. The working part of the rod is made in form of an absorber pins bundle. The load-bearing element in the working member of the rods of the BN-350 first loading was the central absorber pin with a thicker cladding. Its ends were fitted with lattices in which, as in a separator, the absorber pins were assembled. In the second-loading rods the wrapper tube was used as a load-bearing element that allowed to avoid rods seizing in the guide tube because of peripheral absorber pins bending and to adjust the conditions of heat removal from absorber pins as well. At present the number of working members in experimental rods is reduced to one (at a length of ~ 1 m) that allowed to increase rods efficiency due to elimination of hinges (in the safety rod of the BN-350 core first loading the working section consisted of three members).

The rods efficiency is also markedly affected by the choice of structural materials. In control and safety rods of the BN-600 and BN-350 reactors the following structural materials are used:
- for absorber pins - X16H15 (ЭИ-847) type austenitic steel;
- for the rest of rod elements - X18H10T type austenitic steel.

At present the life time of the BN-600 reactor standart control and safety rods is set at 365 eff. days and is limited by structural material (X18H10T) performance. In the BN-600 experimental control and safety rods more radiation-resistant materials are used: for absorber pins - X16H15+Ti c.d. austenitic steel; for the rest of rod elements 1X13M2ФP-type ferrite steel. The above rods were operated for 500 eff. days up to a dose of 70 dpa. At present these materials have been operated in fuel subassemblies up to 100 dpa that validates their performance as part of control and safety rods during 700 eff. days. It should be noted that in control and safety rods the same structural materials are used as in fuel subassemblies.

In Table 1 main characteristics of the BN-350, BN-600 and BN-800 control and safety rods are presented.

2. Design Development of Control Rods Containing Absorber and Moderator Materials (rod-traps).

As it was mentioned above, the experience of fast reactor development shows that expensive absorber materials are to be used at least for the reactor control rods. High cost of the absorber based on the high enrichment boron resulting in the high cost of the control rod, stimulated the search for technical solutions that would allow replacing high enrichment boron by other materials or decreasing its amount in the control rods.

The most attractive way of this problem solution is using of so called rod-traps, containing absorber and moderator materials. The moderator softens neutron spectrum in the absorber section of the rod, resulting in the increase of absorber neutron capture
Table 1 Main Characteristics of BN-type reactors control and safety rods.

<table>
<thead>
<tr>
<th>Reactor</th>
<th>BN-350</th>
<th>BN-600</th>
<th>BN-800</th>
<th>BN-600 (test rods)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Rods</td>
<td>Rods</td>
<td>Rods</td>
<td>Rods</td>
<td>Rods</td>
</tr>
<tr>
<td>Total number of rods</td>
<td>12</td>
<td>27</td>
<td>30</td>
<td>4</td>
</tr>
<tr>
<td>The number of various-type rods</td>
<td>3</td>
<td>5</td>
<td>9</td>
<td>1</td>
</tr>
<tr>
<td>Absorber material</td>
<td>BeC 60-80% (B-10)</td>
<td>BeC 60% (B-10)</td>
<td>BeC 80% (B-10)</td>
<td>BeC 92% (B-10)</td>
</tr>
<tr>
<td>Absorber quantity (B-10 and Eu2O3), kg</td>
<td>2.545</td>
<td>2.100</td>
<td>0.225</td>
<td>-</td>
</tr>
<tr>
<td>Average efficiency of a single rod (%Δk/k)</td>
<td>1.10</td>
<td>1.00</td>
<td>0.20</td>
<td>-</td>
</tr>
<tr>
<td>Absorber pin type</td>
<td>v</td>
<td>v</td>
<td>s</td>
<td>v</td>
</tr>
<tr>
<td>Specified life-time (eff. days)</td>
<td>390</td>
<td>365</td>
<td>280</td>
<td>550</td>
</tr>
</tbody>
</table>

Notation used in the table:
- SR - safety rods
- TC - temperature compensators
- SR-L - safety rods (loop-type)
- CR - fuel burn-up compensation rods
- RR - regulating rods
- (v) - vented absorber pins; (s) - sealed absorber pins
rate, i.e. its efficiency. The calculations show that the efficiency of natural absorber materials can be considerably increased in this type rods.

Analytical and experimental studies have proved that the highest rod efficiency can be achieved by combining absorbers on the base of boron carbide or europium oxide with material hydride-based moderators. First technical proposal on this approach to the fast reactor control rod design was developed in the USSR in 1969. This proposal was the base for the development of the BN-350 control rod design. The working section of the rod contained the annular design absorber element made of europium oxide with the stainless steel cladding. Inside this element, seven moderator pins were located, made of zirconium hydride with the stainless steel cladding. This rod was in operation in the BN-350 reactor during about 300 effective days in 1980-1981. According to the results of in-reactor measurements, the efficiency of this rod-trap turned out to be somewhat lower than that of the regular rod design with $B_4C$ of 80% enrichment.

Enriched boron carbide was used for the further design development of the rod-trap. According to the calculation results, in this case $B_4C$ amount in the control rod can be several times reduced as compared to that of regular rod design. This can result in about 10% increase of the rod efficiency. Various designs of the rod-trap with enriched boron on the periphery and zirconium hydride in the central part were tested at the BFS critical facility and in the BN-350 and BN-600 reactors.

Several rod-trap designs with the outer absorber in the form of an annular element and in the form of a set of $B_4C$ pins with boron-10 isotope enrichment of 60, 80 and 92% were tested within the system of scram rods in the BN-350 and BN-600 reactors. These rod-traps were in operation in reactors for 150 to 400 eff. days, and no claims were brought from the reactor personnel on their performance.

Main data on experimental rod-traps tests in the BN-350 and BN-600 reactors are presented in Table 2.

<table>
<thead>
<tr>
<th>№</th>
<th>Reactor, rod type</th>
<th>Absorber</th>
<th>Operation life time, eff. days</th>
<th>Time of tests</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>BN-350, fuel compensator, annular type</td>
<td>$Eu_2O_3$</td>
<td>300</td>
<td>1980-81</td>
</tr>
<tr>
<td>2</td>
<td>BN-350, safety rod, pin type</td>
<td>$B_4C(92%)$</td>
<td>230</td>
<td>1987-88</td>
</tr>
<tr>
<td>3</td>
<td>BN-350, safety rod, pin type</td>
<td>$B_4C(92%)$</td>
<td>400</td>
<td>1989-91</td>
</tr>
<tr>
<td>4</td>
<td>BN-350, safety rod, pin type</td>
<td>$B_4C(92%)$</td>
<td>450</td>
<td>in operation since 1994</td>
</tr>
<tr>
<td>5</td>
<td>BN-600, safety rod, pin type</td>
<td>$B_4C(60%)$-lower member $B_4C(80%)$-upper member</td>
<td>311</td>
<td>1984-85</td>
</tr>
<tr>
<td>6</td>
<td>BN-600, safety rod, pin type</td>
<td>$B_4C(80%)$</td>
<td>311</td>
<td>1989-90</td>
</tr>
<tr>
<td>7</td>
<td>BN-600, safety rod, pin type</td>
<td>$B_4C(80%)$</td>
<td>348</td>
<td>1990-91</td>
</tr>
<tr>
<td>8</td>
<td>BN-600, safety rod, pin type</td>
<td>$B_4C(80%)$</td>
<td>320</td>
<td>1991-92</td>
</tr>
</tbody>
</table>
Physical processes in the rod-trap have rather a complicated pattern for the description of which sufficiently precise calculation methods should be used. At choosing a design of such a rod the optimization of the absorber and moderator materials ratio is required. As an example, some results of pin-type rod-traps calculations for the BN-600 reactor are presented. Europium oxide was used as absorber, zirconium hydride-as moderator.

In the calculations various versions of traps were used: direct (moderator inside the rod), reverse (moderator on the outside), homogeneous ones-placed in the core centre. The calculation model of the rods was a cylinder of the core height and with a radius of =5.36 cm. For direct and reverse versions the cylinder was two-zoned (the zones differing by their material). In the calculation variants the dimensions of the inner zones of the rod were varied, with material density in them being retained. In a homogeneous version the absorber-moderator ratio is in accordance with the direct version in which this ratio changed at the expense of the inner zone radius variation. Besides, for the direct version the outside radius of the rod was also varied.

The calculations were carried out by the multigroup diffusion codes. The results are presented in Fig. 1. As it follows from these data, the direct trap is most efficient: its efficiency is 30% higher than that of a rod consisting of the

![Graph](image)

Fig. 1. Various-type trap-rods efficiency as a function of absorber fraction in the rod ($e = S_{abs}/S_{cell}$)
moderator alone (at the same volume fractions). The rod-trap with homogeneous composition loses in efficiency by about 5%. The reverse trap "loses" in efficiency by about 25%. At an increase of the outside radius of the rod the gain in efficiency as compared with a rod consisting of moderator alone increases: at a rod radius increase by about 2 times the gain is 50%.

A set of investigations carried out on the validation of the rod-trap designs give good prospects for their use instead of expensive rods with enriched boron carbide.

3. The Development of Fuel Compensators.

At the first stage of the BN-800 and BN-1600 reactors design it was expected to use fuel compensators for burn-up compensation. Such solution was caused by an aim to decrease the control and safety system effect upon the reactor breeding characteristics. At using the absorbing compensators, e. g., in the BN-600 reactor, total decrease of the breeding ratio is 0.05-0.06 (it can be roughly considered that compensation of 1 % $\Delta k/k$ leads to a loss in the breeding ratio by 0.02). The use of fuel compensators would reduce these losses practically to zero. At the same time it is known that the main disadvantage of traditional-type fuel compensators (as, e. g., in BN-350) is connected with their low efficiency. Therefore, various proposals aimed at increasing the fuel compensator efficiency were considered. One of the proposals is connected with using uranium metal, having density 2 times higher than that of uranium oxide, in the absorbing section of the compensators. The development and implementation of this proposal will allow to increase compensators efficiency by about 30%. Another proposal is related with the use of somewhat unusual composition, - uranium-238-free material, - in the fuel section. It is known that the positive reactivity of a fuel composition is the difference between the positive reactivity of fissionable material and the negative reactivity of raw material. Hence it follows that if to remove the raw material (uranium-238) then the efficiency of the remaining fissionable part increases substantially. Let us note, that it is impossible to increase efficiency simply at the expense of increasing the amount of fissionable material, because of the limitation from the viewpoint of heat removal.

At the same time, uranium-238-free fuel material in the process of irradiation loses its efficiency much faster than conventional fuel material, as there is no compensation due to fuel production from raw material. Therefore, the comparison of these two type compensators efficiency should be carried out with account of their burn-up.

Fuel compensators (or burn-up compensators) are two-member constructions consisting of the absorbing and fuel sections their lengths being approximately equal to the core height. The fuel element of the upper absorbing section contains raw material (U-238). Fuel elements of the bottom fuel section are similar to those of the core. In Russian fast reactors both absorbing control rods and fuel compensators are of a cylindrical form. This determines the round form of hole in the hexagonal tube, which is inserted into the cell to be used for control and safety rods. Thus, the core cell where the fuel compensator is located is characterized by additional "dilution" with steel and sodium compared to the conventional fuel subassembly that results in some increase of breeding ratio (BR), though not a large one. Actually, the fuel compensators do not reduce the BR, in contrast to the absorbing rods. Compared to these rods, the fuel compensator (at equal efficiency) renders a smaller effect on the energy release fields. This is caused by oppositely directed perturbing effects of the absorbing and fuel sections. In the course of burn-up compensation, fuel compensators are
inserted into the core from the bottom upwards; this kind of motion has been practiced from the viewpoint of safety - in case of a compensator drop for example, at reloading, negative reactivity will be inserted.

Table 3 presents the calculated values of efficiency of various types burn-up compensators for the BN-350 reactor.

Efficiency of various types fuel compensators positioned in the centre of the BN-350 fast reactor.

<table>
<thead>
<tr>
<th>Rod type</th>
<th>Efficiency of fuel section (%Δ k/k)</th>
<th>Efficiency of absorbing section (%Δ k/k)</th>
<th>Efficiency of the rod (%Δ k/k)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0.137</td>
<td>0.094</td>
<td>0.231</td>
</tr>
<tr>
<td>2</td>
<td>0.137</td>
<td>0.145</td>
<td>0.282</td>
</tr>
<tr>
<td>3</td>
<td>0.237</td>
<td>0.145</td>
<td>0.382</td>
</tr>
<tr>
<td>4</td>
<td>0.181</td>
<td>0.145</td>
<td>0.326</td>
</tr>
</tbody>
</table>

The rod types presented in the table have the following compositions:

Type 1 - the fuel section contains PuO$_2$+UO$_2$, and the absorbing one-UO$_2$ (depleted uranium);

Type 2 - PuO$_2$+UO$_2$ in the fuel section, and depleted metallic uranium in the absorbing one;

Type 3 - the fuel section contains the low enrichment zone plutonium without uranium-238, with an inert diluent, and the absorbing section is made of depleted metallic uranium;

Type 4 - the fuel section contains U-238-free uranium-235 with inert diluent, and the absorbing section is similar to types 2 and 3.

It can be seen from Table 3 that in a maximum case the efficiency of the improved compensator exceeds that of a conventional fuel compensator 1.65 times. However, this comparison is not conclusive, since the consideration has been given to “fresh” compensators. If the values of economically justified burn-up of fuel compensators are taken into account, the above gain is decreased approximately to 1.34.

This improved efficiency of fuel compensators appears in sufficient for the BN-800 reactor to provide the required period of reactor operation between reloadings. Therefore, the development of fuel compensators for the BN-800 design were terminated. However, the above developments can be of interest for perspective fast power reactor with high breeding.

CONCLUSION

Work performed during nearly thirty-years period has resulted in practically complete solution of the problem of the control rods development for fast power reactors.

By the present time there have been developed and experimentally tested in the BN-350 and BN-600 reactors:

- control and safety system rods of all types (safety rods, compensating rods, control rods) with higher efficiency, high reliability and extended specified life time of ~ 500 eff. days; the rod designs have some margin for an increase of their life time up to 650 eff. days;

- "trap"-type safety rods allowing to exclude or substantially reduce the use of expensive high-enrichment boron-bearing materials.

Work on partial experimental validation of increased-efficiency fuel compensators for their possible use in advanced fast reactors with high breeding ratio has been also carried out.
DEVELOPMENT AND IMPROVEMENT OF CONTROL RODS FOR THE BN-350 AND BN-600 REACTORS

A.I. EFREMOV, V.I. RYAKHOVSKIKH, V.B. PONOMARENKO, V.M. CHERNYSHOV
Moscow Factory Polymetals, Moscow, Russian Federation

Abstract

This paper reports on the main direction of activities regarding improvement of technical and economic characteristics of control rods, results of the rod modernization, and perspectives of further improvement of control rods for the fast reactors BN-350 and BN-600.

At all time in the design and operation of Russian fast neutron reactors problems connected with the development, tests and fabrication of control protection system (CPS) rods were resolved. CPS rods are the most critical components of the nuclear reactor.

CPS rods for fast reactors must be designed for operation in high neutron fluxes and at operating temperatures of cladding up to $600^0C$. At this conditions CPS rods must be conformed to following requirements:
- to provide the free moving in directional tubes and the reliability of the insertion in core at an emergency;
- to keep the mechanical strength;
- to keep the requirement level of the efficiency during the operating life.

Using absorbing and constructional materials must:
- to be compatible to one another in the whole operating temperature region and emergency conditions;
- to have the minimal irradiation destruction and swelling under neutron fluxes.

Efforts of the developers designing modern CPS rods focus on two main directions:
- improvement of the reliability and operating life of rods;
- development of the economical constructions.

Stages of the development and improvement of rods is shown in scheme 1.

1. Cassette-type design of the CPS rods with AELs, containing different enriched $B_4C$. At development CPS rods for BN-350 and BN-600 reactors were used sealed AELs with special gas compensator for reducing pressure.

<table>
<thead>
<tr>
<th>Increasing of reliability and resource</th>
<th>Development of economical constructions</th>
</tr>
</thead>
<tbody>
<tr>
<td>Development of optimal construction of gas removing and sodium filling AEL block</td>
<td>Development of “trap”-type rod, increasing of absorption effect due to local softening spectrum</td>
</tr>
<tr>
<td>Search for radiation-resistant constructional materials</td>
<td>Development of dual-purpose CPS rods - producers radioactive isotopese</td>
</tr>
<tr>
<td>Changing tube construction</td>
<td></td>
</tr>
<tr>
<td>Renouncement of using of activating absorbers in standard rods</td>
<td>Development of repeated using enriched boron technology</td>
</tr>
<tr>
<td>Design of modern rods: resource - 550 days fluence - 3$\times$10$^{27}$n/m$^2$ dose - 77 dpa</td>
<td></td>
</tr>
</tbody>
</table>

Scheme 1.
of helium, which release from boron carbide under irradiation (fig.1).

The construction of sealed AEL is shown in fig.2. The operating life of sealed AEL with variously enriched boron carbide depends on pellet swelling and gas release. Gas outlet from boron carbide isn’t more than 20% of helium, produced at burnup of boron atoms up to 10% and is compensated by creating of free volumes in AEL, but this reduce the part of the absorber in the rod and its efficiency. Swelling of the boron carbide are taken into account by sufficient radial and axial gaps within the AEL at the condition of 0.5% geometry size changing per 1% boron burnup.

Fig.1. Construction of cassette-type regulating rod.
Sealed AEL with boron carbide don’t provide enough long operating life, especially at high intensity burnup.

2. Investigations of the radiation behavior of these AELs show, that sealed AEL with $B_4C$ can be recommended only at small boron burnup. At further improvement of CPS rods constructions for BN-350, BN-600 reactors, to prolong their operative life, constructions with so-called slot filters were used for filling of the inner volume of AEL by sodium coolant.
and removing of gas products to coolant are carried out through the same filter (fig.3).

The strength factor for cladding of unsealed AEL is considerably more, than for sealed ones, because helium pressure is absent, but joints and carrying constructions are the same in both cases. Increasing of radial gaps between absorber pellets and cladding are made without decreasing of pellet’s diameter due to changing the cladding’s wall to 0.7 mm, because the cladding aren’t affected internal pressure as compared with sealed ones, and the value of the radial gap allows to guarantee the pel-

Fig. 3. Unsealed absorbing element.
lets swelling to 2.3% geometry size changing per 1% burnup of boron-10 atoms at 6 month campaign or to 1.2% per 1% burnup of boron-10 atoms at 12 month campaign.

The presence of sodium layer between the cladding and pellets provides the stable working temperature of the absorber, approximately corresponding to minimum swelling of boron carbide. Using the unsealed AELs with the boron carbide and sodium layer are provided to increase the operating life in 1.5...3 times. The operating experience these AELs showed, that the main limit of their operating life is the interaction between boron carbide and cladding through the sodium layer, however the interaction speed allows to provide the operating life of unsealed AEL up to 3...4 years.

It was a new direction in the world practice for fast sodium reactors as foreign rod designs for these reactors were directed to develop gas removing systems without sodium getting into the AELs. The next operating experience of AEL showed that this design fully justified itself.

At present unsealed AEL with slot filters, containing the enriched $B_4C$, are used in standard CPS rods in BN-350 (rods CR, SR, TC) and BN-600 (rods SR, SR-L, CR) reactors.

This construction based on new constructional materials ChS-68 and EP-450 (ferrite-martensite type) are considered as promising for future CPS rods for BN-350 and BN-600 reactors.

Absorber in CPS rods for BN-350, BN-600 reactors must be possess rather high physical efficiency at given size of rod and keeps it during all operating life not less than allowable level.

Absorbing materials ising for the rods must be compatible with the constructional materials in whole operating temperature range and emer-
gency conditions and also to possess the high resistance to corrosion in sodium of the first contour to have minimal radiation destruction under high neutron flux. In the sealed and unsealed CPS rods for the BN-350, BN-600 reactors boron carbide enriched with boron-10 isotope up to 80-85% pellets were used as absorbing materials. Pellets are prepared by hot powder pressing in free of the protective atmosphere or in vacuum. Boron carbide powder were prepared by the element synthesis.

As the technology of the hot pressing is improved the boron content in pellets of synthesized boron carbide was increased from 68% mas. to 76%, and the physics density was decreased from 2.41 g/cm$^3$ to 2.19 g/cm$^3$. In this case minimal density becomes 1.8 g/cm$^3$, maximum - 2.6 g/cm$^3$.

3. At the development of regulating units it should be considered the following base conception for the fast power reactor regulation system. In this case two main groups are formed:

- regulating units operating outside of core;
- regulating units locating in the core during reactor operation.

This classification allows to select the most favorable absorbing materials and constructions for the different regulating units.

Scram rods (SR) and temperature compensation (SR-TC) assign to the first group. For SR, SR-TC rods it is advisable to use the materials and constructions providing the maximum efficiency of the assembly design with the enriched (80-90%) boron carbide and trap-type constructions. In AEL of trap-type rods the combinations $Eu_2O_3$ or $B_4C$ with effective moderating materials based on hydrides (zirconium, titanium, hafnium, europium) are used. In this case the local softening of neutron spectrum in the rod volume is the result of the neutron moderation on hydrogen
nuclei that are contained in metal hydride and the probability of the neutron absorption is increased.

The construction of such type rod allows significantly to increase the rod efficiency, to reduce the expenses of high effective costly absorbing materials. The using of the metal hydrides in particular zirconium hydride as moderator and europium oxide, boron carbide (enriched) as absorber was found to be the most proficient. Experimental investigations of rods showed, that for trap-type SR, SP-CT for BN-350, BN-600 reactors the efficiency is higher by 10-15% than for cassette-type rods with enriched boron carbide (80%) and absorbing cladding. In this case the trap-rod demands the enriched boron less in ≈ 1.9 times than for identical size cassette-type rods.

In such a manner for the trap-type rod the efficiency of enriched boron carbide using more than twice as larger as that for cassette-type rods. It should be noted that due to increased specific efficiency of the absorbing materials in trap-type rods the burnup rates of nuclei absorber is increased. The obtained results show:

- efficiency of trap-type rod with europium oxide and zirconium hydride 73 mm in diameter is 0.8-0.85 of the efficiency of cassette-type rods with 80% enriched boron carbide;

- trap-type rods based on enriched boron carbide allow to have the efficiency, impracticable for cassette-type construction even with maximum permissible enrichment by boron-10;

- to provide the prolonged operating life of the rod at the relative decreasing of absorber swelling and keeping the high efficiency by mean of generation of the optimal neutron spectrum in the rod volume.
The construction of trap-type rods, containing unsealed absorbing elements with boron carbide enriched with isotope B-10 up to 80-90% and zirconium hydride with special hydrogen confined coating using as a moderator, was successfully approbated in standard cells of SR, CR-TC rods for BN-350, BN-600 reactors (fig.4).

Fig.4. Cross section of the “trap”-type rod.

The operating experience on the study of neutron-physics characters of trap-type rods led to necessity of further development its construction for decreasing of the cost SR rods and increasing its working capacity. The above discussion, the relatively short period of the work of SR trap-type units is mainly connected with boron-10 burnup. The most part of boron-10 burns out under influence of thermal neutrons. It is indifferent to the rod efficiency where thermal neutrons are absorbed. Therefore in the rod construction the second absorber is used to screen off boron-10 against thermal neutrons of the moderator zone. The such absorber must possess rather high cross-section of thermal neutron absorption to absorb
them in the layer of some millimeters thickness and good constructional properties (fig 5). The heat resistance (hafnium, rhenium, tantalum) and rareearths oxides ($Eu_2O_3$) can be used as such materials.

Hafnium is the best suited as the second absorber for the trap-type SR due to its nuclear-physics properties: melting temperature - 2200 $^0C$, density - 13.1 $g/cm^3$, quantity of atoms - 0.044 $10^{24}$ in $cm^3$, cross-section of thermal neutron absorption - 93 barn.

Simple estimates show, that the optimum layer thickness of metallic hafnium, enough for absorbing of 90% thermal neutron, is equal to 4 mm. It should be noted, that in this case the total quantity of absorption in the boron absorbing element is decreased. This value is about 30-50%. The main advantage of developed trap-type rod is significantly more operating life, about ten fold decreased of the nonuniformity of neutron absorption in boron elements, significant decreasing of boron-10 burnup and also the

Fig. 5. Cross section of the “trap”-type rod with the titanium screen.
rod efficiency is as good as unsealed standard PS rod with enriched boron for BN-600 reactor.

The developed trap-type control rod consists from the parts, which contain neutron moderator - zirconium hydride and two neutron absorbers in the center. AEL with enriched boron carbide are mounted around the periphery, and the cylindrical graphite screen is placed between the moderator and AEL.

The compensating rods (CR) and automatic regulating rods (AR) belong to the second group. \((n, \gamma)\) absorbing materials based on europium oxides have been chosen as the material for CR and AR packet type rods with sealed AEL. This AEL make possible to obtain the maximum permissible resource of rods, which is limited, mainly by working capacity of constructional materials due to insignificant radiative destruction of \((n, \gamma)\) absorbers.

The investigations of radiative behavior of such types AEL with the absorber pellets and also pulled-type AEL, obtained by the dragging of the cladding together with powder filler, showed the high reliability of these AELs and confirmed the rightful and the promising of such industrial goods.

Sealed AELs based on europium oxide pellets are successfully used in standard rods (AR,CR-TC) for BN-600 reactor (fig.2). Sealed pulled type AELs are used in the standard rods (AR) of BN-600 reactor (fig.6).

At present promising designs of AR, CR-TC rods are rods, fulfilling the dual function - regulating of reactor power and production of the radioactive isotope Cobalt-60 (fig.7)

Developed rods consist of the external absorbing coating and the inner zone, containing an effective neutron moderator (zirconium hydride) and
Cladding

Absorbing material
(europium oxide + molybdenum or tungsten)

Nickel

Fig. 6. "Pulled"-type unsealed absorbing element.

Absorbing neutron material $\text{Eu}_2\text{O}_3 + \text{Co}$

Moderator zirconium hydride

Ampoules with Cobalt-59

Fig. 7. Cross section of dual-purpose absorbing rod - regulating of the reactor power and the production of radioactive isotopes.
an ampoule with the start Cobalt-59 isotope. The ampoules with the start Cobalt-59 by their geometric dimensions correspond the standard gamma - sources with Cobalt-60.

In such rods neutron absorbing materials based on europium oxide pellets and cobalt may be used for production of cores for radionuclide hard damma-irradiation sources of high activity. This material allows to use it in the nuclear reactor regulating units, providing the production of hard gamma radiation sources and simultaneously decreasing the production expenses.

Due to optimization of the chemical composition there is provided \( \approx 1.5 \) times more specific activity of material, comparing with that of pure cobalt and \( \approx 1.2 \) times more comparing with europium oxide while having the equal irradiation time.

Supplies manufactured of this material have a high mechanical strength and are easy for machine processing. The use of the AR, CR-TC rods provides:

- high efficiency due to ratio between absorbing material and the effective moderator in the internal rod’s volume;
- simultaneous production of Cobalt-60 with the activity not less than 100 Cu/g;
- high speed of Cobalt-60 production due to variation of correlation between moderator’s and ampoule’s with cobalt volumes, and achieving the optimum neutron spectrum in internal rod’s volume.

At present two such rods for BN-600 reactor were designed and produced.
<table>
<thead>
<tr>
<th>Characteristics</th>
<th>CPS rods modifications</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>SR</td>
</tr>
<tr>
<td>AEL-type</td>
<td>Sealed Sealed</td>
</tr>
<tr>
<td>Absorbing material</td>
<td>B₄C (80% B-10)</td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
<tr>
<td>Moderator</td>
<td>-</td>
</tr>
<tr>
<td>Setting operating time</td>
<td>180 360 330 360</td>
</tr>
<tr>
<td>(eff. days)</td>
<td></td>
</tr>
<tr>
<td>Radiative isotope</td>
<td>-</td>
</tr>
<tr>
<td>production</td>
<td></td>
</tr>
<tr>
<td>Maximum factual</td>
<td>130 360 330 360</td>
</tr>
<tr>
<td>operating time</td>
<td>(eff. days)</td>
</tr>
</tbody>
</table>

Note: At using as SR in the lower part of the assembly it can be placed ampoules for irradiation of constructional materials. (Technical documentation was performed).
EXPERIENCE OF THE BN-600 REACTOR CONTROL RODS DEVELOPMENT

OKBM, Nizhny Novgorod,
Russian Federation

Abstract

During the BN-600 reactor operation (1980-1994), the control rod design was modified with a view to improve reliability and lifetime. Control rod updating was carried out in two stages. The design of control rods modified during the first stage (1981-1985), with an aim to reduce deformation of the main units and exclude their jamming in guide sleeves, and the modified design of the control sleeves, allowed a contact between control rods and sleeves in the core area to be excluded. 350 eff. lifetime days were reached as a result of the first updating of control assemblies. At the second stage (1986-1994), the structural materials of absorber pin claddings, rod wrappers and sleeves, as well as the application of europium oxide instead of natural boron as absorber in compensating and control rods were optimized. As a result of the second updating, control assemblies lifetime was extended to approx, 500 eff. days.

1. INTRODUCTION

BN-600 reactor control assemblies (CPS rods and sleeves) were designed in accordance with their functional destination. Composition and main operating performances of these assemblies are presented in Table 1. When developing of control assemblies experience of similar products operation on BOR-60 and BN-350 reactors was taken into account. Russian specialists presented results about their operation on IAEA IWGFR meeting (1,2).

Special report on results of BN-600 reactor control assemblies pilot set and description of disadvantages removal was done on the seminar (3).

During BN-600 reactor operation (1980-1994 years) to enhance reliability and lifetime of control assemblies changes are made in their design.

2. DESIGN OF PILOT SET

Principal design of pilot set (1980-1982 years) of BN-600 reactor control assemblies is shown in Fig-s 1-4. Their main geometric sizes are given in Tables 2-5 and operation conditions and results of post-reactor examinations - in Table 6. Measurement of spent assembly geometry and later investigations showed that their lifetime is limited by deformation of hinge joint parts of CPS rods and guide sleeve wrapper placed in core during power operation.

The main reason of intensive deformation of assemblies is high temperature level (500-550°C) in massive parts of CPS rods hinge joint (~24 mm) corresponding to the maximum of temperature dependence of irradiation swelling of X18H10-type steel.
### TABLE 1. COMPOSITION AND MAIN OPERATING PERFORMANCES OF BN-600 REACTOR CONTROL ASSEMBLIES

<table>
<thead>
<tr>
<th>Name</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Shutdown rod</td>
</tr>
<tr>
<td>1. Number of CPS rods (sleeves) involved in BN-600 reactors CPS</td>
<td>5</td>
</tr>
<tr>
<td>2. Place of rod absorber arrangement relative to core at reactor power operation</td>
<td>Absorber is withdrawn from core</td>
</tr>
<tr>
<td>3. Maximum velocity of rod movement, m/c</td>
<td>1</td>
</tr>
<tr>
<td>4. Feature of CPS channel (sleeve and CPS guide tube)</td>
<td>Complete</td>
</tr>
</tbody>
</table>

### 3. FIRST STAGE OF MODERNIZATION

Basing on results of pilot set operation of BN-600 reactor control assemblies two stages of them modernization were carried out.

Tables 2-5 contain main geometric sizes of control rods after first modernization (1981-1985 years), and Table 6 contains operation conditions and results of post-reactor examinations.

Thickness of massive parts of CPS rods hinge joints was reduced down to 10 mm, conditions of their cooling were improved, height of pads contacting with guide sleeve was decreased.

Wrapper diameter of bottom joined link and diameter of entrance nozzle spacing pad were reduced additionally in control rods and shut down rods. Removal of diffusion chromium-nitride coating from hinge parts of CPS rods and from of CPS guide sleeves wrappers increased plasticity of structural material and stability of their deformation.
Fig. 1. Shut down rod and guide sleeve
Fig. 2. Shut down rod - loop and guide sleeve
Fig.3. Control rod and guide sleeve
Fig. 4. Compensating rod and guide sleeve
TABLE 2. MAIN GEOMETRIC SIZES OF SHUT DOWN ASSEMBLY

<table>
<thead>
<tr>
<th>Name</th>
<th>Pilot set</th>
<th>First modernization</th>
<th>Second modernization</th>
</tr>
</thead>
<tbody>
<tr>
<td>Size of rod wrapper tube, mm</td>
<td></td>
<td>Ø 73x1</td>
<td></td>
</tr>
<tr>
<td>Outer diameter of rod in the hinge region</td>
<td></td>
<td>Ø 74</td>
<td></td>
</tr>
<tr>
<td>Sleeve configuration</td>
<td>Round</td>
<td></td>
<td>Hexahedral</td>
</tr>
<tr>
<td>Inner diameter of sleeve, mm</td>
<td>76</td>
<td>78 (outside the core boundary)</td>
<td></td>
</tr>
<tr>
<td>Inner size of hexahedral wrapper within core, mm</td>
<td>-</td>
<td>92</td>
<td></td>
</tr>
<tr>
<td>Rod length, mm</td>
<td>2100</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Number of absorber pins in rod, pieces</td>
<td>7</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Size of absorber pin cladding, mm</td>
<td>Ø 23x0.7</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pellet diameter, mm</td>
<td>20.4</td>
<td>19.6</td>
<td></td>
</tr>
<tr>
<td>Absorber</td>
<td>B₄C-80% for B-10</td>
<td>B₄C-92% for B-10</td>
<td></td>
</tr>
<tr>
<td>Leaktightness of absorber pin</td>
<td>Non-leaktight</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Structural material of rod</td>
<td>X18H10-type austenitic steel</td>
<td>IX13M2ΦP-type ferritic steel</td>
<td></td>
</tr>
<tr>
<td>Structural material of absorber pins</td>
<td>X16H15-type austenitic steel</td>
<td>X16H15+Ti-type austenitic steel in cold deformed state</td>
<td></td>
</tr>
<tr>
<td>Structural material of sleeve</td>
<td>X16H36-type austenitic dispersion-hardening steel</td>
<td>X16H11-type austenitic steel in cold deformed state</td>
<td></td>
</tr>
<tr>
<td>Coating of head, hinges of rods and sleeve tubes</td>
<td>Diffusion-chromium-nitride</td>
<td>IX13M2ΦP-type ferritic steel</td>
<td></td>
</tr>
</tbody>
</table>


**TABLE 3. MAIN GEOMETRIC SIZES OF SHUT DOWN ASSEMBLY - LOOP**

<table>
<thead>
<tr>
<th>Name</th>
<th>Pilot set</th>
<th>First modernization</th>
<th>Second modernization</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Size of rod wrapper tube, mm</td>
<td>Ø 73x1</td>
<td></td>
<td></td>
</tr>
<tr>
<td>2. Outer diameter of rod in the hinge region</td>
<td>Ø 74</td>
<td></td>
<td></td>
</tr>
<tr>
<td>3. Sleeve configuration</td>
<td>Round</td>
<td>Hexahedral</td>
<td></td>
</tr>
<tr>
<td>4. Inner diameter of sleeve, mm</td>
<td>76</td>
<td>78 (outside the core</td>
<td></td>
</tr>
<tr>
<td>5. Inner size of hexahedral wrapper within core, mm</td>
<td>23</td>
<td>92</td>
<td></td>
</tr>
<tr>
<td>6. Rod length, mm</td>
<td>2100</td>
<td></td>
<td></td>
</tr>
<tr>
<td>7. Number of absorber pins in rod, pieces</td>
<td>7</td>
<td>4</td>
<td></td>
</tr>
<tr>
<td>8. Size of absorber pin eladding, mm</td>
<td>Ø 23x0.7</td>
<td></td>
<td></td>
</tr>
<tr>
<td>9. Pellet diameter, mm</td>
<td>20.4</td>
<td>19.6</td>
<td></td>
</tr>
<tr>
<td>10. Absorber</td>
<td>B$_4$C-19.8% for B-10</td>
<td></td>
<td></td>
</tr>
<tr>
<td>11. Leaktightness of absorber pin</td>
<td>Non-leaktight</td>
<td></td>
<td></td>
</tr>
<tr>
<td>12. Structural material of rod</td>
<td>XI8H10-type austenitic steel</td>
<td>XI13M25ΦP-type ferritic steel</td>
<td></td>
</tr>
<tr>
<td>13. Structural material of absorber pins</td>
<td>X16H15-type austenitic steel</td>
<td>X16H15+Ti-type austenitic steel in cold deformed state</td>
<td></td>
</tr>
<tr>
<td>14. Structural material of sleeve</td>
<td>X16H36-type austenitic dispersion-hardening steel</td>
<td>X16H11-type austenitic steel in cold deformed state</td>
<td></td>
</tr>
<tr>
<td>15. Coating of head, hinges of rods and sleeve tubes</td>
<td>Diffusion-chromium-nitride</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
TABLE 4. MAIN GEOMETRIC SIZES OF CONTROL ROD ASSEMBLY

<table>
<thead>
<tr>
<th>Name</th>
<th>Pilot set</th>
<th>First modernization</th>
<th>Second modernization</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Size of rod wrapper tube, mm</td>
<td></td>
<td>Ø 73x1</td>
<td></td>
</tr>
<tr>
<td>2. Outer diameter of rod in the hinge region</td>
<td></td>
<td>Ø 74</td>
<td></td>
</tr>
<tr>
<td>3. Sleeve configuration</td>
<td>Round</td>
<td>Hexahedral</td>
<td></td>
</tr>
<tr>
<td>4. Inner diameter of sleeve, mm</td>
<td>76</td>
<td>78 (outside the core boundary)</td>
<td></td>
</tr>
<tr>
<td>5. Inner size of hexahedral wrapper within core, mm</td>
<td>-</td>
<td>92</td>
<td></td>
</tr>
<tr>
<td>6. Rod length, mm</td>
<td>2100</td>
<td></td>
<td></td>
</tr>
<tr>
<td>7. Number of absorber pins in rod, pieces</td>
<td>31</td>
<td>4</td>
<td></td>
</tr>
<tr>
<td>8. Size of absorber pin cladding, mm</td>
<td>Ø 9.5x0.5</td>
<td>Ø 23x0.7</td>
<td></td>
</tr>
<tr>
<td>9. Pellet diameter, mm</td>
<td>8.2</td>
<td>19.6</td>
<td></td>
</tr>
<tr>
<td>10. Absorber</td>
<td>Eu$_2$O$_3$</td>
<td></td>
<td>B$_4$C-19.8% for B-10</td>
</tr>
<tr>
<td>11. Leaktightness of absorber pin</td>
<td>Leaktight</td>
<td></td>
<td>Non-leaktight</td>
</tr>
<tr>
<td>12. Structural material of rod</td>
<td>XI8H10-type austenitic steel</td>
<td>XI13M2ΦP-type ferritic steel</td>
<td></td>
</tr>
<tr>
<td>13. Structural material of absorber pins</td>
<td>XI6H15-type austenitic steel</td>
<td>XI6H15+Ti-type austenitic steel in cold deformed state</td>
<td></td>
</tr>
<tr>
<td>14. Structural material of sleeve</td>
<td>XI6H36-type austenitic dispersion-hardening steel</td>
<td>XI6H11-type austenitic steel in cold deformed state</td>
<td>XI13M2ΦP-type ferritic steel</td>
</tr>
<tr>
<td>15. Coating of head, hinges of rods and sleeve tubes</td>
<td>Diffusion-chromium-nitride</td>
<td></td>
<td>-</td>
</tr>
<tr>
<td>Name</td>
<td>Pilot set</td>
<td>First modernization</td>
<td>Second modernization</td>
</tr>
<tr>
<td>----------------------------------------------------------------------</td>
<td>-----------</td>
<td>---------------------</td>
<td>----------------------</td>
</tr>
<tr>
<td>1. Size of rod wrapper tube, mm</td>
<td>Ø 89x1.5</td>
<td>Ø 85x1</td>
<td></td>
</tr>
<tr>
<td>2. Outer diameter of rod in the hinge region</td>
<td></td>
<td>90</td>
<td></td>
</tr>
<tr>
<td>3. Sleeve configuration</td>
<td>Round</td>
<td></td>
<td>Hexahedral</td>
</tr>
<tr>
<td>4. Inner diameter of sleeve, mm</td>
<td>92</td>
<td>82 (outside the core boundary)</td>
<td></td>
</tr>
<tr>
<td>5. Inner size of hexahedral wrapper within core, mm</td>
<td>-</td>
<td>92</td>
<td></td>
</tr>
<tr>
<td>6. Rod length, mm</td>
<td></td>
<td>2874</td>
<td></td>
</tr>
<tr>
<td>7. Number of absorber pins in rod, pieces</td>
<td>48</td>
<td>8</td>
<td></td>
</tr>
<tr>
<td>8. Size of absorber pin cladding, mm</td>
<td>Ø 9.5x0.5</td>
<td>Ø 23x0.7</td>
<td></td>
</tr>
<tr>
<td>9. Pellet diameter, mm</td>
<td>8.2</td>
<td>20.4</td>
<td>19.6</td>
</tr>
<tr>
<td>10. Absorber</td>
<td>Eu₂O₃</td>
<td>B₄C-19.8% for B-10</td>
<td></td>
</tr>
<tr>
<td>11. Leaktightness of absorber pin</td>
<td>Leaktight</td>
<td></td>
<td>Non-leaktight</td>
</tr>
<tr>
<td>12. Structural material of rod</td>
<td>XI8H10-type austenitic steel</td>
<td>XI13M25ΦP-type ferritic steel</td>
<td></td>
</tr>
<tr>
<td>13. Structural material of absorber pins</td>
<td>XI6H15-type austenitic steel</td>
<td>XI6H15+Ti-type austenitic steel in cold deformed state</td>
<td></td>
</tr>
<tr>
<td>14. Structural material of sleeve</td>
<td>X16H36-type austenitic dispersion-hardening steel</td>
<td>X16H11-type austenitic steel in cold deformed state</td>
<td>XI13M25ΦP-type ferritic steel</td>
</tr>
<tr>
<td>15. Coating of head, hinges of rods and sleeve tubes</td>
<td>Diffusion-chromium-nitride</td>
<td></td>
<td>-</td>
</tr>
</tbody>
</table>
TABLE 6. IRRADIATION CONDITIONS OF ASSEMBLY IMPORTANCE UNITS

<table>
<thead>
<tr>
<th>Assembly type</th>
<th>Set name</th>
<th>CPS rods</th>
<th>Guide sleeves</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>lifetime, eff.days</td>
<td>dpa</td>
</tr>
<tr>
<td>Shut down rod</td>
<td>Pilot</td>
<td>200</td>
<td>37</td>
</tr>
<tr>
<td>(Shut down rod - loop)</td>
<td>1-st stage of modernization</td>
<td>330</td>
<td>49</td>
</tr>
<tr>
<td></td>
<td>2-nd stage of modernization</td>
<td>500</td>
<td>67</td>
</tr>
<tr>
<td></td>
<td>Pilot</td>
<td>200</td>
<td>37</td>
</tr>
<tr>
<td>Control rod</td>
<td>1-st stage of modernization</td>
<td>330</td>
<td>49</td>
</tr>
<tr>
<td></td>
<td>2-nd stage of modernization</td>
<td>500</td>
<td>70</td>
</tr>
<tr>
<td></td>
<td>Pilot</td>
<td>200</td>
<td>37</td>
</tr>
<tr>
<td>Compensating rod</td>
<td>1-st stage of modernization</td>
<td>330</td>
<td>49</td>
</tr>
<tr>
<td></td>
<td>2-nd stage of modernization</td>
<td>500</td>
<td>65</td>
</tr>
</tbody>
</table>

Importance units of assemblies:
Shut down rod - spacing pad of entrance nozzle
Control and compensating rods - parts of bottom hinge joint
Guide sleeve - core central plane
Design of CPS guide sleeves was simplified by making them using hexahedral wrapper tube and cylindrical bushing placed above core in sleeves of shut down rods and control rods or under core in sleeves of compensating rods. New sleeve design enabled to increase minimum gap between rod and sleeve, to decrease the gap height, having provided additional gap between CPS rods and their sleeves within core. Therefore flowrate value of coolant, cooling absorber pins bundle reduced, and temperature value of absorber pins cladding increased correspondingly by approx. 30°C.

Change of compensating rod absorber was done from absorber on the base of euoropium oxide, having high decay heat releases and induced radioactivity to natural boron carbide.

As a first modernization result design lifetime (350 eff. days) for control assemblies was achieved (real lifetime amounts to 330 eff. days).

4. SECOND STAGE OF MODERNIZATION

Principal design of BN-600 reactor control assemblies on second stage of modernization (1986-1994 years) is showed in Fig-s 1-4. Tables 2-5 contain their main geometric sizes and Table 6 - operation conditions and results of post-reactor examinations.

Radiation-resistant IX13M2БФР-type ferrite steel was used for manufacturing control assembly wrapper tubes, X16H15+Ti c.d. austenitic steel - for absorber pin claddings.

Absorber on the base of euoropium oxide was replaced by natural boron carbide in control rods and the size of absorber pin cladding (Ø 23x0.7 mm) was unified.

Hinge joints along length of rod absorber section were excluded in shut down rods.

Design of second modernization sleeves is similar to that of first modernization sleeves (besides structural materials).

Change of core to new modernized loading (core height was changed from 750 mm to 1000 mm) was carried out simultaneously with modernizing of BN-600 reactor control assemblies. Neutron fluence and fuel element thermal load decreased in modernized core. It resulted in temperature decrease of rod importance unit down to 400°C and neutron fluence by ~20%.

Use of new structural materials and introduction of above listed changes of assembly design enabled to increase control assembly lifetime up to 500 eff. days.

CONCLUSION

Main results of BN-600 reactor control assembly operation (1980-1994 years) are considered and main stages of their modernization directed to increase of realibility and lifetime of control members are given.

During first stage of modernization (1981-1985 years) changes are made in CPS rod design that enabled to decrease deformation of main units, and improved design of CPS sleeves provided gap increase between rod and sleeve within core. As result of first modernization the lifetime of control assemblies was ~350 eff. days (~66 dpa).

Change of structural materials control assembly wrapper tubes - IX13M2БФР-type steel and of absorber pin cladding - X16H15+Ti-type steel in cold-deformed state.
was done for second modernization (1986-1994 years). As result of this modernization the lifetime of control assemblies was ~500 eff. days (~80 dpa).

REFERENCES

1. Материалы совещания специалистов МРГБР МАГАТЭ "Поглощающие материалы и стержни регулирования быстрых реакторов"
Димитровград, СССР, 1973 г.

2. Материалы совещания специалистов МРГБР МАГАТЭ "Поглощающие материалы и стержни регулирования быстрых реакторов"
Обнинск, СССР, 1983 г.

3. Материалы британо-советского семинара "Поведение облученных конструкционных материалов и элементов активных зон реакторов БН".
Великобритания, 1990 г.
CONTROL ASSEMBLY TO BE USED IN CEFR

XIE GUANGSHAN, ZHANG RUXIAN
China Institute of Atomic Energy

WANG YONGLAN
Xian Jiaotong University

LI SHIKUN
China Institute of Atomic Energy
China

Presented by R. Voznesenski

Abstract

This paper describes the structure and design data selected for control assembly in CEFR in detail. The control assembly of CEFR mainly consists of a guide tube and seven absorber pins. Each pin includes a section absorber material $\text{B}_4\text{C}$ of 510mm long and a stainless steel cladding, and sodium medium is filled in the gap between $\text{B}_4\text{C}$ pellet and cladding. Then the characteristic properties of the prepared $\text{B}_4\text{C}$ pellet to be used in CEFR are presented, and further researches to be done for $\text{B}_4\text{C}$ pellet are also given in the paper.

1. INTRODUCTION

China Experimental Fast Reactor (CEFR) contains eight control assemblies in which, based on the conceptual design, two control assemblies serve as safety rod to perform rapid shutdown during off-normal conditions, four as compensation rod for temperature effect and fuel burn-up and two as regulator rod to perform neutronic start-up and shutdown capability and control power level for normal operations. Although these control assemblies will perform different functions, their structure is quite same and their positions in the core are only different. These control assemblies all are independent parts, and each control assembly occupies a position corresponding to a fuel assembly in the core.

Boron carbide $\text{B}_4\text{C}$ is selected as absorber material. The $\text{B}_4\text{C}$ pellet has preliminary been prepared and its main properties have been measured by Xian Jiaotong university. Thermal conductivity of the $\text{B}_4\text{C}$ pellet is testing in China Institute of Atomic Energy at present.

2. STRUCTURE

The control assembly consists of a fixed guide part and a movable part containing absorber pins, as shown in Fig 1.

The guide part includes a guide tube with inside of circular and outside of hexagon, a operating handle and a positioning leg. The guide tube is made of Austenitic 316(Ti) stainless steel, i.e. minor titanium is added in standard AISI316 stainless steel. Its out flat-to-flat has a distance of 59.0mm and inner circular has a diameter of 55.0mm.
The movable absorber part consists of a circular duct, seven absorber pins, positioning grids and a handling head. There are orifice sets on the surface of the upper and lower positioning grids, to make most coolant pass through the absorber pin bundle and provide best cooling.

One control assembly contains seven pins which are arranged in a trigonal pitch-16.0mm. The wire wraps are used as the radial separator for the absorber pins within an assembly, to keep up the coolant channel. These absorber pins are fixed on the upper and lower grids, then lower grid is fixed on the leg of this assembly. So the axial position of these absorber pins is determined in the assembly.

The absorber pin is designed as sealed structure. It includes a cladding made of 316(Ti)S.S, $\text{B}_4\text{C}$ pellets of 510mm long, hold-down spring, a helium gas plenum and upper and lower end-plug (see Fig 1). Table 1 gives main design data of the control assembly for CEFR.

A wider gap (normal diametric gap of 0.8mm) is selected between the cladding and $\text{B}_4\text{C}$ pellet and sodium medium is filled in this gap. The gap accommodates the $\text{B}_4\text{C}$ matrix swelling by irradiation, the sodium can get best heat transfer behaviour in the gap, so the $\text{B}_4\text{C}$ pellet temperature decreases.

* A vented absorber pin is also selected as a candidate for CEFR.
The absorption of neutrons by $\text{B}_4\text{C}$ leads to the $(n, \alpha)$ reaction which yields both lithium and helium atoms larger than the original boron atom, though some of the helium is released, all lithium products and the part of helium retained cause $\text{B}_4\text{C}$ matrix swelling. To accommodate this swelling, besides lower $\text{B}_4\text{C}$ pellet density is selected, the wider gap is designed between the cladding and pellet. Assuming that CEFR control assembly operates for 402 full power days in this reactor, maximum capture level in $\text{B}_4\text{C}$ is higher up to $1.0 \times 10^{22} (n, \alpha)/cm^3$, corresponding diametric swelling of 5% ($\Delta D/D$).

The number of the helium atoms yielded by $\text{B}_4\text{C}$ absorption neutrons is approximately the same as the number of neutrons absorbed, and helium generation correlates directly with boron burnup. Helium gas release from the $\text{B}_4\text{C}$ matrix is temperature dependent. For the operating temperature of CEFR control pin, the percentage released is assumed to be 30% (average), when the plenum height is 200mm, helium gas pressure builds up 8.8MPa.

When the surface temperature of the $\text{B}_4\text{C}$ pellet is more than 700°C, FeB or Fe$_2$B is formed on the inner surface of cladding$^{(3)}$, as a result of cladding material reaction with $\text{B}_4\text{C}$ pellet, with aggravating this reaction in sodium.
Table 2 comparison of examination and design values for chemical composition of B₄C pellet

<table>
<thead>
<tr>
<th>constituent</th>
<th>unit</th>
<th>examination values</th>
<th>design values</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>4.0</td>
<td>3.8</td>
</tr>
<tr>
<td>total boron + total carbon</td>
<td>wt%</td>
<td>99.04</td>
<td>98.92</td>
</tr>
<tr>
<td>total boron</td>
<td>wt%</td>
<td>78.23</td>
<td>77.40</td>
</tr>
<tr>
<td>Nitric acid-soluble boron</td>
<td>wt%</td>
<td>0.12</td>
<td>0.11</td>
</tr>
<tr>
<td>water-soluble boron</td>
<td>wt%</td>
<td>0.07</td>
<td>0.05</td>
</tr>
<tr>
<td>B₂O₃</td>
<td>ppm</td>
<td>492</td>
<td>475</td>
</tr>
<tr>
<td>chloride</td>
<td>ppm</td>
<td>50</td>
<td>46</td>
</tr>
<tr>
<td>fluoride</td>
<td>ppm</td>
<td>23</td>
<td>24</td>
</tr>
<tr>
<td>Ca</td>
<td>ppm</td>
<td>130</td>
<td>134</td>
</tr>
<tr>
<td>Si</td>
<td>ppm</td>
<td>1600</td>
<td>1435</td>
</tr>
<tr>
<td>Al</td>
<td>ppm</td>
<td>118</td>
<td>135</td>
</tr>
<tr>
<td>Mg</td>
<td>ppm</td>
<td>105</td>
<td>109</td>
</tr>
<tr>
<td>F</td>
<td>ppm</td>
<td>159</td>
<td>143</td>
</tr>
<tr>
<td>free C</td>
<td>wt%</td>
<td>0.16</td>
<td>0.19</td>
</tr>
<tr>
<td>moisture (water)</td>
<td>ppm</td>
<td>135</td>
<td>124</td>
</tr>
</tbody>
</table>

environment. When sodium is filled in the gap between B₄C pellet and the cladding, the temperature of the B₄C pellet is substantially decreased. The maximum surface temperature of B₄C pellet for CEFR is not expected more than 550 °C, so this reaction products (FeB or Fe₂B) can be a small quantity and not obviously effect on the effective thickness of cladding.

3. PROPERTIES OF B₄C PELLET

The B₄C powder* has been prepared using higher pure boron and carbon powder by directly synthetic technique, then by hot pressing and sintering, finally the B₄C pellet is finished into the geometric size and surface specification required.

3.1 Chemical composition of finished B₄C pellet

Chemical composition prepared B₄C pellet had been analyzed, the examination result corresponds with the standard GB5152-85 of the People's Republic of China and ASTM C791-83. The comparison between examination and design values is listed in table 2.

Although from table 2, we can see that three of stoichiometry for B₄C (i.e., with three B/C ratios—3.8, 4.0 and 4.2) have been used in the property investigation for B₄C pellet, a stoichiometry of 4.0 is only selected in this design for CEFR.

3.2 Properties of B₄C pellet

The grain size of B₄C pellet samples is observed by metallographical examination, (a) and (b) in figure 2 shown respectively microscopic examinations for two grain sizes of 5~10 μm and 10~15 μm by sintering with different hot

* Natural boron powder is used in the investigation for the preparation technique and the samples of non-irradiation test.
Fig 2 Metallograph of B₄C sample

-pressing time. It has been seen from this figure that their grain sizes are basically homogeneous, the impurities are not existence almost in B₄C pellet matrix and the form of pore appear as equiaxed, the porosity corresponds to the density measurement. The average value over the pore size is around 0.2 μm, maximum size among this pores is 3 μm.

The mechanical properties and thermal performance had been examined for the grain size of 5~10 μm for B₄C pellet sample. This characteristic data include melting point, linear thermal expansion coefficient (room temperature to 1000°C), bending strength, compression strength, young's modulus, poisson ratio, rupture toughness and thermal fatigue. These test results were listed in table 3.

4. FUTURE TEST

The characteristic data of B₄C pellet is based on the prepared sample during preliminary design phase for CEFR control assembly. Besides the optimum process technology will continue investigating, main two experimental works will be done as follows.

- Compatibility of B₄C with stainless steel cladding material. A compatibility test for B₄C with stainless steel cladding material will be investigated under sodium medium environment and simulated operation condition of CEFR control pin. The product Fe₂B and its thickness on the inner surface of cladding will be examined.

- Irradiation test of B₄C in reactor. The main purposes of this test are to obtain the resistance to swelling behaviour and release rate of helium from B₄C matrix, to evaluate the operation behaviour of the absorber pin.

5. CONCLUSION

This structure design is under the conceptual phase, although it can be changed in the detial design for the future, this structure form and main structure parameters will basically satisfy for CEFR’s requirements from the behaviour analysis of the control assembly.
Table 3  main properties data for B$_4$C pellet

<table>
<thead>
<tr>
<th>item</th>
<th>unit</th>
<th>density of sample (%T.D)</th>
<th>test temp. (°C)</th>
<th>examination value</th>
<th>average</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>melting point</td>
<td>°C</td>
<td>92</td>
<td>room temp.</td>
<td>423.0 356.0 380.0</td>
<td>383.0</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>200 389.8 363.3 375.1</td>
<td>376.1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>400 304.3 289.0 338.0</td>
<td>310.4</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>600 256.5 301.8 325.8</td>
<td>307.7</td>
</tr>
<tr>
<td>bending strength</td>
<td>MPa</td>
<td>92</td>
<td>room temp.</td>
<td>944.0 752.0</td>
<td>847.0</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>1273 1163 2143 1890</td>
<td>1617</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>3511</td>
<td>&gt;351</td>
</tr>
<tr>
<td>compression strength</td>
<td>MPa</td>
<td>92</td>
<td>room temp.</td>
<td>374 374 374 374</td>
<td>374</td>
</tr>
<tr>
<td>young’s modulus</td>
<td>GPa</td>
<td>92</td>
<td>room temp.</td>
<td>0.22 0.21 0.19 0.19</td>
<td>0.193</td>
</tr>
<tr>
<td>poisson ratio</td>
<td>/</td>
<td>92</td>
<td>room temp.</td>
<td>7.73 8.09 7.71</td>
<td>7.84</td>
</tr>
<tr>
<td>rupture toughness</td>
<td>(MPa)$^{0.5}$</td>
<td>92</td>
<td>room temp.</td>
<td>5.832</td>
<td>5.832</td>
</tr>
<tr>
<td>linear thermal</td>
<td>$10^{-6}$/°C</td>
<td>92</td>
<td>room temp.</td>
<td>140.4</td>
<td>140.4</td>
</tr>
<tr>
<td>expansion coefficient</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>thermal fatigue(R)</td>
<td>°C</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

The mechanical and thermal properties examinations of the trial-produced B$_4$C pellet show that this process technology will be adoptable and the trial B$_4$C pellet will be promising for CEFR, but further some researches for B$_4$C pellet will be done in the future.

REFERENCES

2. ASTM Standard: C751-92 standard specification for Nuclear-Grade Boron Carbide Pellets
THE CONTROL ROD MODELLING CODE REGAIN

J. TRUFFERT
Commissariat à l’Énergie Atomique (CEA),
Centre d’études de Cadarache,
DRN/DEC/SDC/LEMC,
Cedex, France

Abstract

We present here the CEA's control rod modelling code REGAIN. REGAIN is applied to modelling both normal reactor control rod operations and experimental irradiation of absorber pins.

It models long permanent periods and transient pin behaviour during shutdown. Calculations are performed for a single pin with its subchannel, including the cladding, one or more different columns and possibly a shroud. The models cover both thermal, mechanical and some chemical aspects of the problem.

Written in Ada language this code is built according to an object oriented method; these choices lead to high modularity, safety and portability. This architecture enables a clean division to specific modules: properties for different materials, thermal analysis, mechanical analysis and mathematical methods.

Adding new materials (absorber or cladding) is possible either by inheritance of material properties from parent material and slight modification or in the case of very different materials (e.g. metal, ceramic, cermet ... and including cracks) by rewriting a specific package unit: specification and body.

For boron carbide material it can deal with both existing design concepts: sodium bonded (diving bell) and helium bonded.

REGAIN is thus able to provide details of the absorber material and cladding temperatures, stresses and strains through the life of the rod. Absorber material is also calculated in some detail, giving porosity, $^{10}$B concentration and capture rate.

REGAIN is now being licensed and will be used for the calculation of the SAC PHENIX.

1. INTRODUCTION

1.1 The problem

At first it is the evaluation of the thermo-mechanical behaviour of an absorber pin depending on:

- maximum burn-up of $^{10}$B,
- radial variation and,
- axial variation of this burn-up,
- inlet temperature and flow of coolant.

But also evolution of structures for cladding and shroud:

- inelastic deformation,
- swelling,
• irradiation and thermal creep.

And evolution of absorber columns:
• volume modification by swelling and cracks,
• gap closure,
• isotopic decrease for absorber materials,
• production, retention and release of helium.

REGAIN takes in charge two pin's architectures: helium bonded (closed pin) and sodium bonded (vented pin).

For both types, calculations are made for one or several pellet stacks (possibly annular pellet) with different diameter or $^{10}$B enrichment.

### 1.2 Basic assumptions

#### 1.2.1 Geometry and axial transfer

Pin remains always an axial symmetry,
For one level all components are concentric, physical and mechanical variables: temperature, stresses, strains are only radius dependent.
Due to the high thermal conductivity of absorber materials axial thermal transfers are calculated for pellet stack.
There is a heat generation in cladding, shroud and pellets but no thermal source in the coolant.

Both normal and transient operation are treated allowing to calculate long term experimental irradiation's, operating controls rods with very slow and rapid vertical movements.

In this model the changes in the pellet dimensions and properties are calculated as the control rod moves slowly or rapidly in the reactor neutron flux, and these are used to calculate the stresses and strains in the cladding and in the pellet (including creation and closure of cracks). The cladding and the pellet are allowed to creep under stress if the pellet dimensional changes are sufficient to close the initial gap. The program also calculates the temperatures of the pellet, cladding and coolant.

### 2. FIELD

Due to modelling limitations REGAIN can be used in the following field:
• pellet diameter $[5:50]$ mm
• density of $\text{B}_4\text{C} [80:96]$%
• cladding temperature $[0:650]$ °C
• peak linear power: such as maximum temperature in $\text{B}_4\text{C} < 1500$ °C
• burn up (low density ~84%) $[0:150] \times 10^{26}$ capture/m$^3$
• burn up (high density 96%) $[0:200] \times 10^{26}$ capture/m$^3$
• lifetime $[0:1000]$ e.p.f.d.
But the program can held irradiation up to:

- density of B₄C [70 : 100] %
- cladding temperature: [0 : 800] °C
- peak linear power: such as maximum temperature in B₄C < 2300 °C
- cladding deformation [0 : 3] %
- lifetime [0 : 3000] e.p.f.d.

3. MATERIAL PROPERTIES

For standard materials: sodium (and water), stainless steel like 316 (coldworked and annealed) and boron carbide, all the properties are included in REGAIN.

It is easy to introduce slight modifications from standard material with additional data.

For new materials the "Ada mechanism of inheritance" allows the creation of a new set of properties from a parent standard material. REGAIN with this property of changeability is very suitable for new absorber pin design (even for reflector replacement).

3.1 Standard material properties

For one typical material the set of properties are bring together in an "Ada package" with explicit specifications like in Figure 1.

This allow simplifications in writing and increase legibility; for example whatever the material of one elementary objet calculation of thermal values has the same form in Ada source language see Figure 2.

4. LOCAL DESCRIPTION

4.1 Thermal and Physical

An elementary physical point is described by the following values corresponding to an "Ada structure":

For any kind of material

\[
\begin{align*}
\text{TEMPERATURE,} & \quad \text{-- K} \\
\text{DIAMETER,} & \quad \text{-- m} \\
\text{LEVEL,} & \quad \text{-- m} \\
\text{THICKNESS,} & \quad \text{-- m} \\
\text{POWER,} & \quad \text{-- kW/m}^3 \\
\text{SWELLING,} & \quad \text{-- nd} \\
\text{TANGENTIAL_CRACKING} & \quad \text{-- nd} \\
\text{AXIAL_CRACKING} & \quad \text{--nd} \\
\text{MASS_VOLUMIQUE} & \quad \text{-- kg/m}^3 \\
\text{HEAT_CAPACITY} & \quad \text{-- J/kg} \\
\text{THERMAL_CONDUCTIVITY} & \quad \text{-- W/m.K} \\
\text{POWER_SECTOR} & \quad \text{-- W}
\end{align*}
\]
And more
for steel
DOSE, -- dpa
DOSE_RATE, -- dpa/s
D_G_ON_DDOSE, -- dérivée versus la dose
K_and_ALPHA; -- irradiation creep values

for B₄C
CAPTURES, -- capt/m³
SIGMA_PHI, -- 1/s
DENSITY_for_B₄C -- nd
N_4_HE ,
N_10_B ,
N_11_B ,
N_12_C , -- mole/m³
ENRICHMENT_10_B , -- nd
HELIUM RETAINED. -- volume tpn /volume

for fluids
SPEED, -- m/s
Ps, Pd , -- static and dynamic pressures
PRANDTL,
REYNOLDS -- nd

4.2 Mechanic
An elementary mechanical point is described by the following values corresponding to an other “Ada structure”:

RADIUS, -- m
TEMPERATURE, -- K
YOUNG, -- MPa
POISSON, -- nd
ELASTIC_LIMIT, -- MPa
YIELD_STRENGTH, -- MPa
ALFA_T,
EPSILON_SWELLING,
EPSILON_PLASTIC_and_CREEP ( r, t, l ),
EPSILON_CRACKING ( r, t, l ),
EPSILON_ELASTIC ( r, t, l ), -- nd
SIGMA ( r, t, l ), -- MPa
SIGMA_EQ, -- MPa
K et ALFA -- irradiation creep,
EPSILON_PLASTIC_EQUIV, -- nd
DG_ON_DT,
CRACK_STATUS ( r, t, l ). -- boolean
with CONSTANTES_PHYSIQUES_ET_CONVERSIONS;
with PARAMETRES_MATERIAUX, UNITES_ET_TUBES;

package PROPRIETES_B4C is

use UNITES_ET_TUBES;

 TABLE_DE_COEFFICIENT:PARAMETRE_MATERIAUX.VERS_COEFFICIENTS;

function CAPACITE ( TEMP_K : TEMPERATURE ) return Joule_par_Kg_K;

function CONCENTRATION_10_B ( TENEUR_10_B : PROPORTION ) return RF_STD;

function CONDUCTIVITE ( TEMP_K : TEMPERATURE;
          DENSITE_REL : PROPORTION ) return Watt_par_metre_K;

function DILATATION ( TEMP_K : TEMPERATURE ) return RF_STD;

function ENTHALPIE ( TEMP_K : TEMPERATURE ) return RF_STD;

function MASSE_VOLUMIQUE ( TEMP_K : TEMPERATURE ) return Kg_m3;

function POISSON (TEMP_K : TEMPERATURE -- Utilise si t < T_solidus
          := CONSTANTES_PHYSIQUES_ET_CONVERSIONS.Tk0 ) return RF_STD;

function RUPTURE ( DENSITE_REL : PROPORTION ) return MPa;

function YOUNG ( DENSITE_REL : PROPORTION;
          TEMP_K : TEMPERATURE -- Utilise si t > T_solidus
          := CONSTANTES_PHYSIQUES_ET_CONVERSIONS.Tk0 ) return MPa;

-- Pour gestion des appelS a PROPRIETES_B4C

ERREUR_FORMULATION_B4C : exception;

end PROPRIETES_B4C;

FIG. 1. Boron carbide properties package specification.
with CONSTANTES_PHYSIQUES_ET_CONVERSIONS;

package body XXX_P is

procedure ACTUALISATIONS_MECANIQUES (LA_TRANCHE: in VERS_TRANCHE;
LE_DISQUE: in out VERS_DISQUE) is

DIAM: DIAMETRE;
INTERPOL_THER: BASE;
tk: TEMPERATURE;
densite: PROPORTION;

begin
  for I in LE_DISQUE'range loop

declare
  POINT_MIS_A_JOUR: POINTS renames LE_DISQUE(i);
begin
  DIAM := 2.0 * RF_STD(POINT_MIS_A_JOUR.RAYON);
  INTERPOL_THER := INTERPOLATION_TH(DIAM, LA_TRANCHE);
tk := INTERPOL_THER.T;
densite := INTERPOL_THER.DENS_REL4C;
POINT_MIS_A_JOUR.T :=
CONSTANTES_PHYSIQUES_ET_CONVERSIONS.T_CELCIUS(tk);
POINT_MIS_A_JOUR.YOUNG := RF_PRECIS( PROPRIETES XXX.YOUNG (densite, tk));
POINT_MIS_A_JOUR.POISSON := RF_PRECIS( PROPRIETES XXX.POISSON(tk));
POINT_MIS_A_JOUR.EPSILON_GONFLEMENT -- Volumique := RF_PRECIS(gonfV);
POINT_MIS_A_JOUR.SIGMA_RUPTURE := RF_PRECIS( PROPRIETES XXX.RUPTURE (densite - gonfV));
POINT_MIS_A_JOUR.ALFA_T := RF_PRECIS( PROPRIETES XXX.DILATATION(tk));
*CONSTANTES_PHYSIQUES_ET_CONVERSIONS.T_CELCIUS(tk);
end loop;
end ACTUALISATIONS_MECANIQUES
end XXX P;

FIG. 2. General set up of properties (extract from package body).
5. OBJECT ORIENTED CONSTRUCTION

Representation of the whole pin is built with elementary objects. An elementary object is a cylinder made with only one material. Two different meshes are used to represent this object: a coarse mesh for thermal and physical calculations and another sharp mesh for mechanical description; both have the same axial (possibly irregular) pitch. Optimisation leads to use a quadratic thermal radial pitch (equal section) assorted with a regular mechanical radial pitch (equal thickness). No problem for binding, along radius and axially, using thermal (adiabatic, given flux, given temperature or exchange) and mechanical (pressure, axial force, given axial or radial displacements) interfaces. Any major component: subchannel, cladding, (shroud), stack(s) and internal gaseous volume is a cluster of same material elementary objects. The whole pin is the binding of the subchannel, the cladding, possibly the shroud and one or several (number not limited) of stack(s); There is no software limits for the total number of elementary objects.

6. CODE DESIGN

The main routine manages the assignation of basic files: input data, standard output, supplementary and graphic outputs, it also carry back run time errors at the upper level (monitor). The LECTURE procedure takes in charge the set of elementary tasks:

- reading material name and properties,
- reading of initial characteristics,
- allocation of memory, initialisation of the values,
- launch of normal or transient operations,
- control of output informations,
- calling of save and recover routines,
- closing of program.

7. CONCLUSIONS

REGAIN is a light, easy to use, code running on classical work station (< 10 minute for a normal case: 10 levels, 2 stacks, 300 e.p.f.d. in 50 time steps). The sharpness of mesh and the total number of different stacks have no software limits owing to the dynamic allocation of memory (limitation is caused by allocated memory space) REGAIN is now being licensed and will be used for the interpretation of experiments and the design of control rods, passive safety devices and core radial subassemblies.
EXPERIENCE WITH CONTROL ROD DRIVE MECHANISM OF FBTR

P.V. RAMALINGAM, M.A.K. IYER, R. VEERASAMY, S.K. GUPTA, L. SOOSAINATHAN, V. RAJAN BABU
Indira Gandhi Centre for Atomic Research, Department of Atomic Energy, Tamil Nadu, India

Abstract

This paper explains the principle of operation of Control Rod Drive Mechanism (CRDM) in Fast Breeder Test Reactor (FBTR) which is a 40 MWt loop type reactor. It discusses the problems faced and solutions evolved during testing of CRDM in air and in sodium and during operation in reactor. Surveillance tests carried out with CRDMs in pile are also discussed.

1. INTRODUCTION

Fast Breeder Test Reactor (FBTR) is a mixed carbide fuelled, sodium cooled, loop type, experimental reactor of 40 MW(t) or about 13 MW(e) capacity. Six B$_4$C control rods in the reactor core are held by grippers of individual drive mechanisms which are in-line with the respective control rods and housed in control plug.

The functions of control rods and Control Rod Drive Mechanisms (CRDM) are

- to start-up and to have controlled shutdown of the reactor
- to control reactivity during normal operation of the reactor
- to scram the reactor on emergency conditions and
- to have burn-up compensation.

FBTR has operated upto 10.2 MW(t). In the present core any two control rods out of six are sufficient enough to bring the reactor from power level to cold shutdown state.

2. DESCRIPTION

Fig. 1 shows the control rod and its sheath assembly. B$_4$C pellets are in a single pin which is vented having sodium as bonding material. Gripper of CRDM holds the head of control rod. During normal operation of reactor, mobile assembly of CRDM and control rod act as a single unit. Electromagnet (EM) in upper part of CRDM holds the mobile assembly. On receiving scram signal, EM is deenergised and mobile assembly along with control rod falls down under gravity.

Oil dashpot provides deceleration for last 50 mm of scram travel. Motor operated screw-nut mechanism moves the EM up/down. Gripper is operated manually during shutdown condition of the reactor.

Primary leak-tightness is achieved by metallic bellows in sodium and secondary leak-tightness is achieved by silicone bellows and 'O' ring seals. Fig. 2 shows the lower part of the mechanism with the leak-tight barriers.
FIG. 1. FBTR control Rod.
FIG. 2. Primary and secondary leak tight barriers of crom.
3. TESTING IN AIR

Eight CRDMs manufactured in India were tested in air prior to their shipment to site. They were again subjected to some important tests before installation in reactor.

The following tests were conducted on CRDMs in air:

- Measurement of insulation resistances of electrical items
- Primary leak tightness test using helium leak detector to check soundness of two primary metallic bellows
- Secondary leak-tightness test by drop in pressure to find soundness of silicone bellows and 'O' ring seals
- Measurement of straightness of CRDM with a dummy control rod
- Measurement of scram (i.e., free fall and braking travel) time of the mechanism at different drop heights and deceleration of the mobile assembly by dashpot
- Tests to find gripper torque, gripper operations and corresponding microswitch actuation

Air test results were satisfactory.

4. TESTING IN SODIUM

One CRDM was tested in sodium at 180°C and at 550°C. Tests were conducted with misalignment of 0 & 10 mm between CRDM & dummy control rod.

Modifications had to be done in sodium test facility in order to achieve the required temperature gradient in CRDM.

At 180°C the following measurements were done:

- differential expansion between translation tube & gripper control tube
- translation operation and torque measurement
- gripper operation & torque measurement with and without control rod.

In the control rod gripped condition, sodium temperature was raised to 550°C. Due to the leakage of sodium from test vessel at weld region, dished head was replaced. Even though the vessel was kept under argon purging, entry of air during dished head replacement could not be avoided. After soaking the vessel with sodium at 400°C and draining for purification, sodium was filled at 180°C and temperature was raised to 550°C. It was found that the control rod could not be released to measure the differential expansion between the mobile part and gripper control tube.

Translation and fast drop experiments were carried out at 550°C. After 60 translations and 50 fast drops, interseal argon pressure suddenly reduced indicating leakage. Subsequent interseal pressure tests indicated main bellows failure.
Sodium was dumped and filled at 180°C and a few experiments were carried out under argon purging. Control rod could not be released. Sodium was drained and radiography was taken in the region of control rod & grippers. Misalignment between gripper body and control rod head was 10.7 mm.

Then mechanism was moved and aligned over the control rod and tests were carried out at 180°C and at 550°C. Control rod could be released. Normal scram & translation operation could be performed. Only gripper torque measurement showed an increase in the values.

At 550°C in the newly aligned condition with argon purging, 100 translation operations, 100 fast drop operations and 10 translation torque measurements were made. All were done with failed bellows.

CRDM was cleaned by vacuum distillation and inspected. The outer tube sheath of the mechanism was found to have bent on the cover gas region about 900 mm above sodium level. Bending was spread on the entire cover gas region. Reasons for this type of bending could not be identified specifically.

The test results were satisfactory even with 10 mm misalignment. Also it can be concluded that CRDM can be operated with failed primary bellows for short period.

5. OPERATING EXPERIENCE IN REACTOR

5.1 Failures leading to safety related incidents

In April '87 during a reactor start-up, control rod-F continued to raise even after removing the 'RAISE' command. Operator action immediately brought the reactor to safe shutdown state by ordering 'MANUAL LOR'. The cause of the incident was analysed and reason was traced to sluggish behaviour of the power contactor for "raising" in the logic circuit. The contactor was replaced. Also to prevent recurrence of the problem, an input to Reactor Protection system was included so as to order LOR when the difference in control rod levels exceeds a preset value.

5.2 Component failures

5.2.1 Increased flow of Interseal Argon

In 1988, an increase in the flow rate of interseal argon was observed. Tests were conducted to detect the mechanism which had developed leak and also to locate the leakage source. The methods followed for leak check included adjusting the interseal argon pressure in a stepped manner so as to reduce the differential pressure with respect to the reactor vessel cover gas and monitoring the flow rate and injection of nitrogen in the interseal argon system and subsequent analysis of reactor vessel cover gas for the same.

Results of the tests indicated that secondary leak-tight barriers of CRDM-B, viz., O-ring seals were leaky and interseal argon was leaking towards reactor vessel cover gas side. Subsequently the O-ring seals were replaced and the interseal argon flow rate came down to background.
5.2.2 Failure of EM of CRDM-C

In July 1991, while checking the CRDMs, mobile part of CRDM-C dropped from 120 mm position and repeated operations and checking confirmed that EM showed abnormally high electrical resistance indicating coil inter-turn shorting.

Subsequent removal and inspection of the electromagnet coil revealed insulation failure which could be attributed to ageing and a kink found in the terminal leads caused during initial assembly. Exposed strands of the coil were reinsulated and assembled. Before installing it for reactor service, the mechanism was tested with dummy load and operations were found to be normal.

5.2.3 Failure of Metallic Bellows of CRDM-B

During April '91, after 1 MW(t) operation of the reactor, a significant increase in interseal argon flow was observed. Leak tests carried out indicated leak in the metallic translation bellows of CRDM-B. The interseal argon supply to the mechanism was kept isolated to avoid loss of argon. Subsequently, primary sodium system temperature was raised to 375°C for carrying out certain test. After lowering the temperature to 250°C mechanism ceased to operate and mobile assembly could not be lifted even manually with a force 1.5 times of what is normally required. Further analysis showed that pressure increase in the interspace of CRDM-B during temperature raising got relieved by argon bubbling through sodium and subsequent temperature decrease caused a vacuum in the interspace. This resulted in the entry of sodium through the leaky bellows into narrow clearance regions of the translation bellows and guide tube sheath and its subsequent solidification. This caused the sticking of the mobile assembly of CRDM-B.

Since the translation mechanism was stuck, if the normal procedure was to be adopted, it would have become essential to take out the control rod also along with the lower part. This would have caused wastage of a healthy control rod, since once taken out of the reactor, it is not advisable to reuse it in reactor. To overcome this difficulty, special tooling and procedure was developed and the lower part was taken out using a special shielded flask, leaving the control rod inside. The maximum radiation field was observed at the stellited guide bush.

Another spare lower part was installed in reactor. Removed lower part was kept in decay pit for sufficient period and after decontamination, components of the lower part were cut, disassembled remotely in a dismantling cell for further disposal.

The above incident indicates that in case of a confirmed leak in translation bellows, interseal argon supply should not be isolated as long as the leaky CRDM is inside the reactor.

5.2.4 Silicone Bellows Failure in CRDM-F

In the middle of 1994, global increase in interseal argon flow was observed. Tests indicated a leak in the silicone bellows of CRDM-F.

Special tools were made for insitu replacement of leaky silicone bellows in lower part of CRDM without removal of nearby CRDMs and mechanism box. The work was successfully carried out insitu in the next shutdown period and later observations showed that interseal argon flow came down to normal value.
5.3 Miscellaneous failures

There were many occurrences of 'stuck closed' position of limit switches used for indication and control logic interlocks. They were minimised by introducing regular preventive maintenance. Events which can be caused due to such 'stuck closed' switches affecting the control circuit logic were identified and modifications were carried out to prevent unusual occurrences.

On a few occasions, failures of shear pin in the gear drive of mechanisms occurred. These were due to over travel beyond the bottom limit and top limit. Although interlocks from limit switches are available to stop the drive motor at the bottom limit, the reason for these failures were mainly disturbances in disconnectable connectors, deficiencies in operating procedure and non-actuation of limit switches. This problem was solved by replacing the connectors with new types which are more reliable, modifying operating procedure and introducing regular preventive maintenance steps.

5.4 Overhauling of upper parts

The preventive maintenance programme includes complete dismantling of the upper parts and inspection of internals once in two years. Dashpot oil is replaced and functional checks of the limit switches are carried out. So far, 3 upper parts have been overhauled. In all the cases uneven wear of teflon pads fitted to the guide tube was observed. The unevenness was corrected by machining.

5.5 Provision of strain gauge

In order to facilitate on-line measurement of friction force of the mechanism, a strain gauge has been provided.

6. SURVEILLANCE TESTS

To ensure proper operation and availability of CRDMs, the following surveillance tests are done at specified intervals:

- Measurement of drop time of the mobile assembly of the mechanism

- Measurement of minimum required current for electromagnet for latching the mobile assembly.

- Measurement of friction force while operating the mobile assembly, with and without control rod.

- Logic checking.

- Integrity checking of CRDM bellows and O-rings.

These checks are done according to the Technical Specifications. Tests are also done whenever logic modification is done or there is a deviation observed from normal parameters.
7. CONCLUSION

The experience obtained during manufacture, testing in air and in sodium, operation in reactor and surveillance tests of FBTR CRDM enhances the confidence level on overall reliability of shutdown system. This has also helped in design and development of shutdown systems for Prototype Fast Breeder Reactor (PFBR).
DEVELOPMENT OF PASSIVE SHUT-DOWN SYSTEMS
FOR THE EUROPEAN FAST REACTOR EFR

M. EDELMANN, G. KUSSMAUL, W. VÄTH
Forschungszentrum Karlsruhe (FZK),
Institut für Neutronenphysic and Reaktortechnik,
Karlsruhe, Germany

Abstract

This paper presents a simple device for passive shut-down of liquid metal cooled fast reactors (LMR) during unprotected transients. It uses enhanced thermal expansion of control rod drive lines (CRDL) to provide for automatic scram and forced control rod insertion when pre-fixed coolant temperatures are exceeded. In this way the reactor is brought into a safe permanently subcritical state and temperatures are kept well below the boiling point of the coolant.

A prototype of such a device called ATHENa (German for: Shut-down by THermal Expansion of Na) has been manufactured and tested. The paper presents the principle, design features and thermal characteristics of ATHENa as well as results of the experiments and reactor dynamics calculations of unprotected loss of flow and transient overpower accidents for the European Fast Reactor (EFR) equipped with enhanced thermal expansion devices.

1. INTRODUCTION

In hypothetical accident and risk analyses for LMRs it is supposed that the conventional safety system may fail completely in emergency situations. So, for instance, it is assumed that during coolant temperature or power transients due to loss of primary coolant flow or uncontrolled withdrawal of control rods either the safety system would fail to activate the rod drop mechanisms or none of the absorbers would move into the core (control rod jamming). Then coolant boiling or fuel melting, respectively, with subsequent severe core damage (Hypothetical Core Disruptive Accident - HCDA) would occur in present design fast reactors with large mixed oxide fuelled cores.

For licensing of future fast reactors it seems necessary to completely exclude any release of radioactivity during a nuclear accident. In principle, this might be achieved in two different ways. The first possibility would consist of definitely excluding core disruptive accidents. The other might be the absolutely safe confinement of radioactive materials within the reactor building. In a more realistic practical solution a combination of these two measures might be feasible, i.e. preventing the most severe core disruptive accidents and safely confining radioactivity from all others within the reactor building.

There are two essentials to fulfil in order to prevent core melting or disruption:

- Reactivity control and sufficient shut-down reactivity must be available under all imaginable circumstances even when the conventional "active" safety system fails to shut down the reactor as postulated in unprotected transients, also called Anticipated Transients Without Scram (ATWS).
- Sufficient heat removal capacity for evacuating decay heat has to be assured even with loss of primary flow or heat sink.

This paper deals with the first item. The ATWSs such as, ULOF (Unprotected Loss Of coolant Flow) and UTOP (Unprotected Transient OverPower) are classic initiators of hypothetical core disruptive accidents. Since the failure of the engineered safety system is postulated in these cases the reactor can be shut down only by inherent passive reactivity feedback mechanisms. The normal reactivity feedback with temperature increase in large LMRs with MOX fuel presently under consideration (e.g. EFR) is too small to outbalance the large sodium density effect and Doppler reactivity during unprotected transients (e.g. ULOF). In present designs reactivity feedback mechanisms cannot prevent the reactor from reaching sodium boiling temperature with the resulting severe consequences.

A detailed study of thermal feedback effects performed at FZK has shown that most of the phenomena producing negative reactivity (e.g. core expansion, and bowing or flowering) are not easy to increase and would provide only minor improvement. In addition, the resulting reactivity effects are hard to predict with sufficient precision and reliability, which is necessary for licensing and public acceptance. Furthermore, all of the inherent thermal reactivity effects have the disadvantage that they are present only as long as temperatures remain significantly increased. As a consequence, the reactor cannot in general be shut down by negative reactivity feedback. It can only be stabilised at elevated temperature levels in this way. However, the reactor has to be shut down finally. Reactor shut-down by reactivity feedback would be possible with a 'one-way feedback' only. This means that the negative reactivity input caused by a coolant temperature rise beyond a certain threshold cannot be reduced by a subsequent temperature decrease.

It was concluded from our study that the easiest and most efficient way to provide sufficient negative reactivity feedback in an LMR would be the enhancement of control rod drive line thermal expansion. In addition, one-way feedback can then be achieved which provides for passive reactor shut-down in case of safety system failure.

Passive safety features in present fast reactor designs have been discussed elsewhere in more detail [e.g. 1, 2]. Generally, most effort is concentrated on new core designs with the aim of increasing negative and decreasing positive reactivity feedback effects with some penalties in cost and performance (e.g. small reactors). Also, in some cases, special devices providing additional negative reactivity feedback are built in. So, for instance, in the Fast Flux Test Facility gas expansion modules (GEM) [3] are used to increase neutron leakage during a loss-of-flow incident. However, the GEMs are not sufficient in large cores and produce negative reactivity on loss of hydraulic pressure only.

In other cases, devices for passive rod drop are introduced in the CRDLs utilising the Curie effect in magnetic materials [4] or enhanced thermal expansion [5] only to disconnect the absorber element from the CDRL. This increases the diversity of the safety system with respect to reactor scram but does not provide forced insertion of absorbers in case of control rod jamming. The ATHENA device however produces a strong and irreversible negative reactivity feedback on excessive coolant temperatures which is sufficient to bring the reactor into a stable safe state during unprotected loss-of-flow or transient overpower accidents even in case of control rod jamming.
The basic elements of ATHENa are shown in Fig. 1. The enhanced thermal expansion CRDL consists of two individual (coaxial arranged) drive lines. The "primary" drive line corresponds to the conventional control rod drive line. However, it consists of two separate shafts linked together by a special release mechanism, which is operated by a "secondary" drive line.

Fig. 1. Basic elements of ATHENa
The complementary "secondary" drive line is an enhanced thermal expansion module, which at normal coolant temperatures has no effect on the position of the absorber element. Its lower end is fixed to the lower shaft of the primary drive line, whereas the upper end can move up and down according to its thermal expansion.

In ATHENA a hydraulic expansion module is used. (Elsewhere bimetallic devices were investigated [6].) It consists of an axially expandable container (metal bellows), which is (partly) filled with sodium. The switching temperature at which the thermal expansion switches from normal to enhanced can be adjusted by the sodium filling. The sodium has a fairly large thermal expansion coefficient, which is more than four times as large as the value of the container material (SS). Consequently, with increasing temperature the sodium level in the container increases and starts extending it when completely filled.

The additional volume of sodium increases only the length of the container, thus moving the movable end of it. This is the upper end of the expansion module as long as the release mechanism is closed. When the upper end has reached the stroke limiter the release mechanism is opened and the expansion is directed downwards. Then the absorber is pushed into the core. The elongation of the container depends on the cross-section of its expandable part. Therefore, the expansion coefficient of the container can easily be multiplied by large factors by simply reducing the cross-section of its bellows part. In this way extremely large thermal expansions of the module can be realised. However, for a given maximum internal pressure reducing the bellows cross-section also decreases the force by which the absorber can be pushed into the core.

Under normal operating conditions, i.e. at sodium temperatures not exceeding maximum design values, the (enhanced expansion) CRDL works as before without any interference from the complementary secondary drive line. However, when the coolant temperature exceeds a pre-fixed level, the secondary drive line opens the release mechanism thus disconnecting the upper and lower primary drive line shafts from each other. Since there is no other link between the upper drive line shaft and the secondary drive line, the absorber element is disconnected for free falling into the core. In this way a fully passive reactor scram is obtained.

If the absorber element is impeded from free falling due to rod jamming, the secondary drive line takes over from the primary by partly replacing its lower shaft. For any further temperature increase the position of the absorber element is determined by the enhanced expansion module, which pushes it downwards with some force to overcome possible mechanical resistance in the absorber channel. The enhanced thermal expansion drive line can be designed in such a way that it forces the absorber far enough into the core to produce sufficient negative reactivity to shut down the reactor.

The design of the ATHENA prototype developed for EFR had to fit to the existing above-core structure (ACS) geometry. Its thermal expansion coefficient, reactivity stroke and mechanical strength had to be chosen such that the passive shut-down capability under worst-case conditions (EOC, control rods withdrawn) would be assured and pushing forces significantly higher than the weight of the absorber element would be obtained without affecting too much the thermal response time of the expansion module. The expansion module is also designed as fail-safe in the sense that any leak from it would cause a rod drop in the same way as excessive coolant temperatures. In [7] ATHENA is described in more detail.
The design optimisation of the ATHENa prototype resulted in the following functional parameters:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Switching temperature</td>
<td>500 °C</td>
</tr>
<tr>
<td>Delatching temperature</td>
<td>505 °C</td>
</tr>
<tr>
<td>Thermal response time</td>
<td>~10 s</td>
</tr>
<tr>
<td>Expansion coefficient</td>
<td>1 mm/°C</td>
</tr>
<tr>
<td>Maximum expansion</td>
<td>200 mm</td>
</tr>
<tr>
<td>Pushing force</td>
<td>10 kN</td>
</tr>
</tbody>
</table>

3. EXPERIMENTS WITH THE ATHENa PROTOTYPE

The ATHENa prototype is an experimental device. It is mounted on a test rig and equipped with thermocouples and displacement transducers for measuring internal sodium temperature profiles and thermal expansion. It also has different types of Na level sensors to be tested as a means of continuously monitoring the leak-tightness of the expansion module which proves its availability. This is complementary to the fail-safe feature of ATHENa which should lead to a rod drop if the expansion module should leak [7].

The test rig also provides for simulating rod jamming. This is achieved by a pneumatic device which tends to move the lower end of ATHENa upwards, i.e. in the direction opposite to its expansion. The resisting force can be adjusted through the air pressure in the pneumatic cylinder. Although the air pressure was kept constant during a single experiment, the counter-force increased with increasing thermal expansion. This is due to the spring effect (20 N/mm) of the metal bellows.

Except for the thermal time constant all essential parameters of ATHENa could be measured by slowly heating it up in an electric oven. Fig. 2 shows the signals of a Na level sensor (temperature difference between a heated and an unheated thermocouple) and a displacement

![Graph](image-url)

Fig. 2. Na sensor and displacement signals versus Na temperature
transducer connected to the lower end of the expansion module. At the beginning, at temperatures below 330 °C, these signals do not change significantly with increasing internal sodium temperature.

At about 330 °C the sodium level reaches the heated thermocouple at the upper end of the sodium container. This is clearly indicated by a sharp temperature decrease. The same signal would be obtained in case of a leak in the expansion module. At about 345 °C the module is completely filled with sodium and enhanced thermal expansion starts. In these experiments the switching temperature from normal to enhanced expansion was reduced by about 120 °C by compressing the expansion module (metal bellows) in order to avoid deformation during expansions for long periods at elevated temperatures.

In Fig. 3 results of rod jamming experiments are shown for two different resisting forces (150/400 kp). It is seen that the thermal expansion is almost linear with temperature. The

![Graph showing thermal expansion characteristics for two different resistance forces.](image)

Fig. 3. ATHENa expansion characteristics for two different resistance forces
slight nonlinearity is due to the non-linear change of the sodium density with temperature. The thermal expansion coefficient is about 15% smaller than the design value of 1 mm/K. It is assumed that this comes from an underestimation of the effective bellows cross-section which had not been measured. Impeding the expansion by a constant resistance only delays the onset of enhanced expansion but does not change the over-all expansion characteristics. The switching temperature for enhanced expansion increases by about 1 K per 100 kp resistance.

The thermal time constant has not yet been measured. The necessary rapid temperature increase of about 10 K/s requires a sodium test loop in which LOF-type coolant flow and temperature transients should be simulated for demonstrating the correct functioning of the whole device under realistic accident conditions. For this reason the test rig is already designed for dynamic tests in sodium.

4. **CALCULATION OF UNPROTECTED TRANSIENTS IN EFR**

For calculating the efficiency of ATHENA devices to prevent severe consequences of unprotected transients in LMRs the dynamic behaviour of the expansion module has to be known. Since experimental results are not yet available thermal hydraulic calculations were performed with the code FLUTAN [8]. As a result, it was found that the thermal expansion as a function of time during LOF-type flow and temperature variations can be described fairly well by a series of two first-order low-pass characteristics. This model together with the calculated time constants was then used in the calculations with the reactor dynamics code DYANA2 [9]. Details of these thermal hydraulics as well as of reactor dynamics calculations for the "first consistent design" of EFR (1.4 m core height) have been published earlier [7]. Thereafter, extensive calculations were performed for the latest core design CD 9/90 of EFR with 1 m core height. A comprehensive report on these calculations is given in [10].

In the following some of the results of these calculations are given. However, it has to be pointed out that in these calculations ATHENA is not used in its most efficient way. In order to avoid any interference with the first and second shut-down systems of the reactor the switching temperature from normal to enhanced expansion was set to 590 °C. This is 45 K higher than the nominal coolant temperature. Therefore, increased negative reactivity feedback becomes effective only when coolant or fuel temperatures have already significantly exceeded normal operating values in ULOF or UTOP accidents, respectively.

It is assumed that all of the 24 CRDLs are equipped with ATHENA devices but none of the absorber rods would drop due to control rod jamming which is extremely unlikely. All calculations were performed with control rods only 10 cm inserted (end of cycle configuration). At these control rod positions the thermal CRDL expansion has the minimum effect on reactivity. Since EFR primary pumps do not have flywheels the flow halving time during a pump coast-down is only 10 s. This leads to a very rapid coolant temperature increase during ULOF accidents.

In Fig. 4 the normalised primary coolant flow and reactor power during a ULOF at full power are shown. Initially, power decreases slowly due to normal reactivity feedback. About 20 s later enhanced thermal CRDL expansion becomes effective. But even then power decreases slower than coolant flow which causes a rapid coolant temperature rise. Resulting sodium temperatures in the hot channel, at core outlet as well as in the CRDL shroud tube outside and
inside ATHENa are given in Fig. 5. In the hot channel sodium boiling temperature is reached but in all other channels the peak sodium temperature remains below 800 °C with a large margin to boiling.

However, even under these worst-case conditions sodium boiling can be avoided including in the hot channel. If only one control rod drops the reactor would be shut down and all temperatures kept well below the boiling point. Alternatively it would be sufficient to reduce the ATHENa switching temperature by only 10 K. This can be seen in Fig. 6 where the hot channel temperature is shown for three different switching temperatures. Similar results are obtained for slightly increased flow halving times, e.g. 12 s [cf. 10]. Finally, hot channel temperatures might also be reduced by flattening the power distribution of the core.

It has to be emphasised that the EFR design is not optimised for the implementation of ATHENa devices. This can be seen from Fig. 5, for instance. It shows that the difference between coolant temperatures at the core outlet and at the ATHENa position in the shroud tube is much bigger than that between shroud tube and ATHENa. This is due to the unfavourable flow conditions in the ACS of EFR. Only a small fraction of the core flow is passing the shroud tubes which contain large quantities of sodium. Thus, the temperature rise in a shroud tube is strongly attenuated and delayed by coolant mixing. ATHENa would be much more efficient if shroud tube temperatures were close to core outlet temperature.

In a UTOP accident an uncontrolled withdrawal of a control rod is supposed. Other than in a ULOF the fuel temperature is primarily affected and coolant temperature rise is only
Fig. 5. Na temperatures during ULOF in EFR with enhanced CRDL expansion

Fig. 6. EFR hot channel Na temperatures for different ATHENA switching temperatures
Thus, the major concern is fuel melting which may finally lead to pin failure and core disassembly with severe consequences for the environment. Due to the moderate coolant temperature rise reactivity feedback from CRDL thermal expansion is less efficient than in a ULOF. Nevertheless, for reactivity ramp rates up to 1 \( \phi/s \) ATHENa would limit the power transient by disconnecting the absorbers from the CRDLs.

However, fuel melting in the hot channel would be prevented only if at least one absorber dropped into the core. Otherwise the ATHENa switching temperature has to be reduced to about 570 °C or the reactivity ramp rates have to be further limited. This is illustrated by Fig. 7 which shows the peak fuel temperature in the hot channel for different switching temperatures (\( 'Tcr' \)) and reactivity ramp rates. In all cases the reactor power can be stabilised at significantly high levels only (130 to 140 % nominal power) if none of the control rods drops into core. A single rod drop would stabilise the reactor at a power level and coolant temperature below nominal values. The reactor will be shut down when at least two absorber rods drop.

As in the case of the ULOF flattening of the power distribution and improved heat transfer between core outlet and shroud tubes would be very helpful in preventing fuel melting and would enable the ATHENa devices to bring the reactor into a permanent subcritical state even if all control rods were jamming. This would provide for more flexibility in reactivity ramp rates and ATHENa switching temperature.

---

**Fig. 7.** Maximum fuel temperatures during UTOPs in EFR with ATHENa
5. CONCLUSION

ATHENa, a prototype passive shut-down system for Na cooled fast reactors has been developed and tested. It utilises enhanced thermal CRDL expansion at excessive coolant temperatures to disconnect the control rods from the drive lines and to push them into the core. It was demonstrated that such a device can be robust and reliable in providing either passive rod drop or forced insertion of control rods at a thermal expansion rate of about 1 mm/K against a resistance of 10 kN. Reactor dynamics calculations have shown that in this way sufficient negative reactivity is produced to prevent sodium boiling and fuel melting during ULOF and UTOP accidents in EFR under the most pessimistic conditions.

This is true even though in the EFR design the implementation of ATHENa was not foreseen. If in future LMR designs this is properly taken into account HCDAs during unprotected transients in fast reactors might be definitely excluded even for large cores with mixed oxide fuel.

REFERENCES

DESIGN PHILOSOPHY OF PFBR SHUTDOWN SYSTEMS

Indira Gandhi Centre for Atomic Research,
Department of Atomic Energy,
Tamil Nadu, India

Abstract

This paper presents the overall design philosophy of shutdown system of 500 MWe Prototype Fast Breeder Reactor (PFBR). It discusses design criteria, parameters calling for safety action, different safety actions and the concepts conceived for shutdown systems. In tune with the philosophy of defence-in-depth, additional passive shutdown features, viz., Self Actuating Device (SADE) and Curie Point Magnetic (CPM) switch and protective feature like absorber rod Stroke Limiting Device (SLD) are contemplated. It also discusses about suitability of Gas Expansion Module (GEM) as one of the safety devices in PFBR.

1. INTRODUCTION

Prototype Fast Breeder Reactor (PFBR) is a 500 MWe, mixed oxide fuelled, sodium cooled pool type reactor. It consists of two primary sodium pumps and four IHX and two secondary loops each loop having one sodium pump and four integrated steam generator modules. Two important aspects on which safety of the reactor depends are the reliability of shutdown systems and the reliability of decay heat removal systems. All engineered safeguards are well tuned towards this objective. Overall reliability of shutdown systems, of course, depends upon the well conceived design, manufacture, quality control, prototype testing, on-line monitoring and surveillance.

2. DESIGN CRITERIA FOR SHUTDOWN SYSTEM

Broad guidelines for design of shutdown system are as follows:

• Atleast two reliable, independent, automatic, fast acting shutdown systems shall be provided operating on diverse principles. Atleast one of the systems shall meet all functional requirements even in case of postulated core deformation. The reliability of each system shall be such that its non-availability is less than $10^{-3}$ per reactor year and the overall non-availability of the two systems shall be less than $10^{-6}$ per reactor year.

• The design shall provide sufficient redundancy so that failure of a single most effective absorber rod of a shutdown system shall not result in impairment of that system to an extent that it will not meet the minimum specified requirements of negative reactivity.

• One of the shutdown systems could be used for reactivity control. However, while doing so, its functional capability to shutdown the reactor shall not be jeopardised.
The reactivity worth, speed of action and delay in actuation of each shutdown system shall be such that during all operational states and postulated accident conditions of the reactor, including the most reactive state of the core,

- the reactor is rendered sufficiently subcritical and maintained subcritical under cold condition, taking into account uncertainties in the neutronic calculations/measurements,
- the specified fuel design limits are not exceeded,
- the reactor coolant system design limits are not exceeded.

The total reactivity worth of the shutdown systems shall be such that in the shutdown state, with all absorber rods in the core, the reactor shall be subcritical with \( k_{\text{eff}} \) not more than 0.95 such that the reactor remains subcritical under postulated fuel handling errors (e.g. replacement of the most reactive absorber rod by most reactive fuel sub-assembly, removal of absorber rods).

The availability of safety support systems necessary for actuation of a shutdown system shall be commensurate with the availability requirements of the shutdown system.

All equipment shall be designed such that its probable failure modes will not result in an unsafe condition.

The design shall be such that all maintenance and availability testing which may be required during reactor operation can be carried out without a reduction in the effectiveness of each system below the minimum allowable requirements.

The design shall be such that each shutdown system can be actuated manually from the main and emergency control rooms.

The design shall be such that it is not readily possible for an operator to prevent a safe automatic action from taking place.

The control logic of the absorber rods and their drive mechanisms shall be designed to prevent unintended movement in the directions which add reactivity.

Maximum reactivity worth of an absorber rod, together with its maximum possible withdrawal speed, shall be limited such that the fuel, coolant and cladding design limits are not exceeded in the event of uncontrolled withdrawal of the rod.

3. CONCEPTS CONCEIVED FOR SHUTDOWN SYSTEM

Reactor safety is assured by two independent, fast acting, diverse shutdown systems, each comprising of sensors, logic circuits, drive mechanisms and absorber rods. Type-A system consists of a bank of 9 absorber rods while the Type-B system consists of a bank of three absorber rods. Type-A is for reactivity control as well as for reactor shutdown, whereas Type-B is only for reactor shutdown. Each rod is operated by an individual mechanism in-line with the rod. Their disposition is shown in Fig. 1. In both the systems, shutdown of reactor is achieved by dropping the absorber rods by gravity. But the scram release Electromagnet (EM) of Type-A system is housed in upper part of mechanism above top of control plug and in argon atmosphere, whereas that of Type-B system is at lower end of mechanism immersed in primary sodium hot pool near the top of sub-assemblies. Details of these shutdown mechanisms and absorber rods are dealt with separately in another paper [1].
<table>
<thead>
<tr>
<th>SYMBOL</th>
<th>TYPE OF SUBASSEMBLY</th>
<th>No.</th>
<th>MASS PER SUBASSY. IN Kg</th>
</tr>
</thead>
<tbody>
<tr>
<td>☻</td>
<td>FUEL (INNER)</td>
<td>85</td>
<td>245</td>
</tr>
<tr>
<td>☼</td>
<td>FUEL (OUTER)</td>
<td>96</td>
<td>245</td>
</tr>
<tr>
<td>☼</td>
<td>PRIMARY CONTROL ROD</td>
<td>9</td>
<td>200</td>
</tr>
<tr>
<td>☾</td>
<td>SECONDARY CONTROL ROD</td>
<td>3</td>
<td>200</td>
</tr>
<tr>
<td>☾</td>
<td>BLANKET</td>
<td>186</td>
<td>320</td>
</tr>
<tr>
<td>☾</td>
<td>STEEL REFLECTOR (INNER)</td>
<td>72</td>
<td>355</td>
</tr>
<tr>
<td>☾</td>
<td>$B_4C$ SHIELDING (INNER)</td>
<td>69</td>
<td>185</td>
</tr>
<tr>
<td>☾</td>
<td>STORAGE LOCATION</td>
<td>75</td>
<td>245</td>
</tr>
<tr>
<td>☾</td>
<td>RESERVE STORAGE LOCATION</td>
<td>24</td>
<td>355</td>
</tr>
<tr>
<td>☾</td>
<td>ENRICHED BORON SHIELDING</td>
<td>56</td>
<td>185</td>
</tr>
<tr>
<td>☾</td>
<td>STEEL SHIELDING (OUTER)</td>
<td>180</td>
<td>330</td>
</tr>
<tr>
<td>☾</td>
<td>$B_4C$ SHIELDING (OUTER)</td>
<td>903</td>
<td>265</td>
</tr>
</tbody>
</table>

**FIG. 1. PFBR core configuration.**
Triplicated sensors are used to measure parameters important to reactor safety and are connected by a '2 out of 3' coincidence logic to the reactor protection system. A hot standby channel aids in maintaining the '2 out of 3' coincidence logic. Selection of parameters is based on the well developed concepts of diversity. Physical separation of redundant channels of the safety system is proposed. The routing of redundant signals will be physically independent.

Two independent Reactor Protection Logic Processing Systems are provided. One system will be based on hard wired solid state logic circuits working on pulse coding mode, whereas the other system will have either relay logic or two microprocessors one aching standby to the other for logic functions. Protection systems will have facility for on-line testing, wherever needed, without causing safety action.

4. PARAMETERS CALLING FOR SAFETY ACTIONS & TYPES OF SAFETY ACTIONS

Depending upon the nature of fault, the reactor protection system is designed to initiate two types of safety actions on the reactor; a gradual shutdown and a fast shutdown. The gradual shutdown is effected by Lowering of Rods (LOR) and the fast shutdown (Scram) is effected by simultaneously dropping all the absorber rods from their existing positions into the reactor core. The important parameters calling for safety actions are given in Table I.

<table>
<thead>
<tr>
<th>No</th>
<th>PARAMETER</th>
<th>SAFETY ACTION</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>LOR</td>
</tr>
<tr>
<td>1</td>
<td>High Neutron Flux in start-up range</td>
<td>-</td>
</tr>
<tr>
<td>2</td>
<td>Short period in start-up range</td>
<td>-</td>
</tr>
<tr>
<td>3</td>
<td>High Neutronic Power (Log P)</td>
<td>-</td>
</tr>
<tr>
<td>4</td>
<td>High Neutronic Power (Lin P)</td>
<td>-</td>
</tr>
<tr>
<td>5</td>
<td>Short period in power range</td>
<td>-</td>
</tr>
<tr>
<td>6</td>
<td>High reactivity in core</td>
<td>-</td>
</tr>
<tr>
<td>7</td>
<td>Deviation of temperature rise for each sub-assembly from the calculated temperature rise for that power</td>
<td>yes</td>
</tr>
<tr>
<td>8</td>
<td>Deviation from mean core outlet temp.</td>
<td>yes</td>
</tr>
<tr>
<td>9</td>
<td>Deviation from mean gradient temperature</td>
<td>yes</td>
</tr>
<tr>
<td>10</td>
<td>DND (bulk) due to Fuel clad failure</td>
<td>-</td>
</tr>
<tr>
<td>11</td>
<td>Earthquake horizontal acceleration</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>- X component</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>- Y component</td>
<td>-</td>
</tr>
<tr>
<td>12</td>
<td>Power/Primary flow rate</td>
<td>yes</td>
</tr>
<tr>
<td>13</td>
<td>Primary pump trip</td>
<td>yes</td>
</tr>
<tr>
<td>14</td>
<td>Secondary pump trip</td>
<td>yes</td>
</tr>
<tr>
<td>15</td>
<td>Feed water pump trip</td>
<td>yes</td>
</tr>
<tr>
<td>16</td>
<td>Loss of electric power</td>
<td>yes</td>
</tr>
<tr>
<td>17</td>
<td>Turbine trip</td>
<td>yes</td>
</tr>
<tr>
<td>18</td>
<td>Water/steam leak into sodium in SG</td>
<td>yes</td>
</tr>
<tr>
<td>19</td>
<td>OR ineffective</td>
<td>-</td>
</tr>
</tbody>
</table>
5. DEFENCE-IN-DEPTH

It is observed that among various incidents leading to Loss of Flow event and Transient Over Power event, loss of off-site power and inadvertent withdrawal of an absorber rod are respectively the most probable events and they lead to severe consequences, if they are coupled with simultaneous failure of both shutdown systems, i.e., Unprotected Loss of Flow (ULOF) and Unprotected Transient Over Power (UTOP) events. ULOF and UTOP are now considered as Beyond Design Basis Events (BDBE).

With the site as Kalpakkam, evaluation carried out indicates probability of 120 grid supply failure of 1 h duration and 5 grid supply failure of 15 h duration in the life time of the plant. In the philosophy of defence-in-depth, if the reactor is protected from this incident by passive means, then the overall unavailability of reactor protection system can be reduced to less than $10^{-7}$ per reactor year.

5.1 Loss of Off-site power

Reactor must be shutdown, in case of a confirmed loss of off-site power (i.e., duration >2.5 s). To achieve safe shutdown by passive means during this event, the following three concepts have been studied:

- Gas Expansion Module
- Self Actuating Device
- Curie point magnetic switch.

5.1.1 Gas Expansion Module (GEM)

GEM is essentially a passive shutdown device to insert negative reactivity during a primary system LOF. The device is basically a hollow removable sub-assembly sealed at the top and open at the bottom. An argon gas trapped inside the sub-assembly expands when core inlet pressure decreases due to flow reduction and expels sodium from the sub-assembly. The effect of putting GEM in PFBR at the interface of core and radial blanket giving 1.5 $\delta$ of reactivity on voiding of GEM have been studied theoretically. The studies have clearly shown that for a flow coastdown incident of 12 s flow halving time constant, the reactor is passively safe and the peak coolant temperature reached is 903 K (630°C).

Even though passive safety can be achieved with GEM Sub-assemblies against an ULOF event with flow halving time of a few seconds (and hence reducing the need for high moment of inertia for primary pump fly wheels), they have the following disadvantages:

- Inservice inspection of GEM is difficult to check argon pressure and sodium level. If argon leaks out from GEM then sodium level falls and consequent negative reactivity during LOF event is not assured. The knowledge on dissolution and leaks of argon in sodium is not extensive.
- Noise in reactivity may be introduced due to fluctuation of sodium level in GEM sub-assembly.
- Argon under high pressure may enter the central part of the core and add positive reactivity.
• At low power operation, speeding up of pumps would add positive reactivity ($> 1\,\text{s}$).

• GEM reactivity worth depend on sodium level and its pressure which varies with flow. GEMs are not effective below 25% flow.

• Operating experience is minimum.

Hence GEM is not favoured.

5.1.2 Self Actuating Device (SADE)

SADE, proposed by France [2], is a motor-generator system with a flywheel on the same shaft. Motor is operated by off-site power supply. The generator supplies power to scram release EM of shutdown mechanism. In case of loss of off-site power, the speed of the flywheel reduces and also the voltage of the electrical generator. After a short-time, this voltage is no more sufficient and EM releases the absorber rod. Direct aching SADE system is proposed to give supply to EMs of Type-B shutdown system absorber rods.

5.1.3 Curie Point Magnetic (CPM) switch

CPM switch deenergises scram release EM when it is demagnetised on reaching a temperature of 873 K (600°C) under power excursion, thereby scramming the reactor [3]. Initially it was considered to have CPM switch as part of Type-B shutdown mechanism at its lower end. Refer Fig. 2.

Studies were carried out to find response time of the switch, from the time at which sodium temperature exited from fuel sub-assembly starts increasing (due to LOF) to the time at which the temperature reaches curie point (873 K) of the magnet. Fig. 3 shows the behaviour. Since the response time is of the order of 2.6 s to 3.6 s (depending upon the gap between the magnetic switch and outer sheath of the mechanism), it is decided to locate the CPM switch just above a fuel sub-assembly. Studies are under progress to position it at proper location in control plug.

5.2 Inadvertent withdrawal of absorber rod

Type-B rods, of course, need not be considered, since they are poised outside the core during normal operation of reactor. But Type-A rods are to be considered, since they are partially inside core. They can not move w.r.t. the mechanisms, as they are firmly held with the mechanisms. But mobile assembly along with absorber rod could raise up inadvertently due to failure of logic circuit or due to operator error. Hence it has to be prevented from raising up by a physical stop which is known as Stroke Limiting Device (SLD). SLD which can be adjusted based on the power level of the reactor, is provided in each Type-A mechanism. Proper care is taken in the design so that SLD no way hinders the safety (LOR & Scram) functions of the shutdown mechanisms.
FIG. 2. Magnetic switch and electromagnet assembly.
6. CONCLUSION

A systematic approach has been followed in the design of shutdown system of PFBR. The design criteria has been well laid out and as a defence-in-depth philosophy, passive features like SADE, CPM switch and SLD have been incorporated to enhance the overall reliability of the reactor protection system.

REFERENCES

DESIGN OF SHUTDOWN SYSTEM FOR PFBR

V. RAJAN BABU, R. VIJAYASHREE, R. VEERSAMY,
L. SOOSAINATHAN, GOVINDARAJAN, S.M. LEE,
S.C. CHETAL

Indira Gandhi Centre for Atomic Research,
Department of Atomic Energy,
Tamil Nadu, India

Abstract

This paper presents the principles adopted in the design of two independent, fast acting, diverse shutdown mechanisms and their absorber rods for Prototype Fast Breeder Reactor (PFBR). It describes features of the shutdown mechanisms and their absorber rods, passive features, present status in R & D and program on passive and protective feature as a measure of defence-in-depth, testing and verification.

1. INTRODUCTION

Prototype Fast Breeder Reactor (PFBR) has two independent, diverse, fast acting shutdown systems working on fail safe mode. Functions of the shutdown system Type-A are

- to start-up and have controlled shutdown
- to control reactor power
- to scram the reactor on emergency conditions
- to have burn-up compensation

Function of Type-B system is limited to scram the reactor on emergency conditions.

Nine Type-A rods in two radial banks facilitate adjustment of radial form factor by differential insertion of rods. Out of nine rods, three are in inner PCD and six are in outer PCD. Type-B system has three rods in inner PCD. In order to avoid common mode/cause failure and to achieve diversity, the design of both the systems is in such a way that parts related to safety action of absorber rod under scram action are entirely different.

2. DESIGN OF ABSORBER ROD

For both the systems B$_4$C is chosen as absorber material because of its relatively high neutron cross section, commercial availability, ease of fabrication, low cost and good operating experience in Fast Reactors. B$_{10}$ enrichment is kept in the range of 50 - 70 % so as to have a margin for extra reactivity requirements in future without changing number of rods.

2.1 Reactivity Worth

Reactivity worth of absorber rods must be adequate to satisfy the following criteria:

- Adequacy of shutdown margin (SDM) to handle postulated incidents like LOF, TOP due to uncontrolled full withdrawal of the most reactive absorber rod or due to any other reason,
fuel melting and slumping in a few sub-assemblies, sodium boiling and voiding in a few
sub-assemblies etc.

- Adequacy of reactivity worth in each individual shutdown system to bring the reactor to cold
  shutdown state assuming that other system has failed and the most reactive rod of the
  working system is also stuck.
- Adequacy of SDM in the fuel handling state to cater to postulated errors such as replacement
  of most reactive rod by a high enrichment fuel SA during fuel handling, accidental withdrawal
  of two absorber rods etc.

Worth of individual rod is relatively small, so that flux distortion and shadow/anti-shadow
effects are not large.

2.2 Raising/Lowering Speed

With the raising speed of Type-A rod being 2 mm/s, the maximum rate of reactivity addition
is limited to 2 pcm/s. This enables smooth start-up and also leaves adequate margin to protect
the reactor in the event of an uncontrolled withdrawal of a rod. In the case of Type-B rods, a
speed of 10 mm/s or less is acceptable. Speed of travel of 4 mm/s is selected. Considering safety
and simplicity in design, raising and lowering speeds of individual system are made same.

2.3 Mechanical Design

Stationary hexagonal wrapper tube of absorber rod is like any other fuel sub-assembly
externally, whereas mobile outer sheath of absorber pins is cylindrical. In each Type-A rod, there
are 19 pins each having B\textsubscript{4}C pellets in clad tubes. Fig. 1 shows the Type-A absorber rod and its
wrapper tube. Pins are vented having sodium as bonding material. Wrapper tube, outer sheath and
clad are all made of 20 % CW D9. Type-B rod has similar design except having comparatively
more clearance between absorber rod outer sheath and wrapper tube. It also has a swivel joint
between head and stem of the rod.

2.4 Thermal-Hydraulic Design

Sodium flow in absorber sub-assemblies is such that

- Maximum cladding mid-wall temperature shall not exceed 953 K (680°C) under normal
  operating conditions when the rods are in withdrawn position
- The sodium hot spot temperature which is in between B\textsubscript{4}C pellet and cladding shall not
  exceed the boiling point during the fall of one absorber rod into the core when the reactor
  is operating at full power under new equilibrium configuration.

3. DESIGN OF SHUTDOWN MECHANISM

Type-A and Type-B mechanisms are in-line with the corresponding rods in core and are
housed in control plug (CP). There is a mechanical coupling between mobile assembly of Type-A
mechanism and rod, whereas there is an electromagnetic coupling between Type-B mechanism
and rod. The mobile assembly of Type-A is held by an electromagnet kept above the top of CP.
Schematic arrangement of Type-A and Type-B mechanisms are shown in Fig. 2 & 3.
FIG. 1. PFBR primary control rod.

CONTROL ROD POSITIONS
- P1: CR resting on sheath
- P2: CR held in CRDM at bottom position
- P3: CR held in CRDM at top position
FIG. 2. Type-A mechanism.

FIG. 3. Type-B mechanism.
Two Plant Protection Logic provide scram signals to the corresponding shutdown systems separately. On receiving scram signal, electromagnets (EM) of both the systems are deenergised. Hence in the case of Type-A system, mobile assembly of the mechanism along with absorber rod is released to fall under gravity; but in the case of Type-B mechanism, only absorber rod falls under gravity. More than 2/3 of the core height is covered by the free fall of the rods and then they are decelerated by oil dashpots (provided at the top of CP) in the case of Type-A system and sodium dashpots (provided in wrapper tube of absorber rod) in the case of Type-B system.

Screw-nut mechanism is used in both the systems for raising and lowering of mobile assembly. Since it is not participating in scram action, common mode/cause failure is not of specific concern here.

In Type-A mechanism, positions of EM and mobile assembly are indicated by synchro resolver and potentiometer respectively, whereas in Type-B mechanism only potentiometer indicates position of mobile assembly. Microswitches / proximity switches are used for control of interlocks.

Since EM of Type-A system is at low temperature and in argon atmosphere, low carbon steel is used as magnetic material. But EM of Type-B system is immersed in hot pool sodium and directly holding the head of the absorber rod. Hence pure iron is considered as material for core of the magnet, whereas 2.25 Cr 1 Mo steel is considered for head of the absorber rod to avoid self welding of similar metals.

Both the mechanisms are hermetically sealed to achieve leak-tightness even during hypothetical Core Destructive Accident (CDA) and during Safe Shutdown Earthquake (SSE).

Temperature at top of control plug (CP) is maintained at 110°C so that sodium vapour deposition does not hinder the free movement of mobile assembly. 'O' ring seals are used wherever there is no relative movement or where there is slow relative movement between the parts whose movement is of no safety concern. Lip type seals like 'V' ring seals or oil seals are proposed to be used between mobile assembly and the stationary tube sheath at the level of CP top to act as a barrier restricting sodium vapour coming into upper part. Positive pressure in hermetically sealed upper part w.r.t. reactor cover gas facilitates to have flow of argon from upper part into the reactor vessel in case of leakage of seals i.e., from non-active to active region. Argon gas supply to the upper part is bubbled through NaK to avoid oxygen entering into the system.

Major concern on the overall reliable performance of the mechanisms with the rods is the proper functioning of scram release EM and dashpot and the friction between stationary and mobile parts.

4. DESIGN OF CONTROL SYSTEM FOR SHUTDOWN MECHANISMS

Two independent and diverse control systems are proposed. For Type-A mechanism, Programmable Logic Controllers (PLC) will be used considering the importance and the large number of control components. But for Type-B mechanism, conventional EM relays will be used, since the number of control components are relatively less. Triplicated fail-safe control logic will be used to improve the reliability of PLC.
5. PRESENT STATUS IN R & D

It is planned to do testing on sub-assemblies of both the mechanisms which are of safety concern. In this context 1:1 scale model of EM and dashpot of both the systems have been planned. It is also planned to do experiments on seals and sodium vapour deposition in annular gap between mobile and stationary parts.

5.1 Scram release Electromagnet

EM of Type-A mechanism, made of Low carbon steel, was tested in air to find its characteristics on load carrying capacity, response time and temperature rise. This EM has to handle a load of about 5000 N in argon atmosphere. Results of load carrying capacity and temperature rise of EM are well satisfying the design requirements; but the response time is more than the specified limit of 100 ms, due to large eddy current induced in EM. Design modifications are being done to reduce eddy current.

First to ascertain the design of EM in Type-B system, 1:1 scale model of EM with low carbon steel and absorber rod head made of 2.5 Cr 1 Mo was tested in air with different gaps between the contact faces. Increased air gap necessitates more excitation current. If the actual air gap is less compared to the assumed gap, then that will cause large magnetic holding force leading to high magnetic response time. The results are satisfactory. It is planned to test EM made of pure iron and head of absorber rod made of 2.5 Cr 1 Mo steel, first in air at 813 K (550°C) and then in sodium at different operating conditions.

5.2 Dashpot

1:1 scale model of oil dashpot of Type-A system and its test set-up are being fabricated to find characteristics of dashpot, viz., velocity and deceleration of mobile assembly and piston and oil pressure with respect to time. Weight of mobile assembly falling on piston of dashpot is about 4500 N. Dashpot travel is 250 mm. Design modifications will be done based on the test results.

It is planned to test 1:1 scale model of Type-B absorber rod and the wrapper tube first in water at room temperature and then in sodium at reactor operating temperature. Weight of absorber rod considered is 400 N.

5.3 Seals

Lip seals like oil seal and 'V' ring seal are being tested in a set-up with 1:1 scale model. Tests are proposed to be done in static condition and in dynamic reciprocating condition of the mobile assembly. Leak rate and friction while raising up and lowering down are studied. Test results are encouraging. In the present studies, various configuration of the seals are being tested to find an optimum one. Then endurance test on different aged seals will be done for scram action of mobile assembly of the mechanism.
5.4 Sodium vapour deposition

Sodium vapour deposition in the annulus between the mobile and stationary parts with different gaps and convection barriers is being studied in a test set-up. Based on the test results, annular gaps in the mechanisms will be properly modified. Here temperature of the mechanism is to be maintained in such a way that the top of CP is at 110°C.

5.5 Prototype Testing

It is planned to test prototype mechanisms with dummy absorber rods of both the types of systems first in air and then in sodium to ascertain the overall functioning of the systems at all operating conditions. Fabrication of prototype systems is in progress.

6. PROGRAM ON PASSIVE AND PROTECTIVE FEATURES

6.1 Curie Point Magnetic (CPM) switch

In order to improve the reliability of shutdown systems during Loss of Flow (LOF) event, Curie Point Magnetic (CPM) switch which cuts off the power supply to EM when the sodium near the switch reaches 873 K (600°C) is considered. As mentioned in [1], studies are under progress to position it at proper location in CP.

6.2 Stroke Limiting Device (SLD)

Inadvertent withdrawal of absorber rod during operation of reactor is to be considered for Type-A rod, since it is only partially inside core. In control logic, it is provided to restrict the upward movement of mobile assembly only a few mm continuously. But if there is a fault in logic circuit, then it would lead to TOP event. To avoid this, a physical stop which can be adjusted to any specific power level by administrative control, is provided in Type-A mechanism. A mechanical device operated on screw-nut principle is coupled in parallel with the motor drive-screw assembly. Based on the travel position of mobile assembly and hence the absorber rod, the position of stopper will be changed. When the device touches the stopper, further rotation of drive-screw is not possible. At that position, actuation of microswitch gives indication and stops the motor. Damage to drive-screw and motor will be avoided by safety pins.

7. CONCLUSION

PFBR shutdown system incorporates adequate diverse and redundant features and additional passive features to ensure high reliability. Overall reliability of the shutdown system is based on

- proper design foreseeing all conditions and factors which would have effect on proper functioning,
- well defined procedure for fabrication & quality control,
- testing of sub-assemblies & 1:1 scale prototype system simulating all operating conditions in reactor to satisfy all the design requirements and
- design modifications based on the test results
- regular surveillance tests.

R & D works related to Shutdown systems of PFBR are under progress.
REFERENCES

DEVELOPMENT OF PASSIVE SAFETY DEVICES FOR SODIUM-COOLED FAST REACTORS

IPPE, Obninsk, Russian Federation

Abstract

In the recent years, development of passive safety devices for sodium-cooled fast reactors advanced significantly. The paper presents some research results of this subject, done at the IPPE (Institute of Physics and Power Engineering) during 1990-95.

INTRODUCTION

The development of enhanced-safety NPP units is the most important problem of nuclear power development.

Calculation estimates have shown that in a sodium-cooled fast reactor considerable core damages at the most severe accidents accompanied by a failure of the main safety system can be avoided if a small effect upon reactivity by passive means is provided. The reactor with passive safety features actually acquires in this case inherent safety properties as all its safety effects and features are based on natural processes proceeding in the core. The passive safety devices are additional means (to the main safety system) and are designed to control beyond design basis accidents at a failure of the main safety system with the aim to preclude sodium boiling and severe damages in the core.

Various principles of passive safety engineering realization are known. In Russia at present the most emphasis is being given to two principles of passive safety means actuation /1 - 3/:
- by decreased primary coolant flow rate;
- by increased temperature of coolant at the core outlet
In both cases the insertion of the rod into the core takes place by gravity.

1. PASSIVE SAFETY SUBASSEMBLIES (PSS) WITH A HYDRAULICALLY SUSPENDED ABSORBER ROD.

In 1988-89, two passive safety subassemblies based on the BR-10 standard subassemblies were developed, fabricated and put for tasting into the BR-10 reactor (Fig.1.1 and Table1.1). The subassembly mockups were preliminarily tested at the hydraulic (water) rig. The calculation method developed has allowed to use the results obtained in water tests for sodium as well.

In December 1994 the program of life time testing of the two subassemblies in the reactor including on-power actuation was completed. The tests have confirmed the design parameters of the subassemblies are recommended for use in the BR-10 reactor as standard ones.
FIG. 1.1. PSS subassembly with hydraulically suspended rod for the BB-10 reactor.
TABLE 1.1. TECHNICAL PARAMETERS OF PASSIVE SAFETY SUBASSEMBLIES (PSS)

<table>
<thead>
<tr>
<th>N</th>
<th>Parameter</th>
<th>Units</th>
<th>N 1</th>
<th>N 2</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Absorber rod cladding diameter</td>
<td>mm</td>
<td>22.5*0.3</td>
<td>21.5*0.3</td>
</tr>
<tr>
<td>2</td>
<td>Absorber rod weight</td>
<td>g</td>
<td>~242.0</td>
<td>~225.0</td>
</tr>
<tr>
<td>3</td>
<td>Absorber rod efficiency</td>
<td>%Δk/k</td>
<td>0.146</td>
<td>0.22</td>
</tr>
<tr>
<td>4</td>
<td>Flow rate through the reactor at a rod rise</td>
<td>m³/h (% QN)</td>
<td>96 (48)</td>
<td>81 (40.5)</td>
</tr>
<tr>
<td>5</td>
<td>Flow rate through the reactor at a rod drop</td>
<td>m³/h (% QN)</td>
<td>70 (35)</td>
<td>63 (31.5)</td>
</tr>
<tr>
<td>6</td>
<td>Rated flow rate through the PSS at a raised rod</td>
<td>m³/h</td>
<td>0.93</td>
<td>0.97</td>
</tr>
<tr>
<td>7</td>
<td>Time of rod drop into the core</td>
<td>s</td>
<td>1.14</td>
<td>0.67</td>
</tr>
</tbody>
</table>

The effect of absorber rods insertion on system reactivity of coolant flow rate through the core was determined at power levels of 1-2000 kW. In this case coolant flow rate was decreased to a value no less than 25% of the rated one.

Main data on life time tests of PSS 1,2 in the BR-10 reactor are presented in Table 1.2.

TABLE 1.2. MAIN DATA OF PSS N1, 2 TESTS IN THE BR-10 REACTOR

<table>
<thead>
<tr>
<th>N of PSS</th>
<th>Core cell</th>
<th>Time duration</th>
<th>On-power operation, eff. days</th>
<th>Accumulated fluence (E&gt;0, Mev) n/cm²</th>
<th>Number of actuations (including those on-power)</th>
</tr>
</thead>
<tbody>
<tr>
<td>PSS N 1</td>
<td>110</td>
<td>3.01.89-5.01.89</td>
<td>0</td>
<td>2.6*10²¹</td>
<td>38 (10)</td>
</tr>
<tr>
<td>110</td>
<td>29.03.89-1.08.89</td>
<td>39.55</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>95</td>
<td>14.09.92-17.1192</td>
<td>21.96</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>PSS N 2</td>
<td>95</td>
<td>15.05.91-11.08.92</td>
<td>151.07</td>
<td>1.70*10²²</td>
<td>125 (10)</td>
</tr>
<tr>
<td>95</td>
<td>23.11.93-25.11.94</td>
<td>127.42</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>95</td>
<td>25.11.94-15.06.95</td>
<td>55.5</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Total time of operation in the reactor was for PSS N 1-218 days, for PSS N 2-1020 days.

The number of rod drops during the time of the tests was 38 and 116 for RSS N 1 and N 2 subassemblies, respectively. No cases of rod jamming in the subassemblies were noted. Flow rate values through the reactor at rod rise and drop during the tests did not change.

In 1989 work on the development of passive safety devices with the hydraulically suspended rod for the BN-600 type reactor was started. In 1988-89 a test absorber subassembly for the BN-600 reactor was designed on the basis of its standard shutdown absorber subassembly, and its full-scale mock-up was
manufactured for testing in the hydraulic (water) rig Fig. 1.2. Several design versions of the subassembly were tested. At the end of 1994 rig tests of the subassemblies were completed; depending on their results, a subassembly design version for hydraulic testing in the BN-600 reactor will be recommended.

FIG. 1.2. PSS subassembly with hydraulically suspended rod for the BB-600 reactor.
For the subassembly developed the following algorithm of its operation has been adopted. At the shut-down reactor the rod in the subassembly is in the ultimate lower position - on the rigid stop of the guide sleeve. Prior to a reactor power rise the rod is engaged with the drive stem by means of its gripper and then it is lifted into the upper working position. The primary coolant flow rate is increased from a minimum (0.25 Gnom) to a start-up (0.75 Gnom) one and the gripper is opened. In so doing the rod remains in the gripper as the rated flow rate of coolant through the sleeve is chosen in such a way that at a flow rate equal to ~ 0.6 Gnom the hydraulic force acting upon the rod would be no less than its weight. At cutting off of one of three heat-removal reactor loops its power and flow rate are automatically decreased to a level of 0.67 of the rated one. And the rod remains in the upper working position.

At a signal for shut-down of the reactor the rod is automatically put into the lower working position by the drive stem at the open gripper. In this case the absorber remains in the gripper, as during the time of its movement (~1s) from the upper working position to the lower one the primary coolant flow rate does not practically change. With the reduction of flow rate when changing over the primary pumps to reduced rotations (0.25 Gnom) a decrease of the hydrodynamical force takes place; at its decrease below the rod weight the rod drops from the lower working position (80 mm above the stop) into the brake and is hold in it. At further flow rate reduction the rod from the brake softly descends on the stop. In case of a drive failure the rod starts falling into the core by gravity at a flow rate decrease down to ≤0.6 Gnom. At its fall it stops at first in the brake (40 mm above the stop) and then, at further flow rate reduction, it softly seats against the stop.

At reactor refuelling (G = 0.25 Gnom., the rod being on the stop), in case of an erroneous switching on of the pumps to full capacity the rod does not float up into the core (it only rises in the brake to ~ 40 mm) because the buoyant force acting on it in this case is more than 2 times less than its weight. Non-floating up of the rod is provided due to a hydraulic load relief of the working part of the rod.

In Table 1.3 main hydrodynamical characteristics of the BN-600 standard safety system subassembly and of two PS subassemblies tested in the water rig are presented; coolant flow rates (l/s) are given scaled to sodium at an operating temperature.

<table>
<thead>
<tr>
<th>S/A Type</th>
<th>( \tau_{1,s} )</th>
<th>( \tau_{2,s} )</th>
<th>( Q_{susp} )</th>
<th>( Q_{nom} )</th>
<th>( Q_{float} )</th>
<th>( Q_b )</th>
<th>( Q_{nom,rod} )</th>
<th>( Q_{min,rod} )</th>
<th>( \eta )</th>
</tr>
</thead>
<tbody>
<tr>
<td>safety S/A</td>
<td></td>
<td></td>
<td>2.0</td>
<td>1.0</td>
<td>0.25</td>
<td>2.2</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>PSS N 1</td>
<td>10.1</td>
<td>6.1</td>
<td>3.6</td>
<td>6.0</td>
<td>11.5</td>
<td>2.1</td>
<td>1.0</td>
<td>0.36</td>
<td>2.5</td>
</tr>
<tr>
<td>PSS N 2</td>
<td>8.7</td>
<td>4.7</td>
<td>2.7</td>
<td>4.5</td>
<td>11.5</td>
<td>2.1</td>
<td>1.0</td>
<td>0.25</td>
<td>3.6</td>
</tr>
</tbody>
</table>
where \( \tau_1 \) - total time of rod insertion into the core after the beginning of an accident;
\( \tau_2 \) - time of rod insertion into the core from the moment of beginning of its movement (primary flow rate reduction to 0.6 Gnom.);
\( Q_{\text{susp}} \) - coolant flow rate through subassembly under which rod suspension in the upper working position by coolant flow takes place;
\( Q_{\text{nom}} \) - rated coolant flow rate through S/A;
\( Q_{\text{float}} \) - coolant flow rate through S/A at which the rod would float up from the lower position into the upper one;
\( Q_B \) - coolant flow rate through S/A at which the rod would float up in the brake;
\( Q_{\text{nom}}^\text{rod} \) - coolant flow rate through the rod in the upper working rod position at a rated flow rate through S/A;
\( Q_{\text{min}}^\text{rod} \) - coolant flow rate through the rod in the lower position at a flow rate through S/A of 0.25 G/nom.;
\( \eta \) - margin for rod non-floating up.

In Figs. 1.3, 1.4 the effect of PSSs actuation with various rod worths (0.6 and 1.2% \( \Delta k/k \)) is shown at a beyond design basis accident in the BN-600 reactor with total loss of electric power supply and a failure of all active reactivity control systems for various times of the beginning of PSS actuation (\( \tau_1 = 4 \) and 14 s) from the beginning of an accident and of absorber insertion into the core (\( \tau_2 = 2, 4, 7 \) s). It follows from the figures that for the rod worth, values under consideration, \( \tau_1 \) and \( \tau_2 \)

\[
\begin{array}{ccc}
& \Delta k/k, \% & \tau_1, s & \tau_2, s \\
1 & 0.6 & 4 & 2 \\
2 & 0.6 & 4 & 7 \\
3 & 0.6 & 14 & 2 \\
4 & 1.2 & 4 & 4 \\
5 & 1.2 & 4 & 7 \\
\end{array}
\]

FIG. 1.3. Sodium temperature at the core outlet as a function of time since the onset of an accident.
The core outlet sodium temperature level at an early stage of accident development is mainly determined by the beginning of absorber insertion (Fig. 1.3), and at a later stage - by the value of the reactivity worth inserted (Fig. 1.4). From Fig. 1.3 it follows that at a PSS efficiency value of $\Delta k/k$ (efficiency of one standard safety rod), a time of starting its actuation $\tau_1 = 4s$ (during this time from the beginning of the accident the coolant flow rate decreases to a critical value of 0.6 Gnom.) and a time of absorber insertion into the core $\tau_2 = 6s$ (see Table 1.3) the core outlet sodium temperature will not exceed $720^\circ C$, i.e., considerable margin ($200^\circ C$) is ensured to the sodium boiling point.

2. PS SUBASSEMBLY WITH THERMAL-PRINCIPLE-BASED ACTUATING DEVICES (AD).

The test PS subassembly of this type is being developed on the base of the BN-600 reactor standard subassembly. In the subassembly a shorter fuel element bundle is used, and coolant flow rate is decreased, respectively, so that the coolant temperature at the fuel elements outlet in the PSS and fuel subassembly would be close to each other. In the head piece of the PSS an actuating device holding the absorber rod is located. At an increase of the AD temperature it comes into action - releases the absorber rod which drops into the core by gravity. Such type AD as applied to its operating conditions in BN-600 PSS is under development since 1990.
From Fig. 1.3 it follows that if the temperature for AD to come into action is taken as equal to 650-670°C then to obtain a margin of 100 and 150°C to the sodium boiling point it is necessary that the time of absorber insertion beginning (τ₁) into the core (AD lag time) should not exceed 10 and 5 seconds, respectively.

2.1. Magnetic Material-Based Actuating Device

Fig.2.1 shows a mock-up design of the magnetic actuating device (MAD) developed for experimental testing in the sodium rig as applied to its operation conditions in the BN-600 PSS.

<table>
<thead>
<tr>
<th>MAD type</th>
<th>H₁</th>
<th>H₂</th>
<th>Φ₁</th>
<th>Φ₂</th>
<th>Φ₃</th>
<th>armature</th>
<th>screen</th>
<th>magnet</th>
</tr>
</thead>
<tbody>
<tr>
<td>mock-up N1</td>
<td>135</td>
<td>90</td>
<td>40*2</td>
<td>20*1</td>
<td>25</td>
<td>cylinder</td>
<td>type80Ni20Fe</td>
<td>typeAlNiCo-6</td>
</tr>
<tr>
<td>mock-up N2</td>
<td>85</td>
<td>45</td>
<td>41*4</td>
<td>29*1</td>
<td>25</td>
<td>cone</td>
<td>type80Ni20Fe</td>
<td>typeAlNiCo-6</td>
</tr>
</tbody>
</table>

Dimensions are presented in mm.

**FIG. 2.1. Magnetic actuating device (MAD).**
The magnetic system of the MAD consists of a permanent magnet made of magnet-solid alloy magnetized in the axial sense, a screen made of ferrous-nickel alloy with a Curie point of 620°C and an armature of Armco iron connected to the absorber rod. At a screen temperature increase above the Curie point it loses its magnetic properties that results in MAD load-lifting capacity reduction. At a decrease of the load-lifting capacity below the absorber rod weight, the MAD armature disengages and the rod drops into the core by gravity. By the present time testing of the MAD mock-up in the sodium rig has been conducted (~1000 hours). Its load-lifting capacity in a sodium flow within a temperature range of 300 - 680°C has been determined, and its dynamical characteristics at a fast (~12°C per second) rise of passing-over sodium temperature were studied as well.

The MAD mock-up tests have revealed that it has considerable thermal inertia: its time constant is 6.4 s. Coming into action of the MAD takes place during ~6 seconds after the beginning of the accident, the coolant temperature at the core outlet not exceeding 715°C.

At present preparatory work for the advanced MAD design tests is under way.

2.2. At Actuation Device Functioning on the Basis of Various Physical Effects.

This device (Fig.2.2) includes a number of thermosensitive working elements functioning on the basis of various physical effects (phase transition, memory effect

FIG. 2.2. Operating device functioning on the basis of various physical effects.
of shape, etc.). An absorber drop takes place at actuation of any of thermosensitive elements. The main working element of the device is bellows passed over by coolant, plugged on both ends and filled in the compressed condition with a material having a phase transition with a volume increase at an actuation temperature (aluminium with \( t_{\text{melt}} \sim 660^\circ\text{C} \) and \( \Delta V \sim 6.6\% \)). Calculation and experimental studies have shown that bellows elements have characteristics (a time lag of 2 to 8s, a stroke of 2 to 8 mm, a force of 450 to \( 10^4\text{N} \)) which can ensure the required for device performance. The device design includes also two (upper and lower) elements, made of titanium alloy with the memory effect of shape at a temperature of 630-670\(^\circ\text{C} \), manufactured in form of a disc spring pack. Calculation and experimental validation studies of this element were carried out. It has been shown that these elements have characteristics ensuring reliable actuation of the device as well (the actuation time of 1s, a stroke of 6-8 mm, the force developed of 700H). Calculation and experimental work on obtaining the required characteristics of the device is under way.

**CONCLUSION.**

At IPPE work on the development of PSS subassemblies based on two principles of their actuation is under way:
- by decreased primary coolant flow rate;
- by increased temperature of coolant at the core outlet.

A PSS subassembly with a hydraulically suspended rod for the BR-10 reactor has been designed. Two such type subassemblies have successfully undergone life time tests in the BR-10 reactor including on-power actuation. The tests have confirmed the design characteristics of the subassemblies; such type subassemblies are recommended for routine operation.

A full-scale PSS subassembly with the hydraulically suspended rod for the BN-600 reactor has been designed and tested in the hydraulic (water) rig. The test results obtained allow to recommend such a subassembly for in-pile tests.

A magnetic device actuating at an excess of a designed temperature has been developed and tested in the sodium rig. The device design has been improved on the basis of the test results, and its rig tests are carried on.

A device actuated by increased temperature on the base of various physical effects (memory effect of shape, phase transition, compressed spring energy, etc.) has been developed. Rod insertion into the core takes place at actuation of any of thermosensitive elements. Experimental and calculation studies of the device, thermosensitive elements characteristics are carried out.

**REFERENCES**

MAIN FEATURES OF THE BN-800 PASSIVE SHUTDOWN RODS

Yu.K. ALEXANDROV, V.A. ROGOV, A.S. SHABALIN
OKBM, Nizhny Novgorod,
Russian Federation

Abstract

The major design characteristics of hydraulically suspended shutdown rods, as well as the principles of their operation and test results for the mock-up scram rods, in combination with the simulations of the BN-800 reactor internal structures providing for the rods operation, are given.

1. INTRODUCTION

The analysis of beyond design basis accidents for BN-800 reactor showed that to prevent overheating of the core in case of loss of power or primary MCP shutdown with main shutdown systems failure to the common reason, application of an auxiliary system is required, which is based on the alternative, preferably passive principles of actuation.

Taking into account characteristic features of core structures and BN-800 above-core structures (separate transfer of guide sleeves and rods, the so-called closed FA head, deficit of space above the core for installation of a rod in a sleeve, deteriorated thermal bond between rod channel and sodium at the core outlet) and as a result of possible variants development an additional shutdown system was accepted comprising three HSRs, spontaneously inserted into the core if primary sodium flow rate is decreased.

2. MAIN PRINCIPLES OF HSR FUNCTIONING

A possibility to implement an additional shutdown system based on hydraulically suspended HSR for BN-800 reactor was stipulated by sufficiently high initial level of primary sodium flowrate during usual ("cold") reactor start-up and disconnection of a loop ($-0.7 \ G_{nom}$), what allows to accept the required, in respect to core cooling conditions, sodium flow ($-0.5 \ G_{nom}$) at which HSRs are suspended in the flow and began to drop into the core at MCP-I coastdown. The cocking of HSRs is done by their lifting into upper position before bringing the reactor to power as it is done for rods of main emergency protection system with subsequent opening of grips of actuators at primary sodium flow increase up to $>0.7 \ G_{nom}$.

The suspension of HSRs in their upper position is assured by the choice of sodium flow through the gap between the guide sleeve channel above the core and effective link of rod and the required (to conditions of nuclear safety) holding of HSRs in the core after actuation or during refuelling is assured by considerable decrease of rods hydraulic resistance in their lower position due to bypass holes made in the lower part of extension rod communicating the increased gap between the effective link and sleeve with inner cavity of the extension rod (Fig.1).
<table>
<thead>
<tr>
<th>Refuelling</th>
<th>Decoupling</th>
<th>Power</th>
<th>HSR operation actuation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Power Coupling</td>
<td>Power Coupling</td>
<td>Power Coupling</td>
<td>Power Coupling</td>
</tr>
</tbody>
</table>

Fig. 1 HSR functioning scheme
To mitigate the impact of HSR rod onto the sleeve at actuation a hydromechanical shock absorber is provided between rod head having conical surface for seating onto the upper face of the sleeve and upper extension rod.

3. DESIGN CHARACTERISTICS OF HSR

As a part of BN-800 reactor core project with zero sodium void reactivity effect the HSR development and calculational verification were carried out. When developing HSR the emergency protection rod design is taken as a basis but for assuring the minimal flow of sodium through the sleeve the mass of HSR rod is reduced to minimally possible and is ~17 kg. The diameter of sleeve channel and rod wrapper tubes is taken as for EP rods namely 78 mm and 73 mm, respectively.

The calculations performed have shown that the velocity of sodium in required the gap for HSR rod suspension is ~4 m/s what corresponds to nominal flow of sodium through HSR gap, i.e. ~5 kg/s, thereby ~1 kg of this flow goes through effective link of the rod. Bypassing the effective link and gap between the extension rod and channel through bypass holes assures double margin to HSR floating up in respect to pressure drop at nominal flow of sodium through the core.

The results of calculation analysis of HSR movement dynamics at MCP-I coastdown are shown in Fig.2. Here the results of calculations of sodium temperature variations at core outlet at MCP-I deenergization with failure of main shutdown systems are also given. As the calculations show the time of HSR drop is ~6 s (~13s from begin of MCP-I coastdown) what limits the increase of temperature at core outlet to 740°C.

4. EXPERIMENTAL INVESTIGATIONS OF MODEL HSR HYDROMECHANICAL CHARACTERISTICS

For development of BN-800 reactor HSR rod characteristics the investigations of model HSR characteristics were carried out together with reactor model units assuring HSR functioning (sleeve, guide tube, actuator grip) and using water as working medium having close to sodium hydrodynamical characteristics. Check of model HSR movement was effected with the help of sensors on the basis of magnet controlled contacts.

The tests performed allowed to determine the hydraulic characteristics of model HSR and establish the laws of its movement at lowering the water flow modelling the coastdown of BN-800 MCP-I pump and determine the conditions of model HSR floating up from its lower position. The basic parameters of model rod are close to expected ones.

In Fig.3 the typical experimental plots of model HSR movement at lowering the water through the sleeve are given.

In total, 60 drops of model HSR was done including drops at maximum adverse radial displacements of the sleeve and guide tubes exceeding possible displacements in BN-800 reactor.

The inspection of model HSR and the sleeve after tests did no reveal damages and marked deformation of contact surfaces. Slight gloss on sealing cones of the rod and sleeve head was detected as well as circular attrition on shroud tube of rod effective link, mainly, in the zone of contact with lower edge of guide sleeve.
Fig. 2 HSR movement and reactor parameters variation at deenergization with failure of all main shutdown systems

H-relative movement of HSR, N-relative power, G-relative flowrate,
t-sodium temperature at core outlet
Fig. 3 Movement of model HSR at flow decrease
H—movement of model rod, m; ΔP—pressure drop though the channel, kgf/cm²
CONCLUSION

An additional shutdown system is developed for BN-800 reactor on the basis of hydraulically suspended absorber rods arbitrarily dropping into the core at lowering the flow of sodium through the core below $0.5 \ G_{\text{nom}}$. The HSR system assures safe shutdown of the reactor at MCP-I deenergization or cut off independent of CPS drives or control equipment condition.

Rig tests of model HSP performed using water as working medium in conditions maximally approaching actual ones demonstrated high reliability of hydraulically suspended rod functioning what allows to begin testing the experimental HRS rod in BN-600 reactor.
THE DESIGN OF A BACKUP REACTOR SHUTDOWN SYSTEM OF DFBR

K. OKADA
Mitsubishi Heavy Industries, LTD., Yokohama

K. TARUTANI
Toshiba Corporation, Tokyo

Y. SHIBATA
Hitachi ltd., Tokyo

M. UETA, T. INAGAKI
Japan Atomic Power Company, Tokyo

Japan

Abstract

This paper outlines the status of the development of passive safety measures to achieve subcriticality in DFBR. Two independent reactor shutdown systems (RSSs), a primary RSS and a backup RSS, are provided each of which can rapidly bring the reactor to a safe shutdown condition. The backup RSS is provided with a Self—Actuated Shutdown System (SASS), which is passively released due to the Curie point effect by abnormal rising of coolant temperature. Two independent RSSs reduce the occurrence probability of core damage due to the ATWS (ULOF, UTOP & ULOHS) events to less than $10^{-7}$/ry. Installation of the SASS reduces the occurrence probability of ULOF & ULOHS further by two orders of magnitude. Overall frequency of core damage due to ATWS events is reduced to the order of less than $10^{-9}$/ry owing to the SASS installation. A screening study was conducted to search for promising core safety enhancement technologies based on a consideration of reactivity requirements for passive success scenarios in ATWS events. As a result of this study, an installation of Gas Expansion Modules (GEMs) in DFBR was recently proposed to enhance the core safety further. The reactivity worth of 66 GEMs arranged outside of the core was approximately $-1.5\%$, and it is possible to prevent the sodium boiling on ULOF by GEMs alone. In addition to SASS and GEM, the feasibility study of the Enhanced Thermal Elongation Mechanisms (ETEM) has been conducted. As the results of some experiments and design studies, it was confirmed that ETEM could be one of the promising mechanisms to enhance core safety, and gives less impact on the performance of a future large scale reactor core.

1. Status of DFBR

In the course of developing Japanese FBRs, the experimental FBR "Joyo" is being operated satisfactorily and the prototype FBR "Monju" has achieved criticality as of April, 1994. This leads to the prospect of the construction of the demonstration FBR...
(DFBR) and the commercialization of FBRs at around the year 2030.

The design study and related R&D for the DFBR is being conducted by the Japan Atomic Power Co. (JAPC), which is the principal organization responsible for the construction as entrusted by nine electric power companies and Electric Power Development Co. Ltd. Research and development on the DFBR is being conducted by the Power Reactor and Nuclear Fuel Development Corporation (PNC), the Japan Atomic Energy Research Institute (JAERI), the Central Research Institute of the Electric Power Industry (CRIEPI) and JAPC. To coordinate R&D, these organizations established the Steering Committee for Coordinating R&D on Japanese FBRs.

In designing the DFBR, it is important for the commercialization of FBRs to reduce the plant construction cost, which is rather greater than that of light water reactors (LWRs). From this viewpoint, two types of FBRs were compared. These are the pool type reactor, and the top-entry loop-type reactor, which has the most compact arrangement of the primary system components. The top-entry loop-type was selected for the DFBR because of the following considerations:

a) Major primary components such as the intermediate heat exchangers (IHXs) and the pumps are outside of the reactor vessel. This facilitates maintenance and repair.

b) The system has flexibility to introduce such innovative technologies necessary to achieve the commercialization of the FBRs.

c) Experience gained at the prototype "Monju" must be fully utilized.

Considering that the top-entry system is quite a new concept, the conceptual design study, the evaluation study of commercialization prospects and water hydraulic tests using models of thermal-hydraulic properties peculiar to the top-entry system were conducted.

Through these studies, technical feasibility and the possibility of the FBR commercialization was confirmed, and the plant concept was also established.

Also based on the results of these studies, the presidential committee of the Federation of Electric Power Companies determined the basic specifications for the No.1 DFBR in January 1994, and decided to promote its development aiming for start of construction in the early 2000s.

The design targets of the DFBR are as follows:

a) Safety level should be at least as good as that of LWRs

b) Economic feasibility at about 1.5 times the cost of the LWR on a 1,000MWe basis

c) High burnup and long operating cycle to reduce cost of electricity generated

d) Reactor outlet temperature of 550 °C to achieve high thermal efficiency

e) Easier maintenance and repair, taking advantage of the distributed equipment layout

Table-1 lists the major specifications of the plant. Fig-1 shows the appearance of the top-entry loop-type DFBR.

2. Current design of reactor shutdown systems of the DFBR

Two independent reactor shutdown systems are provided, a primary reactor shutdown system and a backup reactor shutdown system, to increase system reliability and to eliminate common mode failures. (Fig-2) Each system is completely
### TABLE 1. PLANT REFERENCE SPECIFICATIONS

<table>
<thead>
<tr>
<th>Item</th>
<th>Specifications</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Basic specifications</strong></td>
<td>Top entry loop type reactor</td>
</tr>
<tr>
<td>1 Reactor type</td>
<td>1,600MWt (about 660MWe)</td>
</tr>
<tr>
<td>2 Thermal output</td>
<td>1,600MWt (about 660MWe)</td>
</tr>
<tr>
<td><strong>Other major specifications</strong></td>
<td>3 loops (about 530MWt/loop)</td>
</tr>
<tr>
<td>3 Number of loops</td>
<td>550 °C</td>
</tr>
<tr>
<td>4 Reactor outlet temperature</td>
<td>495 °C /169 atm</td>
</tr>
<tr>
<td>5 Main steam temperature and pressure</td>
<td>Homogeneous core</td>
</tr>
<tr>
<td>6 Core and fuel</td>
<td>Pu-U mixed oxide fuel</td>
</tr>
<tr>
<td>(1) Core</td>
<td>pellet with center hole</td>
</tr>
<tr>
<td>(2) Fuel</td>
<td>About 90,000MWd/t (Initial phase)</td>
</tr>
<tr>
<td>(3) Burnup</td>
<td>About 150,000MWd/t (High burnup phase)</td>
</tr>
<tr>
<td>(4) Breeding ratio</td>
<td>With radial blanket : About 1.2</td>
</tr>
<tr>
<td></td>
<td>Without radial blanket : About 1.05</td>
</tr>
<tr>
<td>7 Steam generator</td>
<td>Once-through type</td>
</tr>
<tr>
<td>8 Reactor shutdown system</td>
<td>2 independent systems</td>
</tr>
<tr>
<td>9 Decay heat removal system</td>
<td>4 loops of DRACS</td>
</tr>
<tr>
<td>10 Reactor containment</td>
<td>Rectangular reinforced concrete</td>
</tr>
<tr>
<td></td>
<td>containment with liner</td>
</tr>
<tr>
<td>11 Fuel handling system</td>
<td>Rotating plug, manipulator type</td>
</tr>
<tr>
<td>12 Reactor building</td>
<td>Horizontal seismic isolation building</td>
</tr>
</tbody>
</table>

![Diagram](attachment://fig-1.png)

**Fig-1 Plant configuration**
Fig-2 Arrangement of the reactor shutdown systems

independent from sensor to control rods. Parameters in the safety protection system are selected to fully protect the core from all abnormal occurrences. For the detection of abnormal events occurring relatively frequently which require prompt shutdown, the primary reactor shutdown system and the backup reactor shutdown system have different parameters. Each shutdown systems works by 2 out of 4 logic with diversity.

The primary reactor shutdown system employs rigid control rods of the gas–accelerated insertion type which were developed for the prototype reactor "Monju". The backup reactor shutdown system employs articulated control rods of the gravity–dropping type. The backup reactor shutdown system is also equipped with a Self–Actuated Shutdown System (SASS), which can release control rods after sensing a temperature rise of the coolant. This ensures the control rod insertion even if the safety protection systems fail.

3. Passive safety measures

The reliability of two independent RSSs is high enough to address Anticipated Transients Without Scrams (ATWS) as the beyond design basis events. However, if one assumes ATWS events, the reactor core would be damaged in several tens of seconds which is too short a period to allow the operator to take any action in order to prevent the core damage. Hence the passive shutdown capability, especially against ATWS events, is desired to enlarge the grace period of a large scale FBR toward its commercialization.
There are several ideas to establish passive shutdown capability. But it should be noticed that such a passive countermeasure should be selected by taking account of the safety characteristics of LMFBR. The most crucial issue for the design is that the passive countermeasure should have less uncertainty for its evaluation and developments, without affecting the core and plant design.

As a result the SASS has been selected as the most promising countermeasure to introduce into the DFBR.

The PSA study of the DFBR concluded that two independent RSSs reduce the occurrence probability of the three types of ATWS (ULOF, UTOP & ULOHS) events to less than \(10^{-7}\)/ry. Installation of the SASS reduces the occurrence probabilities of ULOF & ULOHS further by two orders of magnitude, reducing the probabilities of the three ATWS events in the same order of UTOP \((10^{-10}/\text{ry})\) as shown in Fig–3.

A screening study was conducted to search for promising core safety enhancement technologies for the DFBR based on a consideration of reactivity requirements for passive success scenarios in ATWS events. As a result of this study, the gas expansion module (GEM) has been selected for a detail design study for the application in the DFBR. The GEM is considered to show promise as a measure to enhance the core safety by slowing down the event progress during the ULOF and prevent from the coolant boiling.

In addition to SASS and GEM, the enhanced thermal elongation mechanism (ETEM) for the control rod drive line is now under development to improve the inherent safety characteristics in a large scale FBR core. The target is to extend the grace period up to a half hour against ATWS events.

---

**Result of PSA Level-1 Study**

<table>
<thead>
<tr>
<th>Type of ATWS Event</th>
<th>Without SASS</th>
<th>With SASS</th>
<th>Overall</th>
</tr>
</thead>
<tbody>
<tr>
<td>ULOF</td>
<td>2.7E-10</td>
<td>2.5E-8</td>
<td>2.5E-8</td>
</tr>
<tr>
<td>ULOHS</td>
<td>5.7E-10</td>
<td>5.7E-8</td>
<td>5.7E-8</td>
</tr>
<tr>
<td>UTOP</td>
<td>1.0E-10</td>
<td>9.4E-10</td>
<td>9.4E-10</td>
</tr>
<tr>
<td>Overall</td>
<td></td>
<td>8.2E-8</td>
<td>8.2E-8</td>
</tr>
</tbody>
</table>

*a. 8.2E-8 means \(8.2 \times 10^{-8}\)*

---

**Fig–3** Reduction of occurrence probability of ATWS events by installation of the SASS in DFBR
4. Features of design measures
4.1 Self-Actuating Shutdown System (SASS)

The SASS of the DFBR is installed in the backup shutdown system, which consists of six B4C rods of the two-segment articulated type (Fig-2). The SASS concept is shown in Fig-4. The SASS consists of a conventional electro-magnet and a Curie-point temperature magnetic alloy. The latter has a structure with fins to enhance its quick response to a change in its ambient temperature. The SASS has a flow guide which leads the sodium from the exits of the six fuel subassemblies neighboring the SASS rod to the temperature sensitive alloy (TSA). The flow guide has optimized flow holes to shorten the transport delay time of the sodium flow from the exits to the TSA.

![Diagram of SASS concept](image)

(1) Design of the holding force vs. temperature of SASS magnet

Fig-5 shows the design of the holding force versus the temperature of SASS magnet. The SASS rods will be released when the magnetic holding force decreases less than the effective weight of the control rod in the sodium flow.

The SASS must securely hold the rod under normal operating conditions to exclude any spurious reactor scram. Hence the holding force is designed to be twice the weight of the rod at the lowermost window temperature of 640 °C, which is the expected maximum temperature of the TSA during the normal operation under the 3 σ uncertainty condition.

At the uppermost window temperature of 680 °C, the holding force is designed to be a half the weight of the rod to assure the release of the SASS rods when the sodium temperature of core outlet increases abnormally (without scram). The uppermost window temperature is determined from the requirement for prevention of sodium
boiling inception (with an appropriate margin) in the nominal hot subassembly of the core in a typical ULOF event of DFBR (i.e., loss of off–site power without scram during rated power operation).

(2) The selection criteria used for the temperature sensitive alloy are as follows:
   a) the Curie point temperature should be between 650 °C and 680 °C,
   b) the saturated flux density should be more than 0.5T at the preventive temperature of control rod spurious drop (640 °C),
   c) the ratio of saturated magnetic flux densities at 680 °C and 640 °C should be more than two, and
   d) magnetic hysteresis and resistance should be negligibly small.

Three candidate Ni–Co–Fe alloys were selected for the alloy which will satisfy the magnetic holding force characteristic of Fig–5. The chemical compositions and magnetic characteristics of the selected alloys are shown in Fig–6. They were confirmed to have a negligible magnetic hysteresis and resistance. Laboratory tests of thermal aging, metallography and strength of the materials have been conducted. The thermal aging test in high temperature air has been continuing for more than 3740 hours until 24000 hours.

4.2 Gas expansion module (GEM)

Application of GEMs in fast reactors was investigated in the USA over many years and employed in the design of the PRISM plant, which was reviewed by the USNRC in the process of Generic Approval of Nuclear Plant Design. As a part of the PRISM design activity, the effectiveness of GEM against main–pump stop accidents in a small fast reactor was demonstrated by USDOE in the passive safety test in FFTF in 1986.
GEM is a facility where the gas is contained in the upper portion of a duct and the lower end is open to the high pressure plenum. At the time of an accident, the neutron leakage is increased by the expansion of the gas with decreases of the pump discharge pressure and the negative reactivity is inserted as shown in Fig-7. As a result of the design study, the GEM are supposed to surround the entire periphery of the core as shown in Fig-8. The reactivity worth of 66 GEMs arranged outside of the core was $-1.5\$, and it was possible to prevent the sodium boiling on ULOF by GEMs alone.
Pin Diameter 8.5mm
Pins per Assembly 217
Assembly Pitch 158.1mm

○ Core Fuel (Inner) 199
○ Core Fuel (Outer) 96
Core Total 295
○ Primary Control Rod 24
○ Backup Control Rod 6
Control Rod Total 30
○ Gas Expansion Module (GEM) 66
○ Radial Blanket 150
○ SUS Shield 2 Layers 174
○ B4C Shield 3 Layers 306

Fig-8 Core layout for DFBR

GEM reactivity disturbance due to the fluctuation of the inlet plenum pressure during normal operation and partial load conditions imposes another restriction on the GEM design. It was concluded that to limit the reactivity disturbance within an allowable range, the sodium level in GEM must be raised high in the upper axial blanket region, where the slope of the GEM reactivity worth is small enough to accept the sodium level fluctuation.

To satisfy these conditions imposed from the plant operations, it has been proposed to keep the flow rate of the primary system at 100% rated flow whenever the reactor is at criticality.

4.3 Enhanced Thermal Elongation Mechanism (ETEM)

The ETEM is installed at the upper part of the control rod. The ETEM will insert the neutron absorber into the core following the temperature rise of the coolant, and then it gives a negative feedback effect, which improves the self-control capability. Two types of ETEM concept are shown in Fig-9. Control rods in operating conditions are hung in the core and ETEMs are exposed by the flowing sodium from the neighboring Subassemblies, so that ETEMs will easily respond to the temperature changes in the outlet of Subassemblies.

1. Linear Hydraulic Expansion Device (LHED)

The working principle of the LHED is to convert the volume expansion of liquid to the linear elongation. Liquid Sodium or Sodium–Potassium Alloy (NaK) with large thermal expansion coefficients are likely to be used as the working fluid.

2. Amplification of Differential Expansion Device (ADED)

The working principle of the ADED is that the multiple links work to convert and amplify the thermal expansion difference, between the temperature-sensitive cylinder and the low-expansion rod to axial elongation. Austenitic stainless steel for the
temperature-sensitive cylinder and a molybdenum or a tungsten for the low expansion rod are employed as the thermal expansion members. A feature of the mechanism is that a large elongation rate will be expected in the early stage during ATWS events, since the temperature-sensitive cylinder will be heated at first so as to enlarge the thermal expansion difference.

ETEM design requirements have been determined as follows, on the basis of the ATWS analyses:

**Thermal elongation characteristics:**
- 120mm/200 °C (for BCRs)
- 30mm/200 °C (for PCRs)

  BCR : Backup control rod  PCR : Primary control rod

**Response time constant:**
- Less than 10 sec.

5. Evaluation of performance of measures
   (Performance of SASS/GEM system against ULOF)

5.1 Analysis conditions
The performance of the SASS/GEM system of the DFBR on a ULOF events was evaluated under the following conditions:

- Nominal hot channel conditions
  - Peak linear heat rating = 410 W/cm
  - Initial maximum coolant temperature = 627 °C
b) SASS conditions
- Temperature time constant of the SASS magnet = 1.0s
- Coolant transport delay time = 1.5s
- Initial coolant temperature around the magnet = 571 °C
- SASS rod delatching temperature = 680 °C

c) Plant conditions
- Flow halving time of the primary pumps = 6.5s
- Pony motor flow = 15% of rated flow

5.2 In case of SASS alone
On the ULOF events, even if the reactor shut down systems fail, the SASS rod will be released when the SASS temperature sensitive alloy reaches 680 °C about 8 seconds after the pump trip. As shown in Fig–10, in case of the SASS alone (without GEMs), maximum coolant temperature is restricted to be 880 °C, which satisfies the non-boiling condition and prevents the core damage.

5.3 In case of SASS & GEM
Fig–11 shows the relationship between the sodium level in GEM and GEM reactivity worth. On the ULOF events, the rapid negative reactivity is inserted by GEMs with decreases of pump discharges after pump trip, and it makes the reactor power decrease. The SASS rod will be released when the SASS temperature sensitive alloy reaches 680 °C about 10 seconds after the pump trip. The maximum coolant temperature is 790 °C as shown in Fig–10, which is 90 °C less than in case of SASS alone.
Even in case of GEMs alone (for reference), maximum coolant temperature reaches 880 °C as shown in Fig—10, and it is possible to prevent the sodium boiling on ULOF.

6. Future development plans for SASS and GEM

(1) Future development plan for SASS

Future plans for experimental studies consist of materials tests and tests for the components and whole system of SASS.

(Materials tests)
- a) Thermal aging tests of the TSAs & iron core from JFY’93 to ’96
- b) High-temperature test of the electro-magnet coil under the condition of cyclic supply of electric current
- c) Sodium corrosion test of the TSAs & iron core
- d) Irradiation test of the TSAs, iron core & coil in Joyo MK— III

(Components & system tests)
- a) Magnetic holding force of the SASS
- b) Temperature response of the TSAs
- c) High temperature & thermal shock tests in sodium
- d) Whole system test in Joyo MK— III

(2) Development plan for GEM

Based on the results of the preliminary design study for the application of the GEM system in the DFBR, the main technical problems have been identified as follows:
- a) confirmation of the neutronic calculation method for the evaluation of the GEM reactivity worth by conducting criticality experiments
- b) validation and improvement of the calculation model for the gas expansion behavior of GEM by conduction out—of—reactor experiments
7. Conclusion

(1) The PSA study concluded that the two independent reactor shutdown systems of DFBR reduce the occurrence probability of ATWS to less than $10^{-7}$/ry.

(2) The passive reactor shutdown system SASS was introduced to reduce the risk of core damage due to ATWS events. The installation of SASS reduces the occurrence probability of ATWS by two orders of magnitude.

(3) For a further enhancement of reactor core safety in the case of ATWS events, employment of GEM has been proposed to reinforce the passive shutdown capability against ULOF events.

(4) GEM can provide additional diversity in the shutdown system capability against ULOF, and also complement the function of SASS.

(5) The outline of future plans for the SASS and GEM development was described.

(6) In addition to SASS and GEM, ETEM could be one of the promising mechanisms to enhance core safety and has less impact on the performance of a future large scale reactor core.

REFERENCES

(1) T.Inagaki, M.Ueta, Y.Shibata, K.Tarutani, K.Okada, :
"The Development of Demonstration Fast Breeder Reactor (DFBR)", SMiRT 13th, Rio Grande do Sul, Brazil, Aug. 1995

(2) T.Hoshi, K.Harada, H.Endoh, L.Ikarimoto, :
"Development of Actuated Shutdown System for a large scale FBR", ICONE-3, Kyoto, Japan, Apr. 1995

(3) S.Kotake, S.Kasai, K.Nakai, S.Itooka, :
"Development of Enhanced Thermal Elongation Mechanism (ETEM) for a Large Scale Mixed-Oxide Fuel FBR", ICONE-3, Kyoto, Japan, Apr. 1995
IRRADIATION PERFORMANCES OF THE SUPERPHENIX TYPE ABSORBER ELEMENT

B. KRYGER, D. GOSSET
Commissariat à l'Énergie Atomique (CEA),
Centre d'Études de Saclay, DRN/DMT/SEMI/LEMA,

J.M. ESCLEINE
Commissariat à l'Énergie Atomique (CEA),
Centre d'Études de Cadarache, DRN/DEC/SDC.EMC,

Cedex, France

Abstract

Several aspects of the irradiation behaviour of the SUPERPHENIX type absorber element are presented in this paper. A large programme of irradiation tests was performed in PHENIX to assess and to improve the absorber pin design whose main characteristics for the first load are: a sodium bonded and vented pin with high density (96 % TD) and highly enriched (up to 90 at % of boron 10) B₄C pellets.

We present and discuss the main post-irradiation results obtained by this programme which concerns the behaviour of both B₄C pellets (fragmentation, swelling, helium release, thermal conductivity evolution) and stainless steel clad (embrittlement by carburization, mechanical interaction).

It appears that the residence time of the first load of SUPERPHENIX control rods is clearly limited by mechanical interaction between B₄C and the clad, and particularly by relocating of small fragments of B₄C at beginning of life in the initial gap. The irradiation performed in PHENIX led to fix the residence time of the first load of control rods to 240 e.f.p.d. The analyses of the effects limiting the residence time have enabled us to propose an extension of this time by two measures. The first one is reduction of the capture rate in boron carbide. This measure was brought into operation by mean of lowering at 48 at % the boron 10 enrichment of the B₄C pellets in the lower part of the pin. The second measure is preventing the fragment relocation by adoption of a thin stainless steel shroud enclosing the pellet stack. The efficiency of these measures was proved in several irradiation tests (ANTIMAG experiments) in PHENIX. A burn-up of 220 x 10²⁰ capt/cm³ was achieved without any dimensional change of the pin diameter. The shroud failed but could nevertheless prevent any pellet cladding deformation.

Thus, these results have enabled us to fix a residence time of 640 e.f.p.d for the third load of the SUPERPHENIX control rods.

The achievement in the future of lifetime up to 1 000 e.f.p.d. will require the development of both advanced absorbing materials and pin designs.
1. INTRODUCTION

The neutron absorbing material chosen for the absorber elements of SUPERPHENIX is boron carbide. It is desirable for economic reasons to match the endurance of the absorber elements with that of the fuel. In steady-state operating conditions, the fuel elements have been designed to achieve a residence time of 640 e.f.p.d (2 runs of 320 e.f.p.d. each). Irradiation experiments were performed in PHENIX to test the improvements made on the initial absorber pin design in order to increase the residence time from 240 e.f.p.d. (1st load) up to 640 e.f.p.d. (3rd load).

In the first part of this paper we present the criteria which determine the life of the absorber pin. In the subsequent part the basic results issued from the irradiation tests performed in PHENIX are presented with those gathered in the literature. Finally the improvements made on the initial design which led to increase the performances of the absorber pin, are described.

2. THE CONTROL RODS SYSTEMS OF SUPERPHENIX

2.1. Description

The control rods in SUPERPHENIX [1] have two main functions which are:

1) the compensation of reactivity effects due to thermal or power variations and to fuel burn-up,

2) the safety function by negative reactivity introduction to assure the reactor power decrease or shut down when incidental or accidental situation occurs.

In order to increase the diversification to improve the safety, the previous functions are assured by two systems, SCP (Système de Commande Principal - Main Control System) and SAC (Système d'Arrêt Complémentaire - Complementary Shutdown System). Each system includes several subassemblies (S/A) with a moving part - the rod containing the absorber - connected to one mechanism per S/A.

The SCP assures the above two functions and is divided into two groups SCP1 and SCP2 each one with a different type of mechanism again for diversity reasons. The SAC assures the safety function, manual or automatic shutdown. SCP1 includes 11 S/As, SCP2 10 S/As and SAC 3 S/As (Fig.1).

To reach the required rod worth efficiency with a limited number of S/A and mechanisms in the specific environment of SUPERPHENIX core, the absorber material is enriched boron carbide ($B_4C$, $^{10}B$ enrichment = 90 at.%).

The SCP subassembly (Fig.2) is realized with two parts, the outer hexagonal shell and the absorber rod. The external shape of the S/A is similar to the fuel S/A one. The absorber rod itself is composed by the pin bundle, the inner shell, the upper guide tube, and the lifting head connected to the mechanism. The bundle has 31 pins and 12 dummy pins on the periphery to improve the cooling. The pins are
sodium bonded and the gap is determined to accommodate the \( \text{B}_4\text{C} \) swelling to avoid mechanical interaction at end of life and so the gap is a function of the expected residence time of the absorber.

The pins are vented through the upper and lower plugs to release helium directly in the primary circuit. A schematic view of the absorber pin is given on figure 3.

![Superphenix Core Diagram](image)

**FIGURE 1: SUPERPHENIX CORE**
FIGURE 2: SCP SUBASSEMBLY - MAIN CONTROL ROD OF SUPERPHENIX
FIGURE 3: SCHEMATIC VIEW OF ABSORBER PIN
2.2. The absorber element end of life criteria

The theoretical end of life of the control rod is determined by loss of its reactivity worth. Practically the control rod end of life occurs earlier than expected by neutronic calculations, the reason being the failure of the absorber pin due to problems of materials damage. The life time of the absorber element may be determined by one of the following criteria [2]:

- Loss of boron 10 which reduces the effectiveness of the control rod to an unacceptable level.

- Swelling of the B\textsubscript{4}C pellets leading to pellet-cladding mechanical interaction (PCMI).

- Excessive clad embrittlement due either to radiation damage or to pellet-cladding chemical interaction (PCCI).

- Excessively high pellet temperature. The swelling rate of boron carbide is essentially independent of exposure temperature up to a temperature ~ 1500°C, above which the swelling increases rapidly with temperature. Pellet melting at ~ 2450°C is also to be avoided.

3. BEHAVIOUR OF THE ABSORBER ELEMENT

3.1. B\textsubscript{4}C thermal conductivity and structural behaviour

The thermal conductivity of boron carbide reduces very rapidly with burn-up. In the unirradiated state the conductivity decreases at T\textsuperscript{-1} with increasing measurement temperature T.

Figure 4 shows measurements performed in different laboratories on irradiated B\textsubscript{4}C at low burn-up (up to 40 x 10\textsuperscript{20} capt./cm\textsuperscript{3}) and low temperature (500°C). For these irradiation conditions, a drastic reduction in thermal conductivity (more than a factor of five relative to unirradiated material) is observed. Works performed at HEDL [3] indicate that the reduction in thermal conductivity at high irradiation temperatures is not as significant. At 1575°C the material exhibits about half the conductivity of unirradiated B\textsubscript{4}C.

In the majority of boron carbide pellets cracking generation by thermal stresses occurs on first going to power and then the size of fragments decreases, at least partly due to the reduction in thermal conductivity. Fragments of quite small size (< 1 mm) can be generated at high ratings and burn-ups.

3.2. Swelling and helium formation

Swelling is attributed to the retention of helium produced by the \textsuperscript{10}B (n,\alpha)\textsuperscript{7} Li reaction. The literature reports a general consensus on the theory that lenticular bubbles filled with helium gas are responsible for both swelling and microcracking of B\textsubscript{4}C [4-7]. Figure 5 shows collected data on materials of various densities in the
thermal conductivity (W/m.K) at 500 °C

FIGURE 4 : THERMAL CONDUCTIVITY OF IRRADIATED B₄C VERSUS BURN-UP

range of temperature (600 - 1 500°C) independant swelling. Swelling is found to be linear on burn-up Nα, i.e., on helium concentration up to Nα = 1.5 x 10²² capt/cm³.

From Fig. 5, we get a slope of 1.5 x 10⁻²³ m³/He for the higher density (96 % TD) pellets. These small values of the extra volume added by each one of the created helium atoms are very close to the values that are obtained with helium.

FIGURE 5 : EVOLUTION OF B₄C SWELLING VERSUS BURN-UP
under extremely high pressures, i.e., to the covolume term \( B \) in the Van der Waals equation \[8\]. Thus, it has been proposed \[9\] a simple, zero-order, linear model for swelling:

\[
\Delta V/V = B \alpha
\]

where \( B \) is not an adjustable parameter but just the covolume term \( B \) in the Van der Waals equation. The simplicity of this model is eloquent. At the first glance, the model disregards the bubble microstructure or shape and simply assumes that the helium is in its most compressed state, i.e., the covolume term in the Van der Waals equation.

The fact that grain boundaries are preferential traps for helium makes the main source of early intergranular microcracking.

Figure 6 shows measurements of the helium retained as a function of burn-up. In the transient period of low swelling the helium release is substantial, then it decreases to low values and then increases again for burn-up > 100 \( \times 10^{20} \) captures/cm\(^3\), although the swelling continues unabated.

![Figure 6: Measurements of helium retention in B\(_4\)C](image)

3.3. Pellet-cladding mechanical interaction

Pellet swelling is considerably faster than that due to void swelling in the cladding material. Therefore, depending on its initial value, pellet-cladding gap will tend to close at a certain burn-up. Gap closure and continued B\(_4\)C swelling then results in mechanical pellet-cladding interaction inducing cladding strains and stresses which are potential causes of absorber pin failure.
The SCP first load absorber pin, whose main characteristics are presented in table 1, is designed with a gap equal to 11.8% of pellet diameter allowing a free volumic swelling for B₄C up to about 40%. If we consider the swelling law of B₄C (96% TD) (fig. 5), it can be seen that such a pellet-cladding gap should allow the control rod to reach a burn-up equal to $250 \times 10^{20}$ capt/cm$^3$ without mechanical interaction. The irradiation experiments in PHENIX called PRECURSAB were conducted in order to determine the lifetime of the absorber pin. It appeared that the residence time of the pin was clearly limited by the mechanical interaction between the B₄C and the clad and particularly by the presence of small fragments (due to the thermal stress cracking and intergranular microcracking), which relocate at beginning of life in the initial gap, leading to clad failure when burn-up goes beyond $150 \times 10^{20}$ capt/cm$^3$ (fig. 7). As a consequence of B₄C dispersion in the gap we also observe that the B₄C column length after irradiation is lower than that expected from swelling. Figure 8 shows the evolution of B₄C diameter increase due to the combination of swelling and relocation, inducing a premature pellet-cladding mechanical interaction.

### 3.4. Pellet-cladding chemical interaction

It is well known that the presence of boron and carbon in stainless steel produces severe embrittlement and for the long proposed lifetimes these penetration effects can be limiting factors. Figure 9 illustrates the carbon penetration in the cladding on the experimental vented and sodium bonded pin PRECURSAB A1 irradiated to 250 e.f.p.d. in PHENIX.

<table>
<thead>
<tr>
<th>Design</th>
<th>Sodium bonded and vented pin</th>
</tr>
</thead>
<tbody>
<tr>
<td>Material</td>
<td>B₄C</td>
</tr>
<tr>
<td>Boron 10 enrichment</td>
<td>90 at %</td>
</tr>
<tr>
<td>Density</td>
<td>96% TD</td>
</tr>
<tr>
<td>Pellet diameter</td>
<td>17 mm</td>
</tr>
<tr>
<td>Column length</td>
<td>1 145 mm</td>
</tr>
<tr>
<td>Cladding material</td>
<td>316 Ti</td>
</tr>
<tr>
<td>Cladding diameter</td>
<td>19 x 21 mm</td>
</tr>
<tr>
<td>Pin length</td>
<td>1 311 mm</td>
</tr>
</tbody>
</table>
FIGURE 7: EXPERIMENTAL ABSORBER PIN PRECURSAB A1 IRRADIATED IN PHENIX. RELOCATION OF B$_4$C FRAGMENTS (2 AND 3) IN GAP AND SUBSEQUENT CLAD FAILURE (1).

FIGURE 8: DIAMETRAL EXPANSION OF B$_4$C COLUMN DUE TO SWELLING AND RELOCATION
4. ABSORBER PIN DESIGN FOR THE THIRD LOAD OF SUPERPHENIX

The irradiations performed in PHENIX have led to fix the residence time of the first load of SCP to 240 e.f.p.d. The analyses of the limiting effects on the residence time have allowed to propose an increase of this time by two cumulative ways. The first one is the reduction of the capture rate in boron carbide. As in a pin the shape of the capture rate is very sharp it is possible to reduce the maximum rate by decreasing the number of boron 10 atoms in the lower part without significant effect on the rod worth. This can be achieved with use of less enriched boron carbide or by a lower density.

The solution which has been chosen for the second SCP load (320 e.f.p.d.) and the following loads, consists to adopt a 48 at % of boron 10 enrichment for the lower part of the pins, (Fig. 10).

The second improvement made on the initial pin design, is to prevent the fragments relocation. It consists of enclosing the pellets stack inside a confining shroud (thin tubular sheath) made with similar material as the cladding.

Four irradiation tests carried out in PHENIX (ANTIMAG 2 and 3 in a capsule, HYPERBARE 1 and 2 in a control rod) have allowed both improvements to be validated successfully. ANTIMAG 3 experiment achieved the maximum performance (657 e.f.p.d. and 220 x 10^{20} capt./cm^{3}) without clad deformation. The behaviour of this experimental absorber pin is illustrated by figure 11 on which we can observe the presence of a residual gap as a result of the beneficial effect of the shroud. Carbon local measurements from X-rays microbobe analysis, reveal a similar level of carburization (1.8 w % C) at surface of shroud and clad.

\[
\begin{align*}
\text{PRECURSAB A1 (M316 CLADDING)} \\
\text{Effective Diffusion Coeff.} & \quad D_{\text{eff}} = 4.5 \cdot 10^{-13} \text{cm}^2/\text{s} \\
\text{Clad Internal Temperature} & \quad 565^\circ \text{C (Max.)} \\
\text{Time} & \quad 250 \text{ EFPD}
\end{align*}
\]

**FIGURE 9**: PENETRATION OF CLADDING BY CARBON DIFFUSION
FIGURE 10: AXIAL REPARTITION OF BURN-UP IN THE FIRST AND SECOND LOADS OF SUPERPHENIX SCP ABSORBER PINS

FIGURE 11: ILLUSTRATION OF THE USE OF A TUBULAR SHROUD IN EXPERIMENT ANTIMAG 3 (657 e.f.p.d. - 220 x 10^{20} capt./cm^3)
Considering the overall irradiation experiments carried out in PHENIX, the improved absorber pin whose main specifications are given in table 2, has been introduced in the third SCP load for a residence time of 640 e.f.p.d.

### TABLE II: CHARACTERISTICS OF THE THIRD LOAD OF SUPERPHENIX SCP ABSORBER PIN

<table>
<thead>
<tr>
<th>Design</th>
<th>Sodium bonded and vented pin Double enriched and shrouded column</th>
</tr>
</thead>
<tbody>
<tr>
<td>Material</td>
<td>B₄C</td>
</tr>
<tr>
<td>Density</td>
<td>96% TD</td>
</tr>
<tr>
<td>Pellet diameter</td>
<td>17.4 mm</td>
</tr>
<tr>
<td>Column length</td>
<td>1145 mm</td>
</tr>
<tr>
<td>Boron 10 enrichment</td>
<td>90 at % in upper part (995 mm) 48 at % in lower part (150 mm)</td>
</tr>
<tr>
<td>Cladding material</td>
<td>15 - 15 Ti C.W.</td>
</tr>
<tr>
<td>Cladding diameter</td>
<td>19.8 x 21.6 mm</td>
</tr>
<tr>
<td>Shroud material</td>
<td>316 Ti or 15 - 15 Ti</td>
</tr>
<tr>
<td>Shroud diameter</td>
<td>17.6 x 18 mm</td>
</tr>
<tr>
<td>Pin length</td>
<td>1318 mm</td>
</tr>
</tbody>
</table>

5. CONCLUSION

The design of the SUPERPHENIX SCP absorber pin was a significant evolution from the PHENIX one and the performances required were far above. A lot of irradiation experiments were performed to support the design. If the overall design is well confirmed and functions required totally fulfilled, the analyses of the irradiation tests have led to limit the residence time of the SCP first load at 240 e.f.p.d. But significant improvements made on the initial design (adoption of two axial boron 10 enrichment zones in the B₄C column and a shroud to prevent relocation of B₄C fragments), allow the residence time of the SCP third load to be fixed at 640 e.f.p.d., which corresponds to the residence time of the fuel pin.

To achieve in the future a residence time of 1 000 e.f.p.d., it will be necessary to test the endurance of the absorber pin at burn-ups up to about 300 x 10^{30} capt./cm³. Such an increase of life time and burn-up will require to find new solutions to delay the advent of both mechanical and chemical interactions between B₄C pellets and cladding.
REFERENCES


Abstract

This paper reviews the major experimental results on different types of control rods with boron carbide and europium oxide which have been tested in the BN-600 reactor during the operating period.

INTRODUCTION

Shime rods (SHRs) are used for a compensation of an inlet core temperature and thermal power reactivity effects and burn-up effect. There are 19 SHRs in the core.

Automatic power control rods (ACRs) are used for automatic maintaining and change of the preset reactor power level. There are 2 ACRs in the core.

Scram rods (SCRs) are used to scram the reactor in emergency situations (all SCRs) or to decrease its power abruptly by 1/3 of preset power level in the mode of tripping one cooling loop (one rod, named SCR-L). There are 5 SCR and one SCR-L in the core.

During the operating period 5 different types of SHRs and 3 types of ACRs with boron carbide and europium oxide as an absorber and 7 types of SCRs with boron carbide of various enrichment level as an absorber were tested in the BN-600 reactor. Basic ideas behind rod testing were the following:

- the utilization of new structural materials more irradiation resistant for longer lifetime;
- the validation of "neutron trap" design using ZrH₂ as a moderator of annular and rod shape as well as absorbers of annular and rod shape to decrease the absorbing material amount in a rod;
- functional testing of sealed absorbers.
SHIME RODS

Shime rods (SHR) are used for a compensation of an inlet core temperature and thermal power reactivity effects as well as reactivity changes due to the burning-up of the fuel in fuel assemblies. The basic design SHR characteristics and the operation conditions are presented in Tables 1, 2, and 3; principle design is shown in Fig. 1.

Rods, defined as 1161, 1161A, and 1161B in Technical documentation, were designed as SHRs for BN-600. Their principle distinction is the absorbing material used. For the rod 1161 the absorber is made of B₄C while in rods 1161A and 1161B Eu₂O₃+Mo is used. The peculiarity of all the rods is the chromium-nitride coating of the hinges preventing them from selfwelding with a case during operating under the influence of high temperatures. The design of the 1161A and 1161B rods is the same. The difference is only in absorber manufacturing technology. The absorber of the 1161A rod is made of the hot-pressed pellets while the absorber of the 1161B rod is made of powder. As the basic variant the rod based on europium oxide was adopted and for the 1st charge SHRs of the 1161A and 1161B type were produced and installed.

Operation experience showed that the main factor limiting the life-time of the said rods was the sufficient shape-changing of the parts of the bottom hinge under the influence of irradiation. That led to the jamming of a rod during the 3rd recharging and produced some difficulties for the extraction of two rods. The following measuring of the spacing part diameters of the bottom hinge of the said rods showed their increasing up to 2.4%.

To prevent this the second set of rods was improved in the following manner (prior inserting into the reactor):

- spacing part diameters of the bottom hinge were decreased from 90 to 89.2-89.3 mm;
<table>
<thead>
<tr>
<th>Characteristics</th>
<th>Shime Rod</th>
<th>Control Rod</th>
<th>Scram Rod</th>
</tr>
</thead>
<tbody>
<tr>
<td>1161</td>
<td>1161A, 1161B</td>
<td>1161B</td>
<td>1157A</td>
</tr>
<tr>
<td>1. Hinges amount, pcs.</td>
<td>2</td>
<td>2</td>
<td>2</td>
</tr>
<tr>
<td>2. Working links amount, pcs.</td>
<td>1</td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td>3. Absorbers amount in a working link, pcs.</td>
<td>19</td>
<td>48</td>
<td>8</td>
</tr>
<tr>
<td>4. Case diameter, mm</td>
<td>89x1.5</td>
<td>89x1.5</td>
<td>85x1</td>
</tr>
<tr>
<td>5. Absorber cladding diameter, mm</td>
<td>15.6x0.1</td>
<td>9.5x0.5</td>
<td>32x0.7</td>
</tr>
<tr>
<td>6. Case material</td>
<td>0X18H10T</td>
<td>06X18H10T</td>
<td>08X18H10T</td>
</tr>
<tr>
<td>7. Absorber cladding material</td>
<td>ЭИ-847</td>
<td>ЭИ-847</td>
<td>ЭИ-847</td>
</tr>
<tr>
<td>8. Hinge material</td>
<td>12X18H10T</td>
<td>12X18H10T</td>
<td>X18H10T</td>
</tr>
<tr>
<td>9. Absorbing compound length, mm</td>
<td>780</td>
<td>780</td>
<td>740</td>
</tr>
<tr>
<td>10. Absorbing compound</td>
<td>B₄C</td>
<td>Eu₂O₃+Mo</td>
<td>B₄C</td>
</tr>
<tr>
<td>11. Concentration of B¹⁰, %</td>
<td>natural</td>
<td>-</td>
<td>natural</td>
</tr>
<tr>
<td>12. Absorber type</td>
<td>sealed</td>
<td>sealed</td>
<td>non-sealed</td>
</tr>
<tr>
<td></td>
<td>pellets</td>
<td>pell., powder</td>
<td>pellets</td>
</tr>
</tbody>
</table>
TABLE 2. OPERATING CONDITIONS FOR PERMANENT CONTROL AND SCRAM RODS OF BN-600 REACTOR

<table>
<thead>
<tr>
<th>Characteristics</th>
<th>Shime Rod</th>
<th>Control Rod</th>
<th>Scram Rod</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1161</td>
<td>1161A, 1161B</td>
<td>1161B</td>
</tr>
<tr>
<td>1. Physical effectiveness, not less than, % dk/k</td>
<td>0.27</td>
<td>0.27</td>
<td>0.27</td>
</tr>
<tr>
<td>2. Fuel burn-up $^{10}B$, %</td>
<td>24</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>(per year)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>3. Operating time in the reactor (resource), eff.d.</td>
<td>180</td>
<td>180</td>
<td>180</td>
</tr>
<tr>
<td>4. Rated sodium discharge through the case, kg/c.</td>
<td>3.6</td>
<td>3.6</td>
<td>3.6</td>
</tr>
<tr>
<td>5. Maximum absorber cladding temperature, °C</td>
<td>620</td>
<td>493</td>
<td>493</td>
</tr>
<tr>
<td>6. Case inlet temperature, °C</td>
<td>380</td>
<td>380</td>
<td>380</td>
</tr>
<tr>
<td>7. Operating media</td>
<td>liquid sodium</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
TABLE 3. CONTROL AND SCRAM RODS IRRADIATED IN BN-600 REACTOR

<table>
<thead>
<tr>
<th>№</th>
<th>Type</th>
<th>Amount, pcs</th>
<th>Maximum damaging dose range, eff. d.</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.</td>
<td>SCR 1663</td>
<td>65</td>
<td>10-47</td>
</tr>
<tr>
<td>2.</td>
<td>SCR 1663-01 *</td>
<td>12</td>
<td>10-45</td>
</tr>
<tr>
<td>3.</td>
<td>SCR 1641</td>
<td>2</td>
<td>18-19</td>
</tr>
<tr>
<td>4.</td>
<td>SCR 2233</td>
<td>4</td>
<td>37-45</td>
</tr>
<tr>
<td>5.</td>
<td>SCR e9.1583</td>
<td>1</td>
<td>31</td>
</tr>
<tr>
<td>6.</td>
<td>SCR e9.1584</td>
<td>1</td>
<td>46</td>
</tr>
<tr>
<td>7.</td>
<td>SCR e9.1469</td>
<td>2</td>
<td>33-49</td>
</tr>
<tr>
<td>8.</td>
<td>SCR 2637 *</td>
<td>1</td>
<td>64</td>
</tr>
<tr>
<td>9.</td>
<td>SCR 2633</td>
<td>1</td>
<td>63</td>
</tr>
<tr>
<td>10.</td>
<td>SHR 1161</td>
<td>3</td>
<td>18-19</td>
</tr>
<tr>
<td>11.</td>
<td>SHR 1161A</td>
<td>40</td>
<td>19-50</td>
</tr>
<tr>
<td>12.</td>
<td>SHR 1161B</td>
<td>78</td>
<td>14-79</td>
</tr>
<tr>
<td>13.</td>
<td>SHR 1161B</td>
<td>99</td>
<td>14-56</td>
</tr>
<tr>
<td>14.</td>
<td>SHR 2635</td>
<td>9</td>
<td>66</td>
</tr>
<tr>
<td>15.</td>
<td>ACR 1157A</td>
<td>23</td>
<td>15-45</td>
</tr>
<tr>
<td>16.</td>
<td>ACR 2415</td>
<td>2</td>
<td>45-68</td>
</tr>
<tr>
<td>17.</td>
<td>ACR 2631</td>
<td>1</td>
<td>66</td>
</tr>
</tbody>
</table>

SCR - scram rod
SHR - shime rod
ACR - control rod

* - Scram rod for tripping one cooling loop.
Fig. 1. Shime rod with a case design.

a) 1161A rods with a round case
b) 1161B rods with a hexahedron case
- tip diameters were decreased from 80 to 78 mm.

Beginning from July, 1982, in order to decrease irradiational heat in the massive hinge joints and to improve the conditions for the case contact, the design of the spacing elements, hinges, and shank tail part of 1161A and 1161B rods was changed.

In 1982-1983 8 rods of 1161E-3n type were tested. For their hinges ferrito-martensite steel 3Π-450 was used instead of 12X18H10T steel. The measuring of the dimensions of the spacing rod parts made of this steel showed that after operating cycle they increased slightly (appr. 0.1%) but the diameter of the working link jacket pipes (12X18H10T steel) swelled to the hinge dimensions.

Despite high irradiation resistance and absorber efficiency the considerable disadvantage of SHR with Eu-based absorber is the high induced activity and the residual energy release due to the accumulation of highly active Eu-152 and Eu-154 long-live isotopes with hard gamma-irradiation spectrum (a half-life of 13 and 16 years correspondently).

Because of that in 1984 the advanced type of SHR was introduced, defined as 1161B (Pic.3.1), the working link of that rod comprised 8 absorbers, 23 mm in diameter and 740 mm in length. Seven absorbers were uniformly spread along the rod circle and one absorber was put into the middle. Fourteen displacers were spread in the empty space between the absorbers - 7 in the inner row and 7 in the outward row. Thus for 1161B rods absorbers unified with csram rods were used (non-sealed ones with an absorber of boron carbide of natural concentration); the cooling of the scramble cases was improved due to the utilization of jacket pipe 85 mm in diameter instead of 89 mm. The structural material for them is exactly the same as for 1161A and 1161B rods.

This modernization allowed to increase the life-time of the SHR up to 365 eff. days. Such SHR design is currently used in the core of BN-600 reactor.

In 1990-1991 an advanced SHR 2635 with a life-time of 550 eff. days was tested. The main distinguishing feature of the rods is the utilization of 3Π-450 steel
for jacket pipes, hinges, and shank parts, and the use of austenised 4C-68 steel - for absorbers claddings. The rod was designed to operate for 550 eff. days. Although it managed to operate without a complain for 502 days only due to the reactor operating conditions.

In the operation course the shape-change control of the SHR was regulary performed to correct life-time resource. The diameter of the spacing belts of the bottom hinge and that of the spacing part of rod extension was measured. The measurements took place in a "cooling" pond and in a "hot cell". Total 58 rods were tested including 19 - 1161A, 28 - 1161B (of them 6 - 1161E-3П), 10 - 1161B, 1 - 2635.

CONTROL RODS

Automatic control rods (ACRs), of 1157F type, are designed to automatically maintain and change the preset reactor power level. The basic design ACR characteristics and its operation condition are presented in Table1, 2, and 3; the principle design is shown in Fig.2.

The 1st variant of ACR had the same disadvantages as the SHR with oxide absorber. As well as for the SHRs there was a rod jam in a case during a recharging. As a result the rod design was sufficiently changed. Mostly it related to the absorber and the hinge joint. The absorber was changed from a pellet to a powder made (just like for the 1161B rod). As far as the hinge was concerned the chromium-nitride coating was excluded, the weight was decreased and the cooling conditions changed. Five 1157A rods were studied to control the shape-changing and the results obtained were the same as for SHRs.

2415 and 2631 rods were tested in the reactor as experimental ones. The 1st differed from the permanent ACR in the construction material selected for cap, shank, hinges, and jacket pipe (more irradiation resistant 3П-450 steel) manufacture. The rod was charged into the reactor for 312 eff. days, the maximum increase of the bottom hinge diameter was 0.1%. As far as the 2nd ACR is concerned, the difference was not only in the use of
Fig. 2. Control rod with a case design.
   a) 1157A rods with a round case
   b) 1157A rods with a hexahedron case
irradiation resistant 311-450 and UC-68 steels but also Eu$_2$O$_3$+Mo was replaced by B$_4$C. This ACR was operated for 502 eff. days and its dimensions did not changed.

**SCRAM RODS**

Scram rods (SCRs) were designed to scram the reactor in emergency situations (all SCRs) or to decrease its power abruptly by 1/3 of preset level in the mode of tripping one cooling loop (1 SCR). The basic design characteristics of permanent SCR and its operating conditions are presented in Table 1, 2, and 3; principle design is shown in Fig. 3.

Beginning from the 1st microcampaign, SCRs 1663 and 1663-01 were used as permanent rods. They were practically the same in design, the basic difference between them was the amount of B$_{10}$ content in absorber composition. During the first years of operation there also were some difficulties in recharging due to the swelling of the bottom shank of the bottom link - 2 SCRs were jammed in recharging box. That resulted in SCRs modernization. The main changes were made to a hinge joint, absorber, and bottom link:

- the cooling conditions for a hinge joint were improved and chromium-nitride coating was excluded;
- the length of the absorber in top working link was decreased by 84 mm, from 414 mm to 330 mm, because this part did not substantially affect the total rod effectiveness (the position of the bottom border of the absorber on the rod was not changed);
- the diameter of the bottom link mostly affected by the irradiation was decreased from 74 mm to 68 mm.

Shape-changing control was performed for 10 1663 SCRs and for 3 - 1663-01. The bottom shank diameter was measured. The results obtained coincided with that obtained for SHRs and ACRs.
Fig. 3. Scram rod with a case design.

a) 1663, 1663-01 rods with a round case
b) 1663, 1663-01 rods with a hexahedron case

1 - directing pipe of executive facility
2 - directing case of the rod
3 - rod
4 - fuel assemblies surrounding the rod
Seven types of rods were tested as SCRs. The basic ideas in their design were:

- the utilization of advanced irradiation stable structural materials to prolong the life-time (2637, 2633 type);
- the validation of "neutron trap" design using ZrH$_2$ as a moderator of annular and rod shape as well as absorbers of annular and rod shape to decrease the amount of absorbing material in a rod (2233, e3.1583, e3.1584, e3.1460 type);
- functional testing of sealed absorbers (1641, e3.1460 type).

All the experimental rods were functioning during the defined time without any faults.
THE EXPERIENCE OF POST IRRADIATION INVESTIGATIONS
OF THE BN-600 CONTROL RODS

V.P. TARASIKOV, R.M. VOZNESENSKI, V.A. RUDENKO
IPPE, Obninsk, Russian Federation

Abstract

The paper presents the results of post-irradiation examination of the BN-600 reactor control rods. The rods were in operation for more than 500 days, and burn-up was $18.3 \times 10^{21}$ capt/cm$^3$.

1. INTRODUCTION

The control rods of the reactor BN-600 were developed according to their functional purpose and the experience of operation of the similar rods of the Bor-60, BN-350 reactors. The design of the BN-600 control rods of the first loading is considered in detail in the reports of Soviet specialists at the Meetings of International Working Group on Fast Reactors of IAEA. Thus, in this paper it won't be adduced.

The life-time of the control rods of the initial complete set was ~200 EFPD and it was limited by the dimensional changes due to irradiation of the parts of the rods working in the core. According to that, the design of the control rods and wrappers was changed with the aim to increase its reliability and life-time. For the experimental confirmation of the design characteristics of the rods and the efficiency of their design modifications the carrying out of the post irradiation investigation of the spent rods was planned. Two stages of rods upgrading could be distinguished according to the rods modifications.

At the first stage (1981-1985), the modifications of the parts of rods that allowed to decrease its swelling were made. The use of hexagonal guide tube practically excluded the temperature irregularity on the perimeter of guide tubes and the bowing caused by it.

In 1984-1985 four experimental compensating rods (CpR) with natural boron carbide (S/A №1161B) were tested in the BN-600 reactor for the life-time of 310, 406 EFPD (damage dose ~64 dpa, burn-up $5.2 \times 10^{21}$ capt/cm$^3$). In the hot cell of Institute of Physics and Power Engineering (IPPE) two of the above rods and the rod CpR 1161 with the sealed absorber pins and operation time of 94 EFPD were investigated.

The results of investigations permitted to substitute the rods CpR with Eu$_2$O$_3$ by the boron rods. The life-time of the boron rods is determined as 365 EFPD and it is limited by the efficiency of structural material.

During the second stage of upgrading (1986-1994), the post irradiation investigations of standard rods were carried out. In 1990 the regular safety rod SR (S/A 1663) with time of operation of 312 EFPD, dose ~44 dpa, burn-up $7.3 \times 10^{21}$ capt/cm$^3$ was investigated in IPPE. The life-time of this rod was determined as 365 EFPD and also it was limited by the efficiency of structural material.

At this stage of upgrading, the structural material of the absorber cladding and elements of the rod was replaced by the material with higher irradiation resistance (ЧC-68 and ΗП-450 steels, respectively). In the regulating rod RR the natural boron carbide was used. The rods SR, SR-L (the safety rod for drop in power in case of one heat removal loop cutting off), RR, CpR were in nominal operation 502.3 EFPD (dose ~70 dpa, burn-up of the enriched and natural boron carbide $18.3 \times 10^{21}$ and $8.7 \times 10^{21}$ capt/cm$^3$). The post irradiation investigations of these rods showed that after operation the rods saved the margin of performance.

In this paper the results of post irradiation investigations of the experimental rods of the reactor BN-600 with the life-time of 500 EFPD are given. For the comparison the results of investigations of the rods of the BN-600 and BN-350 reactors with shorter life-time are also given.
2. THE DESCRIPTION OF RODS DESIGN

The principal differences of the upgraded rod from the regular one are:

- the upgraded rod has one working section;
- as structural materials the 311-450 steel (for ducts, for the parts of hinges and feet) and the ЧГ-68 steel (for the cladding) are used;
- in the control rod the boron carbide with higher B\(^{10}\) content (92% at.) is used;
- in the regulating rod the natural boron carbide was used instead of absorber on the base of Eu\(_2\)O\(_3\).

The main design characteristics of the rods are presented in the Table I.

### TABLE I. THE MAIN DESIGN CHARACTERISTICS OF THE EXPERIMENTAL RODS

<table>
<thead>
<tr>
<th>№</th>
<th>type, parameter</th>
<th>SR (S/A №2637)</th>
<th>SR-L (S/A №2633)</th>
<th>RR (S/A №2631)</th>
<th>CpR (S/A №2635)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>length of the rod, mm</td>
<td>2100</td>
<td>2100</td>
<td>2100</td>
<td>2874</td>
</tr>
<tr>
<td>2</td>
<td>diameter of the duct of working section, mm</td>
<td>73 x 1</td>
<td>73 x 1</td>
<td>73 x 1</td>
<td>85 x 1</td>
</tr>
<tr>
<td>3</td>
<td>external diameter of the rod on the hinges, mm</td>
<td>74</td>
<td>74</td>
<td>74</td>
<td>88</td>
</tr>
<tr>
<td>4</td>
<td>number of the absorber pins</td>
<td>7</td>
<td>4</td>
<td>4</td>
<td>8</td>
</tr>
<tr>
<td>5</td>
<td>diameter of the absorber pin, mm</td>
<td>23 x 0.7</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>6</td>
<td>absorber ((B^{10}, % \text{ at.}))</td>
<td>B(_4)C (92% at.)</td>
<td>B(_4)C (natural)</td>
<td>B(_4)C (natural)</td>
<td>B(_4)C (natural)</td>
</tr>
<tr>
<td>7</td>
<td>length of absorber pin, mm</td>
<td>800</td>
<td>850</td>
<td>850</td>
<td>930</td>
</tr>
<tr>
<td>8</td>
<td>diameter of absorber, mm</td>
<td></td>
<td></td>
<td></td>
<td>19.6</td>
</tr>
<tr>
<td>9</td>
<td>width of axial gap, mm</td>
<td>28</td>
<td>45</td>
<td>45</td>
<td>28</td>
</tr>
<tr>
<td>10</td>
<td>material of the duct, parts of the hinges and tips, mm</td>
<td>13Cr-2Mo-Nb-P-B (ЧГ-450)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>11</td>
<td>material of the cladding and replacer</td>
<td>16Cr-15Ni-2Mo-2Mn-Ti-P-B (ЧГ-68)</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

The use of one-section design and boron carbide with higher enrichment permitted to increase the efficiency of the rods by ~20%. As the result of the use of less swelling structural materials the calculated life-time of the developed rods is raised from 365 EFPD up to 550 EFPD.

3. THE CONDITIONS OF TEST

The experimental rods were in operation in the BN-600 reactor for -502 EFPD. The conditions of operation are presented in the Table II.

4. THE RESULTS OF POST IRRADIATION INVESTIGATIONS

During the tests of the rods in the reactor (during charge, discharge and operation) there were no troubles with moving. The efficiency of the rods corresponded to design values accounting boron burn-up.

4.1. Structural material

No changes of the rods construction able to make worse their performance were detected. There were no mechanical deformations; the hinges saved their original mobility.
The changes which took place in the construction of the rods are unimportant. The maximum increase of rods diameter was 0.2-0.4 mm, the decrease of the gap between guide tube and rod was 4-8%.

The material of ducts and feet retained sufficient workability after irradiation. Its mechanical properties are characterized by the high level of strength and a marked margin of residual ductility (at the dose of 70 dpa at the nominal temperature - $\sigma_u=1000$ MPa, $\sigma_y=800$, the uniform relative lengthening $\delta_u=0.8\%$). The brittle fracture of the samples wasn't observed.

The claddings changed their sizes as a result of swelling. The maximum swelling of the cladding in the rods SR, SR-L, RR, CpR was 0.7, 1.6, 5.3 and 5.7\%, respectively. The researches of the mechanical properties at the nominal temperatures showed that the irradiation of cladding led to the increase of ultimate strength from 500 to 835 MPa and decrease of ductility from 23 to 2\%.

4.2. Absorber material

As the absorber material the boron carbide produced by direct synthesis of carbon and boron was used. It is used in the form of cylindrical pellets with the density ~2.1 g/cm$^3$, made of boron carbide powder by sintering and pressing. The pellet stack is enclosed in the stainless steel cladding with the axial and radial gaps to compensate its swelling. In the investigated SRs the vented absorber pins are used. The sodium floods through the weep in the top plug during rod loading; the release of helium and H$^+$ produced in B$_4$C also carries up through this weep.

The conditions of absorber tests are presented in the Table II.

The results of these and earlier investigations of the safety rods of BN-350 and BN-600 reactors permitted to reveal basic laws of boron carbide performance under irradiation (in case of vented pins) for the validation of life-time of fast reactor safety rods. The obtained results of boron carbide performance under irradiation are given below.
4.2.1. Dimensional changes of B\textsubscript{4}C pellets (swelling)

The swelling of boron carbide in the range of studied temperatures and burn-ups (up to $19 \times 10^{21}$ capt/cm\textsuperscript{3}, 450-900 °C) directly depends on the burn-up at the constant irradiation temperature. The rate of B\textsubscript{4}C swelling (the ratio of pin diameter changing (in percentage) to the quantity of burned up boron atoms expressed in per cents) depends on irradiation temperature non-linearly. The minimum rate of B\textsubscript{4}C swelling is ~0.1-0.4 and it was observed in the range of temperature between 470 °C and 500 °C. At the temperature lower than 500°C it decreases with the temperature increase; and at the temperatures higher than 500°C it increases.

The ratio of pin diameter changing for the enriched boron carbide (60-90% at.) within the range of temperature of 400-850°C does not exceed 0.6% per % at. of boron burn-up (fig.1). For natural boron carbide it does not exceed 1.0% per % at. of burn-up within the range of 450-750°C (fig.2).

The investigations of boron carbide from the vented pins shown that the interaction between sodium penetrating into the pellet cracks and moisture during pin dismantling and pellets washing leads to additional dimensional increasing of pellets.

The investigations of absorber from the experimental rods SR, SR-L, CpR, and RR shown that pellet-cladding gap is retained. The maximum increase of pellets diameter was, respectively, 6.1, 2.9, 7.6, and 7.9% (the permissible increase - 10.2%).

**FIG. 1.** Linear swelling rate of boron carbide (60-90 at. % B-10) as a function of irradiation temperature.
4.2.2. Structural changes of boron carbide

During irradiation the cracking of the $B_4C$ pellets occurs. It has a complex character showing the participation of different cracking phenomena in this process. According to the character of cracks, three main cracking phenomena could be distinguished.

The first is connected with the thermal stress effect. The cracks caused by it were directed inside the pellets from its surface. This type of cracking is not connected with the absorbers burn-up because the decrease of the thermal conductivity under irradiation completes as a whole at the small burn-up and the thermal stress level is fixed in a short time after reactor going to the full power. The longitudinal cracking is observed in multitude when the radial temperature changing is about 110-120 °C.

The second is associated with the appearance of the stresses connected with the irregularity of the burn-up along the pellet cross-section. It is displayed in the form of cracks directed along the side surface of pellet and belongs only to the absorber of safety rods operated in the softened spectrum.

The third is connected with the violence of the absorber structure effected by the products of the $B^{10}(n,\alpha)Li^7$ reaction and displayed in the form of fine cracking of the absorber because of its saturation by micro fractures. The micro fractures have an arbitrary direction and reflect technological imperfections of the absorber; the presence of sodium apparently promotes to the activation of this process. This effect is displayed with the achieving of the burn-up level of $4\cdot6\times10^{21}$ capt/cm$^3$.

It should be noted, that the $B_4C$ pellet cracking during irradiation does not lead to its fragmentation for the vented sodium bonded absorber pins. As the result of cracking, the stress relaxation of pellets occurs. The pellets extracted from the pin have a visible cracking net, but remain as if intact. During the investigations of big number of vented pins it was observed no case of destruction of the cladding by the absorber fragments got into pellet-cladding gap.
4.2.3. Thermal conductivity of boron carbide

The thermal conductivity of boron carbide is an important characteristic, influencing on its irradiation stability parameters (helium release, structural and dimensional changes).

The irradiation leads to the essential decrease of the thermal conductivity; here it is observed initially sharp and then smooth (with the damped rate) decrease of the thermal conductivity. For the synthesized boron carbide the initial thermal conductivity is 15-25 W/(m·°K). At the burn-up ~1.5*10^{21} capt/cm³ the decrease of thermal conductivity is about 70% of initial one (up to ~4 W/(m·°K)). The increasing of irradiation temperature reduces the rate of the thermal conductivity decrease. It could be affirmed, however, that there is no outstanding recovery of B₄C thermal conductivity at the burn-ups more than 0.5*10^{21} capt/cm³ and the irradiation temperature not more than ~870°C (the maximum average B₄C temperature in the reactor BN-600 is 700°C for the safety rods and 550°C for CpR, RR).

![FIG. 3. Thermal conductivity as a function of boron carbide burn-up.](image)

4.2.4. Helium retention and release

The problem of helium pressure on the cladding does not exist in the vented absorber pins. Thus, the data on helium release are not presented here in detail and helium retention is considered only in the meaning of influence on the structural and dimensional absorber changes.

It could be noted, that the helium release depends on the technological parameters (density, granule value, impurity presence) and the irradiation conditions (temperature, burn-up). With the increasing of the burn-up up to ~1.5*10^{21} capt/cm³ initially the increasing of helium release occurs and then it decreases with the achieving of the practically stationary value at the burn-up level ~4*10^{21} capt/cm³ (Fig. 4). For example, the maximum helium release from the natural B₄C was ~55% at the burn-up ~1.5*10^{21} capt/cm³ and irradiation...
temperature ~700 °C; at the burn-up \( \sim 4 \times 10^{21} \text{ capt/cm}^3 \) the helium release was ~20% of the produced one. The helium release from the enriched B\(_4\)C under the same conditions was less.

### 4.2.5. Pellet-cladding interaction

The stainless steel (type 16Cr-15Ni, thickness 0.7 mm) is used as a cladding in the absorber pins. The pellet-cladding interaction leads to the severe embrittlement of cladding material as a result of carbon and boron penetration effects. To evaluate the thickness of the interaction layer, the data of rig tests are usually used. It was showed by the post irradiation investigation, that the irradiation decreases the temperature level of the beginning of pellet-cladding interaction by 100-120°C compared with the level obtained under the rig conditions. The intensity of the pellet-cladding interaction rises with the burn-up increase.

In the BN-600 absorber rod with moderator the interaction layer between B\(_4\)C and cladding (ЭИ-847 steel) after irradiation for the 7500 hours was up to 100 mkm for the softened spectrum. The temperature was about 430°C, the average burn-up \( \sim 17 \times 10^{21} \text{ capt/cm}^3 \).

During the investigation of the BN-600 control rods with the ЧС-68 steel cladding and with operation time of 500 EFPD (12000 hours) in the temperature range of 390-520 °C, no pellet-cladding interaction was observed.
5. CONCLUSION

The BN-600 experimental control rods containing boron carbide were in operation for 500 EFPD (burn-up up to $18.3 \times 10^{21}$ capt/cm$^3$ and irradiation dose up to 70 dpa). According to the results of their operation and post irradiation investigations, it could be concluded the following:

- There were not detected any changes of rods construction able to make worse their performance. There were no mechanical deformations; the hinges retained their original mobility. The maximum increase of rods diameter was 0.2-0.4 mm.
- The rods retained the margin of performance; the pellet-cladding gap is retained; there is no pellet-cladding interaction (material of cladding - 9C-68 steel); the structural material has sufficient margin of the mechanical characteristics.
- At the reached level of absorber burn-up ($18.3 \times 10^{21}$ capt/cm$^3$), the obtained earlier characteristics of its irradiation resistance didn't change. To evaluate the life-time of safety rods by the swelling of absorber it is possible to use the following values of its linear swelling rate: 0.6% per % at. of boron burn-up for the enriched B$_4$C; and 1.0% per % at. of boron burn-up for natural B$_4$C.
- For calculation of the absorber temperature at the burn-up more than $1.5 \times 10^{21}$ capt/cm$^3$ it is recommended to use the value of thermal conductivity equal to 4.0 W/m$^0$K at rated reactor power.
- The helium release at the burn-up of $1.5 \times 10^{21}$ capt/cm$^3$ in the range of absorber operation temperature was 55%. With the raising of burn-up the helium release reduces to 20% of produced.
IRRADIATION BEHAVIOR OF BORON CARBIDE NEUTRON ABSORBER

T. KAITO
Nuclear Fuel Research Section, Advanced Technology Division

T. MARUYAMA, S. ONOSE, H. HORIZUHI
Material Monitoring Section, Fuels and Materials Division

Oarai Engineering Center,
Power Reactor and Nuclear Fuel Development Corporation,
Ibaraki, Japan

Abstract

Boron carbide pellets were irradiated up to $230 \times 10^7$ cap/m$^3$ burnup at maximum temperature of $1400^\circ$C in "JOYO" MK-II core. Pin puncturing tests, density and thermal conductivity measurements were performed on these pellets, and the irradiation behaviors were evaluated. It is concluded that the high burnup boron carbide pellets take different irradiation behavior from that of low burnup pellets. If the burnup and the irradiation temperature are greater than about $100 \times 10^7$ cap/m$^3$ and about $1000^\circ$C, respectively, the helium release is enhanced due to remarkable microcracking of the pellet, which causes the swelling rate decrease.

1. Introduction

Boron carbide is extensively used as a control rod material for fast reactors, because of its superior properties such as great neutron absorption capacity, high melting temperature, light weight and chemical stability at elevated temperatures. However, the helium produced by $^{10}$B(n,$\alpha$)$^7$Li reaction is released from the pellet, and causes the increase of the gas pressure inside absorber pins. The retained helium in the pellet causes the pellet swelling, and causes the absorber-cladding mechanical interaction (ACMI). Since the life time of control rods are mainly dominated by helium release and pellet swelling, it is important to evaluate the neutron irradiation behavior of boron carbide up to high burnup.

On the last specialists meeting in 1983[1], the irradiation behavior of boron carbide pellets in "JOYO" MK-I core was presented, in which burnup level was limited to only about $30 \times 10^7$ cap/m$^3$. The irradiation behavior of the pellets up to about $80 \times 10^7$ cap/m$^3$ burnup were also reported from each country. Since that meeting, results of
post-irradiation examination have been accumulated in Europe up to about $150 \times 10^{26}$ cap/m$^3$.

In PNC, the experimental fast reactor "JOYO" was improved to MK-II core in 1983, and irradiation test of boron carbide pellets has been executed using Absorber Materials Irradiation Rig (AMIR) for development of long life control rods. Recently, the post-irradiation data were obtained up to burnup level of $230 \times 10^{26}$ cap/m$^3$ at maximum irradiation temperature of 1400°C. This paper describes the results of post-irradiation examination of boron carbide pellets irradiated at such high burnup and high temperature region.

2. Experimental procedure

2.1 Specimens and irradiation condition

Chemical compositions of boron carbide pellets are shown in Table I. These pellets were formed by hot pressing, and boron to carbon ratios were set in $4.0 \pm 0.2$. Specifications of boron carbide pellets are shown in Table II. $^{10}$B enrichments are varied at three levels between 37 and 92at% and pellet densities are varied 90 and 95% T.D. The average grain sizes of all pellets are under 5μm.

<table>
<thead>
<tr>
<th>Table I</th>
<th>Chemical Composition of Boron Carbide Pellet</th>
</tr>
</thead>
<tbody>
<tr>
<td>Item</td>
<td>Specification</td>
</tr>
<tr>
<td>37at%</td>
<td>50at%</td>
</tr>
<tr>
<td>Total B</td>
<td>$\leq 75$</td>
</tr>
<tr>
<td>Total C</td>
<td>$\leq 23$</td>
</tr>
<tr>
<td>B/C Ratio</td>
<td>4.0±0.2</td>
</tr>
<tr>
<td>Soluble B</td>
<td>$\leq 0.2$</td>
</tr>
<tr>
<td>Soluble C</td>
<td>$\leq 1.51$</td>
</tr>
<tr>
<td>Fe</td>
<td>$\leq 0.8$</td>
</tr>
<tr>
<td>Ti</td>
<td>$\leq 0.1$</td>
</tr>
<tr>
<td>Si</td>
<td>$\leq 0.15$</td>
</tr>
<tr>
<td>Al</td>
<td>$\leq 0.20$</td>
</tr>
<tr>
<td>Cl+F</td>
<td>$\leq 0.01$</td>
</tr>
<tr>
<td>Co</td>
<td>$\leq 0.005$</td>
</tr>
<tr>
<td>Cu</td>
<td>$\leq 0.01$</td>
</tr>
<tr>
<td>Mn</td>
<td>$\leq 0.01$</td>
</tr>
<tr>
<td>Na</td>
<td>$\leq 0.01$</td>
</tr>
</tbody>
</table>

These pellets were inserted in stainless steel capsules, and irradiated as AMIR test and "JOYO" MK-I and MK-II driver control rods. Irradiation conditions of boron carbide pellets are also shown in Table II. The maximum burnup reaches $227 \times 10^{26}$ cap/m$^3$ and the maximum irradiation temperature at pellet center is 1380°C. Where, the burnups and irradiation temperatures were estimated using MAGI code and HEATING-5 code, respectively.
2.2 Experimental method

After completion of irradiation, puncturing tests of the capsules were conducted to estimate the helium quantities released from boron carbide pellets. Then visual inspection, density measurement and thermal conductivity measurement were performed on the boron carbide pellets.

Table II: Specification and Irradiation Condition of Boron Carbide Pellet

<table>
<thead>
<tr>
<th></th>
<th>$^{10}$B Enrichment (at%)</th>
<th>Pellet Density (% T.D.)</th>
<th>Average Grain Size ($\mu$m)</th>
<th>Burnup ($10^6$cap/m$^2$)</th>
<th>*1 Irradiation Temperature (°C)</th>
</tr>
</thead>
<tbody>
<tr>
<td>AMIR-1</td>
<td>37</td>
<td>95</td>
<td>&lt;5</td>
<td>33-51</td>
<td>580-920</td>
</tr>
<tr>
<td>AMIR-2</td>
<td>37</td>
<td>95</td>
<td>&lt;5</td>
<td>92-132</td>
<td>590-880</td>
</tr>
<tr>
<td>AMIR-3-1</td>
<td>92</td>
<td>95</td>
<td>&lt;5</td>
<td>115-159</td>
<td>1020-1330</td>
</tr>
<tr>
<td>AMIR-3-2</td>
<td>92</td>
<td>95</td>
<td>&lt;5</td>
<td>168-227</td>
<td>1180-1310</td>
</tr>
<tr>
<td>AMIR-4-1</td>
<td>50, 92</td>
<td>95</td>
<td>&lt;5</td>
<td>5-100</td>
<td>500-1380</td>
</tr>
<tr>
<td>MK-I</td>
<td>92</td>
<td>95</td>
<td>&lt;5</td>
<td>-50</td>
<td>-610</td>
</tr>
<tr>
<td>MK-II</td>
<td>92</td>
<td>90</td>
<td>&lt;5</td>
<td>-80</td>
<td>-1330</td>
</tr>
</tbody>
</table>

*1 Calculated by MAGI code
*2 Calculated by HATING-5 code (at Pellet Center)

(1) Helium release

Helium quantities released from boron carbide pellets were estimated by measurement of helium gas pressure inside the capsules through the puncturing tests of capsules. The released helium quantities are normalized in the helium volume at standard pressure and temperature for unit volume of boron carbide ($m^3$-STP/m$^3$-B$_4$C).

(2) Pellet swelling

To estimate the swelling of boron carbide pellets, density measurements were conducted by means of picnometric method using mercury. The unit of the swelling was converted from density change to diametral change ($\Delta D/D\%$).

(3) Thermal conductivity

The pellets were machined into a plates with 3mm in thickness. Then, the thermal diffusivity was measured using the laser flash method. The thermal conductivity $K$ was evaluated using the following equation with the heat capacity of unirradiated pellet.

$$K = \rho \cdot C_p \cdot \alpha$$

where, $\rho$ is pellet density, $C_p$ is heat capacity and $\alpha$ is measured thermal diffusivity.
3. Results and discussion

(1) Helium release

Results of measurement of released helium quantities from the boron carbide pellets are shown in Fig.1. Released helium is much lower than the helium generation at the burnup level below $100 \times 10^{26} \text{cap/m}^3$. In further irradiation, the helium release is accelerated, and release rate becomes almost equivalent to the generation rate under the condition of over $150 \times 10^{26} \text{cap/m}^3$ and over 1000°C. This result is almost the same as the estimated results in USA\textsuperscript{131}. However, it is reported by Europe\textsuperscript{121} that the helium release rate becomes almost equivalent to the generation rate at $50 \times 10^{26} \text{cap/m}^3$, which is lower burnup level than our result.

Irradiation temperature dependence of fractional helium release is shown in Fig.2, which makes clear that helium release increases linearly with temperature within burnup range of $150 \times 10^{26} \text{cap/m}^3$. When burnup exceeds $150 \times 10^{26} \text{cap/m}^3$, influence of burnup tend to be remarkable rather than temperature dependence. Figure3 shows the apparent view of irradiated boron carbide pellets. Boron carbide pellets are broken into small pieces by neutron irradiation. The higher the temperature and the burnup, the smaller the broken pieces became. As shown in Fig.4, there are many microcracks inside the grain, some of them reached to the...
Fig. 2 Irradiation Temperature Dependence of Fractional Helium Release

(a) $3.8 \times 10^{24} \text{cap/m}^3$, 760°C
(b) $9.8 \times 10^{24} \text{cap/m}^3$, 1110°C
(c) $1.89 \times 10^{25} \text{cap/m}^3$, 1290°C

Fig. 3 Photographs of Irradiated Boron Carbide Pellets
grain boundary. The grain boundary of boron carbide irradiated up to high burnup appears to be brittle, as shown in right hand side of Fig.4. From the results of examination described above, dominant process of helium release seems to be helium diffusion in crystal grain\(^{145}\). At higher burnup region, helium release is accelerated by increasing of helium release paths due to occurrence of microcracking in helium accumulated pellets.

Pellet density, grain size and \(^{10}\)B enrichment are also the fabrication parameters with an effect on helium release. However, from the microstructure as shown in Fig.3 and 4, the effects of those parameters are considered to be negligible at high burnup region where the pellet cracking is a dominant factor for helium release.

(2) Pellet swelling

Figure 5 shows the burnup dependence of diametral swelling of boron carbide pellets. Swelling of boron carbide pellets tends to increase linearly with burnup at the swelling rate of about 3Δ D/D%/100×10^4 cap/m^3 at low burnup. When burnup exceeds 100×10^4 cap/m^3, the swelling rate tends to decrease gradually. This behavior corresponds to the helium release in Fig.1, and it can be considered that increasing of released helium quantities contributes to decreasing of swelling rate. The swelling rate reported in each country\(^{121}\) is about 5Δ D/D%
/100×10^{26} \text{cap/m}^3, which is the greater value than our results. The decreasing behavior of swelling rate beyond 100×10^{26} \text{cap/m}^3 has also never reported even in the study on the irradiated pellets reached about 150×10^{26} \text{cap/m}^3 in Europe.

Relationship between swelling and helium retention in the boron carbide pellets was estimated, because the primary factor of swelling is considered to be helium accumulation in the pellets^{11}. As shown in Fig.6, helium retention increases linearly with burnup up to about 100×10^{26} \text{cap/m}^3, and beyond that burnup helium retention tends to saturate. Figure7 shows the relationship between helium retention and swelling. Although the data are relatively scattered, swelling and helium retention has proportional relation. This result supports that the main factor of swelling is accumulation of helium produced by \(^{10}\text{B}(n, \alpha)^{7}\text{Li}\) reaction in the boron carbide pellets.

According to the report by Zuppiroli et al^{6}, temperature does not influence on swelling below 1500°C. However, since the swelling has the proportional correlation to helium retention which depends on temperature, it is considered that the swelling has a temperature dependence even below 1500°C.

(3) Thermal conductivity

Figure8 shows temperature dependence of thermal conductivity of irradiated boron carbide. The thermal conductivity of irradiated
Fig. 6 Burnup Dependence of Helium Retention

Fig. 7 Relationship between Helium Retention and Pellet Swelling
specimens has not depended on temperature, while thermal conductivity of unirradiated specimen tends to decrease with temperature, that is a typical phenomenon of phonon conduction. These results agree with those reported by Gilchrist\textsuperscript{17} and Mahagin et al\textsuperscript{18}.

The phonon mean free path (mfp) $l$ of irradiated materials can be expressed as

$$\frac{1}{l} = \frac{1}{l_0} + \frac{1}{l_d}$$

where, $l_0$ is the mfp in unirradiated materials and $l_d$ the mfp associated with irradiation induced defects. If the mfp $l_d$ is very small compared with $l_0$, $l$ is effectively equal to $l_d$ and the thermal conductivity is determined by the irradiation induced defects alone. Because $l$ is very small compared with $l_0$, it is considered that thermal conductivity of irradiated boron carbide has less temperature dependence such as unirradiated specimens.

![Fig.8 Temperature Dependence of Thermal Conductivity](image)

Burnup dependence of thermal conductivity of boron carbide at room temperature is shown in Fig.9. Thermal conductivity of unirradiated boron carbide corresponds to about $30 \text{ Wm}^{-1}\text{K}^{-1}$. Thermal conductivity decreases rapidly by irradiation to one third of unirradiated value till about $5 \times 10^{24} \text{ cap/m}^3$ of burnup. Then it decreases gradually with
burnup and reaches 10% of unirradiated value at about $50 \times 10^{26}$cap/m$^3$. The tendency that thermal conductivity decrease still remains at about $50 \times 10^{26}$cap/m$^3$.

4. Conclusion

Irradiation effects on the properties of boron carbide pellets were evaluated with the post-irradiation data, which reaches burnup of $230 \times 10^{26}$cap/m$^3$ at maximum temperature of 1400°C. The results in this study are summarized as follows.

(1) Boron carbide pellets are cracked in pieces with neutron irradiation. Particularly, boron carbide pellets are cracked remarkably over the burnup of $100 \times 10^{26}$cap/m$^3$ and over about 1000°C.

(2) Helium release shows rapid increase with burnup over about $100 \times 10^{26}$cap/m$^3$, and is enhanced by the occurrence of pellet cracking.

(3) The swelling rate of boron carbide pellets takes about 3% per 100 $\times 10^{26}$cap/m$^3$ at burnup of below $100 \times 10^{26}$cap/m$^3$ and below 1000°C. The swelling rate decreases at burnup of beyond $100 \times 10^{26}$cap/m$^3$, which might be due to the accelerated helium release by microcracking.

(4) Thermal conductivity of boron carbide pellets shows great degradation with neutron irradiation. The thermal conductivity of irradiated boron carbide pellets has little temperature dependence.
It is concluded that the properties of boron carbide pellets irradiated at high burnup level are very different from that of low burnup level. If the burnup and the irradiation temperature are greater than about $100 \times 10^2 \text{cap/m}^3$ and about 1000°C, respectively, the helium release is enhanced due to remarkable pellet cracking, and the swelling rate decreases.

These data on high burnup boron carbide pellets such as helium release and pellet swelling are very valuable to develop the long life control rods.

ACKNOWLEDGMENT

The authors express sincere appreciation to the members of Fuels Monitoring Section (FMS) in PNC who executed the puncturing tests to estimate the helium release behavior of boron carbide pellets.

REFERENCES

THIRD SHUTDOWN LEVEL FOR EFR PROJECT

D. FAVET, B. CARLUEC
FRAMATOME Direction NOVATOME
Lyon

S. DECHELETTE
Electricité de France,
Cedex
France

Abstract

The presentation described and evaluated a third shutdown level of the EFR project, which has been developed implementing the following four new device systems:

• a system which terminates the power to the absorber electromagnets after a loss of primary pump electric power supply;
• a device for control rods drive lines, which passively enhances thermal expansion;
• a device which overcomes control rod jamming, by motorized insertion of absorber rods, and
• a mechanical stroke-limitation device, which passively terminates the withdrawal of the faulted control rod.

The goals of the presentation are the following:

. to present the EFR safety approach which leads to improve the preventative measures against the occurrence of whole core accident, and the basis for judging the incentives of implementation of additional shutdown devices (so-called third shutdown level),
. to present the chosen devices and their preliminary design,
. to present the efficiency of the third shutdown level in case of transients not protected by the basic shutdown systems.

Two independent shutdown systems are provided in EFR. The principles and the design options concerning the basic shutdown systems to achieve the reliability requirement are presented.

The risk minimisation approach for EFR is to strive for a comprehensive and balanced safety concept by harnessing favourable features of liquid metal fast breeder reactors. For the shutdown function a so-called third shutdown level has been introduced as an extra safeguard. This basically consists of additional engineering features which include passive and active measures capable of bringing the reactor to a safe condition in case of postulated total failure of the two basic shutdown systems after design basis transients.

The devices are designed in order to achieve the two principle third shutdown functions:

. to disengage the absorber rods so that they may fall into the core,
. to mechanically assist the insertion of the absorber rods.
The main third shutdown devices which will be described in the presentation are the following:

- SADE system which passively terminates the power supply to the absorber electromagnets after a loss of primary pump electrical power supply,
- delatching by control rod enhanced expansion device (CREED) which provides an increased thermal expansion on the rod driveline. At a certain threshold of expansion a delatching mechanism initiates passively rod release,
- bulk rod insertion (BRI) which provides shutdown by motorised insertion of absorber rods. In case of jamming of the rods, BRI provides drive-in forces which are much higher than gravity and only limited by the strength of the drive mechanism,
- rod disconnection initiated by the mechanical stroke limitation device which passively terminates the withdrawal of the faulted rods.

The third shutdown level is assessed in regard to postulated transients. For each transient, it will be verified:

- that the third shutdown level provides an additional, efficient and reliable line of defence,
- the fulfilment of the success criteria by the third shutdown level.

1. **INTRODUCTION**

The overall safety objective for EFR is to reduce the risk of operation to a level as low as reasonably practical (ALARP principle) and no higher than that expected for future LWR’S in the European Countries concerned.

Apart from exotic initiators (e.g. fast structural collapse or insertion of a large amount of gas into the core) a core melt accident in a LMFBR with state of the art safety features and engineering could only occur if either the shutdown systems or the decay heat removal systems fail totally. In general, postulated failure of these systems in past LMFBR’S resulted into core melt scenarios. Therefore, an improvement of these safety functions - i.e. shutdown and decay heat removal - would result in a genuine enhancement of preventive safety. The improvement of the shutdown function of EFR (reference 1) is presented in the following sections.

2. **STRATEGY FOR RISK MINIMIZATION FOR EFR**

For the shutdown function a so-called third shutdown level has been introduced in EFR as an extra safeguard. This basically consists of additional engineering features which include passive and active measures capable of bringing the reactor to a safe condition in case of postulated total failure of the two basis shutdown systems after design basis transients.

In order to ensure a robust and well balanced preventative safety concept, an extended line of defence concept (reference 2) is applied. Basically, two strong and one medium lines of defence are required in order to relegate the intolerable natural consequences of an accident sequence into the residual risk. Two strong lines of defence are in general contributed by the reactor protection systems. Other such features as plant protection system, natural behaviour and operator action may represent medium lines of defence. The low frequency of initiating events may also be equivalent to a line of defence.

In the frame of the EFR risk minimization effort, this basis target has been extended: at least, one additional line of defence should be introduced to interrupt event sequences which would have the potential to cause serious core damage. The extra safeguards against failure
of shutdown have been introduced for the more frequent initiating faults. If a low frequency fault is protected by the two diverse shutdown systems, the overall protection against a whole core accident is already better than basically required.

Exotic initiators (e.g. fast structural collapse or insertion of a large amount of gas into the core) are already excluded by appropriate design measures. In these cases, a deterministic approach resulting from robust design features and attenuating natural behaviour justifies to relegate these events into the residual risk.

Table I presents the lines of defence basically required to relegate whole core melt scenarios into the residual risk for the typical events. The additional lines of defence for further risk minimization are described in the section 3 and listed in Table II.

**TABLE I** BASIC LINES OF DEFENCE FOR REPRESENTATIVE INITIATORS

<table>
<thead>
<tr>
<th>INITIATOR</th>
<th>Required (2a + b) LOD'S</th>
</tr>
</thead>
<tbody>
<tr>
<td>Loss of station service power (LOSSP)</td>
<td>RPS : 2a</td>
</tr>
<tr>
<td>Coast down of all primary pumps (LOF) by other cause than LOSSP</td>
<td>PPS b</td>
</tr>
<tr>
<td>Break of one LIPOSO (Q ~ 42 %)</td>
<td>Prevention a</td>
</tr>
<tr>
<td>Simultaneous break of two LIPOSO's</td>
<td>Prevention 2a</td>
</tr>
<tr>
<td>i) at the same pump (Q ~ 26 %)</td>
<td>Prevention 2a + b</td>
</tr>
<tr>
<td>ii) at different pumps (Q ~ 24 %)</td>
<td></td>
</tr>
<tr>
<td>Loss of secondary pumps (LOHS)</td>
<td>PPS b</td>
</tr>
<tr>
<td>Loss of feedwater or other malfunction of the W/S system in all loops (LOHS)</td>
<td></td>
</tr>
<tr>
<td>Withdrawal of a single rod</td>
<td>Rod runaway protection b</td>
</tr>
<tr>
<td>Withdrawal of all rods</td>
<td>Global power limitation b</td>
</tr>
<tr>
<td>Total instantaneous blockage</td>
<td>Prevention 2a</td>
</tr>
<tr>
<td>Core compaction by SSE</td>
<td>Prevention 2a</td>
</tr>
<tr>
<td>Core support failure</td>
<td>Protection equivalent to (2a + b) LOD'S by design standards, minimal risk of buckling, in-service inspection, leak before break argument, self redundant structure (e.g. secondary skirt), slow development detectable by RPS</td>
</tr>
<tr>
<td>Voiding by fission gas release (severe earthquake or BDB earthquake, severe core transients (ULOF, ULOHS, UTOP) with prolonged high temperature condition)</td>
<td>Prevention of avoidance of an initiator for failure of a large number of simultaneous clad failures 2a + b</td>
</tr>
</tbody>
</table>

RPS : Reactor Protection System  
PPS : Plant Protection System  
BDB : Beyond Design Basis (e.g. Limiting Event)  
BDB/RR : Beyond Design Basis / Residual Risk  
SSE : Safe Shutdown Earthquake  
ULOF : Unprotected Loss of Flow  
ULOHS : Unprotected Loss of Heat Sink  
UTOP : Unprotected Transient Over Power  
LIPOSO : piping connecting the primary pump at the diagrid  
a LOD : strong Line Of Defence  
b LOD : medium Line Of Defence
<table>
<thead>
<tr>
<th>Initiator</th>
<th>Additional LOD'S for Risk Minimisation</th>
</tr>
</thead>
<tbody>
<tr>
<td>ULOSSP</td>
<td>SADE, BRI, CREED</td>
</tr>
<tr>
<td>ULOF other than ULOSSP</td>
<td>CREED, BRI</td>
</tr>
<tr>
<td>ULOHS</td>
<td>CREED, BRI</td>
</tr>
<tr>
<td>Withdrawal of a single rod</td>
<td>SLD with rod release, DND</td>
</tr>
<tr>
<td>Withdrawal of all rods</td>
<td>SLD with rod release</td>
</tr>
<tr>
<td></td>
<td>SLD + CREED for part load</td>
</tr>
<tr>
<td>Subassembly accident propagation</td>
<td>DND or ABND depending on the initiator and the sequence of events</td>
</tr>
<tr>
<td>Passage of gas through the core</td>
<td>Reactimeter</td>
</tr>
<tr>
<td>Radial core movement</td>
<td></td>
</tr>
<tr>
<td>Core support failure</td>
<td>Reactimeter, disconnection of rods after limited core drop</td>
</tr>
</tbody>
</table>

ULOSSP : Unprotected Loss of Station Service Power

ULOFS : Unprotected Loss of Flow

ULOHS : Unprotected Loss of Heat Sink

BRI : Bulk Rod Insertion

SLD : Stroke Limitation Device

DND : Delayed Neutron Detection

ABND : Acoustic Boiling Noise Detection

The fulfilment of the success criteria by the third shutdown level is verified for postulated failures of the main shutdown systems (total failure of the reactor trip systems or of the absorber systems) after serious imbalances of produced and removed power (failure of the pumps, faulty withdrawal of absorbers, loss of main heat sink) which could lead to core disruptive accidents (ULOFS, UTOP, ULOHS), results of transient analysis are given in section 4.
3. **DESCRIPTION OF DEVICES AND FEATURES FOR IMPROVEMENT OF THE SHUTDOWN FUNCTION**

The third shutdown level consists of passive and active measures being capable of bringing the reactor to a safe condition in case of a postulated failure of the two basic shutdown systems. Two different types of failure of the basic shutdown function are considered:

- failure to de-energise the scram magnets,
- failure of rods to drop into the core.

Failure to de-energise all electromagnets is minimised by redundancy and diversity of the trip systems. Because of the diversity of the electromagnet locations (in the cover gas and under sodium), only rod jamming in the core remains as a cause of mechanical rod failure. Regarding the high degree of diversity of the rods and the extreme misalignment and tube deflections which can be tolerated, this type of failure is judged to be extremely unfrequent.

Regarding these types of failures, the two principle functions of the third shutdown level are:

- to disengage the absorber rods so that they may fall into the core,
- to mechanically assist the insertion of the absorber rods.

The main features of the third shutdown level are presented in the following

### 3.1 SADE System

The SADE system would passively terminate the power supply of one type of absorber rod electromagnet after a loss of primary pump electrical power supply, if the trip signals had failed to initiate rod drop (see figure 1).

These electromagnets are electrically fed by a generator which is driven by a motor provided with a flywheel. The design of the flywheel is such that the gravity drop of the rods occurs in less than about 10 seconds. This value is consistent with the halving time of the primary pump coast down and is small enough to avoid sodium boiling in case of loss of station service power combined with a failure of both shutdown systems.

### 3.2 Delatching by Control Rod Enhanced Expansion Device (CREED)

CREED is a passive mechanism for the delatching of rods into the core in response to core outlet temperature increase (Figure2). It essentially consists of an element which provides an increased thermal expansion on the rod driveline. At a certain threshold of expansion, a delatching mechanism passively initiates rod release.

CREED characteristics are:

- the threshold for delatching of about 500°C - 600°C
- a time delay of the enhanced expansion related to the core outlet temperature of about 14 s at nominal flow.
FIGURE 1 PRINCIPLE OF SADE
FIGURE 2  CREED - DSD ROD HOT MAGNET PUSH OFF DEVICE
3.3 Rod Disconnection Initiated by the Mechanical Stroke Limitation Device

The primary purpose of the Stroke Limitation Device (SLD) (Figure 3) is to terminate the withdrawal of the faulted rods in case of inadvertent rod withdrawal and so reduce the risk of unacceptable fuel melting to an acceptably low level. Conditions in the affected fuel pins may further deteriorate after termination of the rod withdrawal. SLD has, therefore, the additional function of rod release should the stroke limit be reached.

3.4 Bulk Rod Insertion (BRI)

In order to prevent rod jamming, the shutdown function by gravitational drop of absorbers is complemented by motorised insertion of absorber rods initiated by the trip signals.

In case of jamming of the rods, BRI provides drive-in forces that are much higher than gravity and only limited by the strength of the drive mechanisms.

4. ASSESSMENT OF THE RISK MINIMIZATION BY ADDITIONAL PREVENTION

4.1 Success criteria

The following success criteria are used to assess the efficiency of the available elements and features:

A/ Short term transient phase

. Avoidance of widespread sodium boiling,
. limiting the fuel melt fraction in overpower transients,
. core support structure integrity must not be jeopardized,
. maintenance of coolable global geometry.

B/ Long term

. Safe neutronic shutdown at an asymptotic equilibrium temperature of 500°C,
. Decay heat removal systems must reach and maintain the safe neutronic shutdown equilibrium temperature.

4.2 Efficiency of the third shutdown level

For three typical faulty transients - Unprotected Loss of Flow (ULOF), Unprotected Transient Overpower (UTOP) and Unprotected Loss of Heat Sink (ULOHS), the effects of CREED, SLD and BRI have been parametrically studied. Both CREED effects enhanced thermal expansion and delatching, have been separately assessed.

4.2.1 Unprotected Loss of Flow (ULOF)

The natural behaviour for an untripped loss of flow rapidly leads to a large increase of the coolant temperature in the short term, whereas the reactor power decreases due to the
The sketch is archived on the VAX under the ID [PES SKETCH]Dwg.045 SKD

FIGURE 3  SLD - MOVABLE STOP TYPE SLD DESIGN
reactivity feedback effects. If sodium sink boiling is avoided, the further development of the accident is linked to the decay heat removal performance, which depends on the accident scenario. Computer simulation have shown that the ULOF initiated by a LOSSP leads to the highest core temperatures.

In this case, the transient analysis performed for the EFR core - including a control rods drivelines enhanced thermal expansion effect of about 0.4 mm/°C with a specific flow dependent time constant $t = 11 + 2 / (\text{primary flow rate})$ seconds - revealed that sodium boiling can be avoided - Table III and Figure 4. These results are based on the assumption that there is no delatching, and all the rods are equipped with CREED.

Improvement of risk minimizations is obtained if the absorber rods are delatched by CREED.

The long-term effect of both CREED functions enhanced thermal expansion and delatching on the asymptotic equilibrium temperature (AET) can be seen in Table IV. The improvement of these passive shutdown features is sufficient to keep the AET below 700°C and to avoid structural failures for about ten days at least, which provide grace time for operator action, manual shutdown for instance.

Assuming that the control rod driveline enhanced thermal expansion is weak and CREED delatching is not efficient, it was derived that boiling for ULOF can be avoided by the motor driven insertion (BRI) of all CSD rods at 1 mm/s and of all diverse and shutdown rods (DSD) at 8 mm/s.

<table>
<thead>
<tr>
<th>Case</th>
<th>Pump halving time (s)</th>
<th>Minimum % flow</th>
<th>Minimum rod insertion %</th>
<th>Maximum sodium temperature °C</th>
<th>Time of maximum temperature (s)</th>
<th>Margin to boil</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>10</td>
<td>0</td>
<td>0</td>
<td>boiling</td>
<td>150</td>
<td>-</td>
</tr>
<tr>
<td>2</td>
<td>10</td>
<td>0</td>
<td>0</td>
<td>923</td>
<td>88</td>
<td>29</td>
</tr>
<tr>
<td>3</td>
<td>10</td>
<td>0</td>
<td>5</td>
<td>886</td>
<td>88</td>
<td>67</td>
</tr>
<tr>
<td>4</td>
<td>10</td>
<td>25</td>
<td>0</td>
<td>873</td>
<td>30</td>
<td>85</td>
</tr>
<tr>
<td>5</td>
<td>10</td>
<td>10</td>
<td>0</td>
<td>918</td>
<td>76</td>
<td>36</td>
</tr>
<tr>
<td>6</td>
<td>20</td>
<td>0</td>
<td>0</td>
<td>884</td>
<td>230</td>
<td>69</td>
</tr>
</tbody>
</table>

4.2.2 Unprotected Transient Overpower (UTOP)

Only faults, leading to an overpower, that can be controlled by the reactor shutdown systems are considered and they are called slow transient overpower.

Such faults are caused by the unscheduled withdrawal of one or several control rods, occurring at full power or part loads. They provoke slow increases in power either widespread in the core or restrictively located in the vicinity of a faulted control rod.
FIGURE 4 MAXIMUM SODIUM TEMPERATURE DURING LOSS OF PRIMARY FLOW TRANSIENT FOR VARIOUS LEVELS OF REACTOR SHUTDOWN
TABLE IV  EFFECT OF CREED ON ASYMPTOTIC EQUILIBRIUM

<table>
<thead>
<tr>
<th>ASSUMPTIONS (T_{FUEL} = T_{COOLANT})</th>
<th>AET [°C]</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>BOEC</td>
</tr>
<tr>
<td>○ No CREED, no jamming of rods</td>
<td>930</td>
</tr>
<tr>
<td>○ CREED with enhanced expansion Δρ / ΔT above 600°C :</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
<tr>
<td>≡ no delatching, rods free movable</td>
<td></td>
</tr>
<tr>
<td>a) Δρ / ΔT = 0.4 mm/°C</td>
<td>674</td>
</tr>
<tr>
<td>b) Δρ / ΔT = 1.0 mm/°C</td>
<td>633</td>
</tr>
<tr>
<td>≡ insertion of Δρ by delatching and failing of only a few CSD rods, other rods not delatched but free movable</td>
<td>616</td>
</tr>
<tr>
<td>Δρ = -1 $</td>
<td>547</td>
</tr>
<tr>
<td>Δρ = -1.5 $</td>
<td>437</td>
</tr>
<tr>
<td>Δρ = -2 $</td>
<td></td>
</tr>
<tr>
<td>≡ delatching of all CSD rods at T = 625°C during the transient, insertion of Δρ by thermal expansion and by falling of a few rods</td>
<td>500</td>
</tr>
<tr>
<td>Δρ = -1$</td>
<td>450</td>
</tr>
<tr>
<td>Δρ = -1.25 $</td>
<td></td>
</tr>
<tr>
<td>≡ delatching and falling of all CSD rods</td>
<td>Cold shutdown</td>
</tr>
</tbody>
</table>

BOEC : Beginning of equilibrium cycle  
EOEC : End of equilibrium cycle

The complete withdrawal of one single rod from half inserted initial position (start of an equilibrium cycle) with automatic core global power compensation would cause a 25 % local overpower for an inner ring control and shutdown rod (CSD) and 60 % to 70 % for an outer ring CSD - best estimate analyses because UTOP is beyond the design basis - showed that no fuel melting would take place in case of the inner ring CSD complete withdrawal whereas it certainly would do in case of the outer ring. As it is difficult to demonstrate that in case of fuel melting clad failure allowing some interaction between the coolant and molten fuel will not occur, it is conceivable that the fault could propagate into a core melting.
The simultaneous withdrawal of all CSD rods from full power operating conditions at the beginning of cycle with a speed of 1 mm/s would cause a 3 \( \epsilon/s \) reactivity ramp. The complete withdrawal would insert 8-9 \( \epsilon \) (best estimate). The overall reactor power would double with a 80 \( \epsilon \) insertion, which corresponds to the total reactivity insertion if the event occurs at the end of cycle.

It is clear from these examples that a UTOP caused by the simultaneous withdrawal of all CSD could initiate an intolerable CDA (Core Disruptive Accident).

In case of failure of the design basis lines of defence, the third shutdown level protection is provided by independent systems.

The stroke limitation device (SLD) is specified to meet the demands of an unprotected withdrawal of all CSD rods. It is designed to stop the rods motion after 15 mm, which allows a reactivity insertion of 45 \( \epsilon \) and an overpower peak of 40 - 50\%. Figure 5 shows that peak fuel temperature are near the melting point at this stage of the transient history.

In case of the withdrawal of one single rod, it would limit the overpower to only 2-3\%.

Bulk rod insertion (BRI) is unlikely to be effective if the absorber rods system has failed to respond to previous commands from the reactor control and protection systems to terminate the withdrawal.

CREED, with same characteristics as in the ULOF analysis, was not introduced to cause rod insertion for rod withdrawal faults because it would not act before the reactivity insertion reaches about 1 \( \epsilon \), the overpower peak being then 250\%. Although CREED would prevent coolant boiling quite extensive fuel melting would occur.

![FIGURE 5 UTOP - ALL CSD RODS WITHDRAWN](image)
4.2.3 Unprotected Loss of Heat Sink

The loss of the main heat sink (LOHS) may be initiated by:

- secondary pump coast down or seizure
- loss of feed water.

LOHS can also be initiated by LOSSP but in this case transient behaviour is ruled by the loss of primary flow (see section 4.2.1 ULOF).

Assuming that the reactor trip is not initiated or fails, the thermal radial core expansion provides a large negative reactivity feedback effect, which leads to a passive power reduction. In the long term, all temperatures in the hot and cold pools slowly increase according to the large thermal capacity of the primary system (see Figure 6). Equilibrium is reached if the direct reactor cooling system is available for the removal of decay power.

BRI and CREED are independent and very effective measures of the third shutdown level.

If trip signals or rods magnets de-energization fail, the bulk insertion of CSD and DSD rods enable exerting insertion forces to override even limited rod jamming. Assuming that all rods are insertable and with the insertion speed as defined of ULOF (see section 4.2.1) that is for the end of equilibrium cycle, -1,5 to -2 $ is inserted within the first minute after the fault initiation a core disruption is definitely excluded for ULOHS. ULOHS is indeed much more benign and develops more slowly than ULOF.

Assuming that the gravity fall of absorber rods fails but excluding irrevocable jamming of all CSD and DSD rods, the asymptotic equilibrium temperature (AET) is about 500°C for BRI insertion of -1 $, which is achievable with the insertion of 2 or 3 rods, if no driveline expansion effect from the other rods is assumed.

The beneficial effects of CREED are illustrated in table 4. No BRI is considered here. The AET is smaller than 700°C. AET will be smaller than 500°C if more than about -1,3 $ for end of equilibrium cycle and -1,7 $ for beginning of equilibrium cycle are inserted by the delatching of only a few rods.

5. CONCLUSIONS

The implementation of new devices into the EFR project leads to a significant improvement of the shutdown function:

- SADE system is implemented because its simplicity, attractive passive features and low costs and its high reliability in case of LOSSP combined with failure of both trip systems.

- Bulk rod insertion with forced drive-in allows to shut down the reactor even in case of mechanical failure with resistance against insertion of absorber rods by gravity.

- Mechanical stroke limitation device limits the maximum reactivity insertion. The repositionning of the stop device is necessary for compensation of burnup reactivity.
FIGURE 6  EFR - ULOHS WITHOUT CREED
Control rod enhanced expansion device could prevent sodium boiling in case of ULOF and ULOHS, even if the delatching function is not assessed. However, the thermal enhanced expansion function of CREED is not required because delatching function and BRI are efficient for all possible shutdown failures.

REFERENCES


IMPROVEMENT OF THE PERFORMANCES OF THE
"POISONED" CAPRA CORE BY USE OF $^{11}$B$_4$C

G. GASTALDO
FRAMATOME Direction NOVATOME
Lyon

G. VAMBENEPE
Electricité de France SEPTEM,
Cedex

J.C. GARNIER
Commissariat à l'Énergie Atomique,
Durance

France

Abstract

Two approaches have been considered in France by the CAPRA team to optimise a fast reactor for plutonium (or minor actinide) burning: the "dilution" and "poisoning" options /1/. At present, "dilution" is the preferred design as it allows better core safety parameters (mainly the Doppler constant).

This paper shows a route, based on the use of a moderator material, $^{11}$B$_4$C, to improve the core safety parameters (Doppler constant and sodium void reactivity worth) of the CAPRA B$_4$C poisoned core. The aim is to obtain a convincing alternative to the "dilution" option.

The introduction of the moderator inside the fuel bundle of the CAPRA B$_4$C poisoned core allows to strongly increase the Doppler constant (+ 62 %) and to reduce the sodium void reactivity worth (- 23 %).

The "poisoned" CAPRA core with $^{11}$B$_4$C has similar plutonium burning performances and better core safety parameters than the "dilution" option. Its main advantages are linked to the larger diameter fuel pins: increased fuel lifetime and "standard" fuel bundle (397 pins instead of 469).

THE CAPRA PROJECT

According to the French cycle strategy based on irradiated fuels reprocessing, the CEA launched the CAPRA project to demonstrate the feasibility of FRs optimised to burn plutonium. Started by CEA in 1993, the CAPRA project involves also European R&D organizations (AEA and KfK) and design companies (European Fast Reactor Associates consortium). Other close cooperations on the related R&D programme have also been established with JAPAN (PNC), RUSSIA (Obninsk), SWITZERLAND (PSI) and ITALY (ENEA).

The CAPRA project must be considered in the french context where the foreseen plutonium stock in the years 2020 - 2050 is important. So, a fast reactor, plutonium burner, is a mean allowing to regulate the amount of plutonium produced by LWRs.
BACKGROUND

Unlike natural boron (19.82 % $^{10}$B isotopic abundance) which is a neutron absorber material, $^{11}$B$_4$C is a moderator as $^{11}$B only participates to the neutron slowing down process (scattering without absorption). Moreover, with $^{11}$B$_4$C pins uniformly distributed in the fuel bundle, no significant local spectral effect is produced so that any impact on the power distribution is avoided.

MAIN DATA

The main data of the two Capra cores used for this study are collected hereafter. They are related to the so called "09/93" preliminary core versions.

<table>
<thead>
<tr>
<th></th>
<th>&quot;Dilution&quot;</th>
<th>$^{11}$B$_4$C poisoned</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core thermal power (MWth)</td>
<td>3600</td>
<td>3600</td>
</tr>
<tr>
<td>Maximum burn-up (GWd/t$_{oxide}$)</td>
<td>170</td>
<td>170</td>
</tr>
<tr>
<td>Number of fissile sub-assemblies</td>
<td>367</td>
<td>367</td>
</tr>
<tr>
<td>Number of diluents</td>
<td>51 (steel)</td>
<td>51 (natural $^{11}$B$_4$C)</td>
</tr>
<tr>
<td>Number of absorber rods</td>
<td>33</td>
<td>33</td>
</tr>
<tr>
<td>Control and Shutdown (CSD) rods</td>
<td>24</td>
<td>24</td>
</tr>
<tr>
<td>Diverse Shutdown (DSD) rods</td>
<td>9</td>
<td>9</td>
</tr>
<tr>
<td>Number of pins in the fuel bundle</td>
<td>469</td>
<td>397</td>
</tr>
<tr>
<td>Number of fissile pins</td>
<td>320</td>
<td>320</td>
</tr>
<tr>
<td>Number of inert pins</td>
<td>149</td>
<td>77</td>
</tr>
<tr>
<td>Sub-assembly pitch (mm)</td>
<td>181</td>
<td>181</td>
</tr>
<tr>
<td>Fissile height (cm)</td>
<td>100</td>
<td>100</td>
</tr>
</tbody>
</table>

The core layout is shown on figure 1.

CAPRA "POISONED" CORE WITH MODERATOR

The moderator, $^{11}$B$_4$C /2/, is introduced in the empty inert pins. Pins with moderator are uniformly distributed inside the fuel bundle (Figure 2).

CALCULATION METHOD

The physics characteristics of the different cores were determined from equilibrium cycle diffusion/depletion calculations performed using the ERANOS code /3/, a cylindrical (RZ) core model and twenty-five groups cross sections.

In particular, the sodium void reactivity was calculated with cross sections processed individually for flooded and voided cells. The sodium void reactivity and the Doppler constant were calculated for the End of Equilibrium Cycle (EOEC) configuration using the exact perturbation theory. The aim was to obtain the distribution, as well as the components by reaction type, of the reactivity effects.
Figure 1
CAPRA Core layout
One sixth of bundle

Figure 2
CAPRA heterogeneous fuel bundle with moderator

MAIN RESULTS

The main results are collected here below.

<table>
<thead>
<tr>
<th></th>
<th>CAPRA &quot;Dilution&quot;</th>
<th>CAPRA B₄C poisoned</th>
<th>CAPRA B₄C poisoned with moderator</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel residence time (EFPD)</td>
<td>3x286</td>
<td>4x231</td>
<td>4x231</td>
</tr>
<tr>
<td>Feed enrichment:</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>E1 (Vol. , %)</td>
<td>40.5</td>
<td>40</td>
<td>40</td>
</tr>
<tr>
<td>E2 (Vol. , %)</td>
<td>44.5</td>
<td>40</td>
<td>40</td>
</tr>
<tr>
<td>Burn-up reactivity loss per cycle (10⁻⁵ ΔK/KK')</td>
<td>8730</td>
<td>6265</td>
<td>6410</td>
</tr>
<tr>
<td>Plutonium consumption (kg/TWh)</td>
<td>65</td>
<td>65 (*)</td>
<td>62</td>
</tr>
<tr>
<td>Minor actinide production (kg/TWh)</td>
<td>8.8</td>
<td>8.8</td>
<td>9.5</td>
</tr>
<tr>
<td>Sodium void reactivity at EOEC (10⁻⁵ ΔK/KK')</td>
<td>1512</td>
<td>1520</td>
<td>1170</td>
</tr>
<tr>
<td>Doppler constant at EOEC (10⁻⁵ ΔK/KK')</td>
<td>- 435</td>
<td>- 299</td>
<td>- 483</td>
</tr>
</tbody>
</table>

(*) The amount of natural boron placed in the diluent sub-assemblies, a volume fraction of about 15%, is chosen so that the plutonium consumption rate is equivalent to that of the "dilution" core.
Burn-up reactivity loss

$^{11}$B$_4$C has no significant impact on the fuel enrichment and on the burn-up reactivity loss.

Heavy nuclide mass balances

The effect of the moderator on plutonium consumption rate is small (-5 %). The impact on minor actinide production is more important (+ 8 %), but the effect remains small.

Sodium void reactivity and Doppler constant

$^{11}$B$_4$C allows to reduce the sodium void reactivity effect and to increase the Doppler constant:

<table>
<thead>
<tr>
<th>VARIATIONS WITH RESPECT TO THE &quot;DILUTION&quot; OPTION (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>CAPRA B$_4$C POISONED</td>
</tr>
<tr>
<td>Na total void worth</td>
</tr>
<tr>
<td>Doppler constant</td>
</tr>
</tbody>
</table>

In comparison with the "standard" B$_4$C poisoned core the gains are very important:
- the Doppler constant is increased by 62 %
- the sodium void reactivity worth is reduced by 23 %.

CONCLUSION

The "poisoned" CAPRA core with $^{11}$B$_4$C has similar plutonium burning performances and better core safety parameters than the "dilution" option. Note however that the safety parameters of the "dilution" core could be significantly improved by using $^{11}$B$_4$C. So, the main advantages of the "poisoned" core with moderator are linked to the larger diameter fuel pins: increased fuel lifetime and lower number of pins in the fuel bundle (397 instead of 469). Moreover, the use of $^{11}$B$_4$C as filling material does not introduce supplementary technological problems.

The "poisoned" CAPRA core with $^{11}$B$_4$C must be further investigated by using a three dimensions core model (power distribution, fuel lifetime and control rod efficiencies) and other aspects such as plutonium multi-recycling, minor actinide transmutation or higher burn-up have to be considered. Then, the interest of the "dilution" option remains, but from now it can be stated that the "poisoned" CAPRA core with $^{11}$B$_4$C is an interesting alternative.
REFERENCES

/1/ J. ROUAULT and al.
Physics of Plutonium Burning in Fast Reactors: Impact on Burner Cores Design
Topical Meeting on Advances in Reactor Physics, 11 - 15 April 1994, Knoxville, USA

/2/ G. GASTALDO and al.
EFR - Sodium void reduction - Core with moderator
ARS '94 - International Meeting - Advanced Reactors Safety, 17 - 21 April 1994,
Pittsburgh, Pennsylvania

/3/ J. Y. DORIATH and al.
ERANOS 1 - The Advanced European System of Codes for Reactor Physics Calculations
Joint International Conference on Mathematical Method and Supercomputing in Nuclear
Applications, 19 - 23 April 1993 - Karlsruhe, Germany.
EXPERIENCE IN DEVELOPMENT, OPERATING AND MATERIAL INVESTIGATION OF THE BOR-60 REACTOR CONTROL AND SAFETY RODS

V.B. PONOMARENKO, A.I. EFREMOV, G.I. GADZHIEV
Moscow Factory Polymetals, Moscow

V.D. RISOVANY, A.V. ZAKHAROV, T.M. GUSEVA
Research Institute of Atomic Reactors, Dimitrovgrad

Russian Federation

Abstract

The present paper is a summary of all results obtained over 25 years of control rods operation, and involves the following data: (1) design and characteristics of the different types of control rods tested during the BOR-60 reactor operation; and (2) main results of post-irradiation examination.

1. Introduction

The unique data presented on behaviour of absorbing and design materials and five modifications of reactor control rods under high fluence irradiation was obtained over 25 years of the BOR-60 reactor operation.

Every type of rods was developed in terms of the experimental data obtained under the BR-5, BR-10, BOR-60 reactor irradiation of absorbing material specimens and their subsequent examinations at the RIAR and PhEI hot cells as well as in performance of the whole complex of prior irradiation tests (technology development of materials and absorbing elements (AE) production, compatibility testings of absorbing and structural materials, joint effort of rods with drives, etc).

The present paper is a generalisation of all results obtained over 25 years of control rods operation.

The paper involves the following data:
- design and characteristics of the different type control rods tested during the BOR-60 reactor operation;
- main results of spent rods investigation.

2. The BOR-60 Control and Safety System

The BOR-60 control and safety system involves seven rods: three safety rods (SR), two automatic control rods (CR) and two shim rods (RR-1, RR-2).

The CRs are set up at the boundary of core and radial reflector, (RR-1) - in the fourth row and (RR-2) - in the center of core. During the reactor operation, the SRs are above the core; the CR lower part (l=200-250 mm) is submerged in the core at all times; as fuel burnup, RR-2 is moved upwards from its extreme position and RR-2 is withdrawn from the core while bringing the reactor to power.

During the reactor operation the safety rods undergo a small thermal load and are subjected to a relative light radiation effect. The major requirement is reactivity.
worth in the unit of volume. The requirements for radiation resistance and heat removal are less stringent and attributed only to the lower part of rods located close to the active core.

During the reactor operation the CRs and RRs are in the core and subjected to intensive irradiation. The requirement for effectiveness of those rods per volume unit is suitable for the small-sized reactors where the number of rods and volume are small. This requirement is less stringent for the large-sized reactors. The major feature of these rods should be high radiation resistance.

While bringing the reactor to power, the shim rod (RR-1, RR-2) compensating the temperature change should have the maximum effectiveness and only a part of shim rod should be radiation resistance because the rod is withdrawn from the reactor zone at the first stage of the reactor operation and the small neutron flux. All seven rods operate in the hexagonal channels of 42 mm inner diameter and are cooled with sodium flux at inlet temperature from 340 to 350°C.

3. Rod Design

At first in accordance with the BOR-60 draft the rods of control and safety system were developed on the base of boron carbide with sealed absorbing elements (AE).

The design (Fig.3.1a) involved head, tail-end and 0Cr18Ni10Ti steel carrying tube 39.5x0.8 mm diameter in which seven (SR, RR - 1057A-0) or four (RR - 1065A-0) absorbing elements are located. The absorbing element is 0Cr16Ni15Mo3Nb steel cladding 12x0.4 mm diameter filled with boron carbide or chromium deboride pellets 10.8 mm diameter with 80% 10B enrichment. The absorbing elements are sealed and filled with air or helium in the manufacture and have the gas collector. The reliability of rods was provided for the corresponding matching of the radial gap between the absorber and cladding as well as the formation of necessary plenum for gas collection.

The next stage of control rods development involved the manufacture of vented designs filled with boron carbide pellets. Unlike the above design the gap between cladding and boron: carbide pellets was filled with sodium but the gas collector was not available (1057D-O, Fig.3.1b).

To provide the large operative reactivity margin for increase of microcampaign, the single-absorbing vented design with boron carbide pellets of 80% 10B enrichment was developed (1309, Fig.3.1, c). The rod involved head, tail-end and 0Cr16Ni15Mo3Nb steel carrying tube 36x0.8 mm diameter in which boron carbide pellets 32-33 mm diameter were located.

In the upper end plug a special vented unit was used to allow the gap between boron carbide pellets and cladding to be filled with sodium immediately in the reactor. When using the rod as SR, in the lower part of design the ampoules may be used for irradiation testing of different design materials.

At first time the europium oxide containing rods were widely used in the BOR-60 reactor. Their design is similar to rods of boron carbide with sealed absorbing elements. The europium oxide pellets 11 mm diameter and density no less than 7 g/cm³ are used as an absorber (1065D-O, Fig.3.1,d).

To increase the effectiveness of neutron absorption the "trap" type rod design based on europium oxide and zirconium hydride was made (1847, 2121) (Fig.3.1e). It involves head, tail-end, two carrying tubes 39.5x0.8 mm and 27x0.5 mm diameter filled with annular europium oxide specimens 37.6 and 27.2 mm outer and inner diameter and the moderating element located along a center of rod. The moderating
element (ME) is a cladding 18x0.4 mm diameter in which zirconium hydride blocks 17 mm diameter are located. The claddings are made of 0Cr16Ni15Mo3Nb steel. The inner gaps of the absorbing and moderating elements are filled with helium and sodium, respectively.

FIG. 3.1. Control rods design of reactor BOR-60:
a- SR, RR 1057A, b- SR, RR 1057D, c- SR, RR 1309.
FIG. 3.1. (cont.)
d- CR 1065D, e- CR "trap" 2121.
4. Physical Effectiveness of Control Rods

Data on the physical effectiveness of CSS rods involved boron-containing absorbing materials are presented in Tab.4.1.

Application of the single-absorbing element design in the RR-2 rods (B$_4$C pellets 10.8 mm diameter) increases the effectiveness by 30% compared with the seven-absorbing elements design (B$_4$C 10.8 mm diameter). Among the CRs the design of 80% boron carbide enrichment has the most effectiveness. The CR-trap containing the europium oxide absorber and zirconium hydride moderator has the effectiveness by 30% higher than that of the control rod based on europium oxide.

Measurement of the control rod effectiveness after operation in the BOR-60 reactor showed it was practically unchanged at least up to 4% burnup.

**TABLE 4.1. EFFECTIVENESS OF BORON-CONTAINING CS RODS**

<table>
<thead>
<tr>
<th>SC rod</th>
<th>Radius, cm</th>
<th>Number of AE</th>
<th>Absorber type</th>
<th>Average neutron energy, E, KeV</th>
<th>$^{10}$B actual content, g</th>
<th>Effectiveness, $% \Delta K_K$</th>
<th>Calcul.</th>
<th>Measurement</th>
</tr>
</thead>
<tbody>
<tr>
<td>RR-2</td>
<td>0</td>
<td>7</td>
<td>B$_4$C</td>
<td>480</td>
<td>395</td>
<td>1.82</td>
<td>1.90 ± 0.02</td>
<td></td>
</tr>
<tr>
<td>RR-1</td>
<td>16.2</td>
<td>7</td>
<td>B$_4$C</td>
<td>405</td>
<td>380</td>
<td>0.95</td>
<td>1.06 ± 0.02</td>
<td></td>
</tr>
<tr>
<td>SR-1</td>
<td>13.5</td>
<td>7</td>
<td>B$_4$C</td>
<td>440</td>
<td>393</td>
<td>1.25</td>
<td>1.25 ± 0.02</td>
<td></td>
</tr>
<tr>
<td>SR-2</td>
<td>11.9</td>
<td>7</td>
<td>B$_4$C</td>
<td>452</td>
<td>392</td>
<td>1.41</td>
<td>1.34 ± 0.02</td>
<td></td>
</tr>
<tr>
<td>SR-3</td>
<td>18.0</td>
<td>7</td>
<td>CrB$_2$</td>
<td>384</td>
<td>275</td>
<td>0.65</td>
<td>0.84 ± 0.02</td>
<td></td>
</tr>
<tr>
<td>CR-2</td>
<td>23.4</td>
<td>4</td>
<td>B$_4$C</td>
<td>250</td>
<td>218</td>
<td>0.36</td>
<td>0.36 ± 0.01</td>
<td></td>
</tr>
<tr>
<td>CR-1</td>
<td>25.1</td>
<td>4</td>
<td>CrB$_2$</td>
<td>250</td>
<td>158</td>
<td>0.24</td>
<td>0.27 ± 0.01</td>
<td></td>
</tr>
</tbody>
</table>

**TABLE 4.2. EFFECTIVENESS OF CONTROL RODS OF DIFFERENT DESIGN**

<table>
<thead>
<tr>
<th>CS rod</th>
<th>Absorber</th>
<th>Number of AE in CS rods</th>
<th>Radius of location, cm</th>
<th>Effectiveness, $% \Delta K_K$</th>
</tr>
</thead>
<tbody>
<tr>
<td>RR-2</td>
<td>B$_4$C</td>
<td>7</td>
<td>0</td>
<td>1.70 ± 0.02</td>
</tr>
<tr>
<td>RR-2</td>
<td>B$_4$C</td>
<td>1</td>
<td>0</td>
<td>2.20 ± 0.02</td>
</tr>
<tr>
<td>CR-2</td>
<td>B$_4$C</td>
<td>4</td>
<td>23.4</td>
<td>0.36 ± 0.01</td>
</tr>
<tr>
<td>CR-2</td>
<td>Eu$_2$O$_3$</td>
<td>7</td>
<td>23.4</td>
<td>0.20 ± 0.01</td>
</tr>
<tr>
<td>CR-trap</td>
<td>Eu$_2$O$_3$</td>
<td>1-AE</td>
<td>23.4</td>
<td>0.26 ± 0.01</td>
</tr>
<tr>
<td>CR-1</td>
<td>CrB$_2$+(Ta,W)</td>
<td>4</td>
<td>25.1</td>
<td>0.27 ± 0.01</td>
</tr>
</tbody>
</table>
5. The Main Results of Spent Control Rods Investigation

Since the BOR-60 reactor has been put in operation (1969), more than 80 control rods of different design were successively inserted in it and 20 rods were later analyzed in the material hot cells in detail (Tab.5.1).

During operation of the control rods in the BOR-60 reactor no case of failure of the main function actions was observed due to decreasing of their operational reliability. In this case no marked loss of physical effectiveness as well as break of mobility in the guide cartridges were observed due to failure of the rods. There took places three cases of premature withdrawal of the control rods from the reactor zone because of the cartridges of CS rods were failed.

**CR.** The rods containing boron carbide (thermally recovered with carbon and magnesium), chromium and tantalum diborides and europium oxide as well as the trap-rod (europium oxide and zirconium hydride) were investigated.

**RR-2.** The rods of three design were investigated:
- seven AE sealed with gas collector;
- seven AE vented with sodium sub-layer;
- one AE vented with sodium sub-layer.

**RR-1, SR.** The rods composed of the sealed AE and containing boron carbide (thermally recovered with carbon and magnesium) as well as the SR after prolonged storage in the cooling pond and the SR containing refabricated boron carbide were investigated. The main results are presented in Tab.5.2.

### TABLE 5.1. MAIN CHARACTERISTICS OF INVESTIGATED CONTROL RODS

<table>
<thead>
<tr>
<th>Rod No.</th>
<th>Design type</th>
<th>Absorber</th>
<th>Filling in gap</th>
<th>Operation conditions</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Oper time, eff day</td>
</tr>
<tr>
<td>RR-2</td>
<td>1057A-0 7AE</td>
<td>$B,C$</td>
<td>air</td>
<td>215</td>
</tr>
<tr>
<td>RR-2</td>
<td>1057-D 7AE</td>
<td>$B,C$</td>
<td>sodium</td>
<td>252</td>
</tr>
<tr>
<td>RR-2</td>
<td>1057-D 7AE</td>
<td>$B,C$</td>
<td>sodium</td>
<td>575</td>
</tr>
<tr>
<td>RR-3</td>
<td>1009 7AE</td>
<td>$B,C$</td>
<td>sodium</td>
<td>540</td>
</tr>
<tr>
<td>CR-2</td>
<td>1065A-O</td>
<td>$B,C$</td>
<td>air</td>
<td>215</td>
</tr>
<tr>
<td>CR-1</td>
<td>1057A-O</td>
<td>$CrB_2$</td>
<td>air</td>
<td>306</td>
</tr>
<tr>
<td>CR-2</td>
<td>1057A-O</td>
<td>$B,C$ magn</td>
<td>air</td>
<td>271700</td>
</tr>
<tr>
<td>CR-2</td>
<td>1065D-O</td>
<td>$EuO_2$</td>
<td>helium</td>
<td>348</td>
</tr>
<tr>
<td>CR-2</td>
<td>1847</td>
<td>$EuO_2$</td>
<td>helium</td>
<td>729</td>
</tr>
<tr>
<td>CR-2</td>
<td>2121</td>
<td>$ZrH_2$</td>
<td>helium</td>
<td>542</td>
</tr>
<tr>
<td>RR-1</td>
<td>1057A-O</td>
<td>$B,C$</td>
<td>air</td>
<td>306</td>
</tr>
<tr>
<td>RR-1</td>
<td>1057A-O</td>
<td>$B,C$</td>
<td>sodium</td>
<td>562</td>
</tr>
<tr>
<td>SR-2</td>
<td>1057A-O</td>
<td>$B,C$</td>
<td>helium</td>
<td>1015</td>
</tr>
<tr>
<td>SR-2</td>
<td>1057A-O</td>
<td>$B,C$</td>
<td>helium</td>
<td>537</td>
</tr>
<tr>
<td>SR</td>
<td>1057A</td>
<td>refab $B,C$</td>
<td>helium</td>
<td>416 18-year stor</td>
</tr>
</tbody>
</table>
TABLE 5.2. RESULTS OF ROD INVESTIGATIONS

<table>
<thead>
<tr>
<th>Rod</th>
<th>Design</th>
<th>Testing conditions</th>
<th>Results</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>CR</td>
<td>80% B(_2)C thermally recovered with carbon</td>
<td>205320 MWh</td>
</tr>
<tr>
<td></td>
<td></td>
<td>4 AE sealed with gas collector</td>
<td>2.7x10(^{22}) n/cm(^2) E&gt;0 max.burnup - 5.2%</td>
</tr>
<tr>
<td>2</td>
<td>80% B(_2)C thermally recovered with magnesium</td>
<td>271700 MWh</td>
<td>Wrapper and AE without defects. Increase of AE diameter by 0.5-1%. 25% gas release under irradiation. The most part of B(_4)C are intact. Great spread under swelling of pellets (%) 1-3.5 - top; 3.5-8.5 - bottom. Mechanical properties are similar to those of the first.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>4 AE sealed with gas collector</td>
<td>3.6x10(^{22}) E&gt;0 max.burnup 2.9%</td>
</tr>
<tr>
<td></td>
<td>Cr(_2)B(_2)+Ta(_2)B(_2) - 80%</td>
<td>293457 MWh</td>
<td>Wrapper and AE without defects. Unchanged diameter of AE. Gas release at irradiation - 77%. Swelling of pellets (%) 1.6 - top; 2.0 - center; 3.0 - bottom.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>4 AE sealed with gas collector</td>
<td>3.6x10(^{22}) E&gt;0 max.burnup 2.6%</td>
</tr>
<tr>
<td>3</td>
<td>Eu(_2)O(_3)</td>
<td>337906 MWh</td>
<td>Wrapper and AE without defects. Unchanged diameter of claddings. 10.95-11.09 mm diameter of pellets (11.2 mm I.D.). Unchanged mechanical properties.</td>
</tr>
<tr>
<td></td>
<td>7 AE sealed</td>
<td>4.7x10(^{22}) E&gt;0</td>
<td></td>
</tr>
<tr>
<td>4</td>
<td>Eu(_2)O(_3)</td>
<td>787000 MWh</td>
<td>Wrapper and AE without defects. Increase of wrapper diameter by 1.5%. Increase of AE cladding diameter by 1%. Pellets diameters - 10.95-11.09 mm. Unchanged mechanical properties.</td>
</tr>
<tr>
<td></td>
<td>7 AE sealed</td>
<td>10.5x10(^{22}) E&gt;0</td>
<td></td>
</tr>
</tbody>
</table>
TABLE 5.2. (cont.)

<table>
<thead>
<tr>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
</tr>
</thead>
</table>
| CR-trap  
1 AE - Eu$_2$O$_3$ sealed  
1 ME - ZrH$_2$ vented | 520367 MWh  
6.9x10$^8$ E$>0$ | Block of moderator without defects. Unchanged diameter of cladding. Hydride blocks - ( %) - 0.5 - 1.3. Unchanged hydrogen content. Failed absorber block. Increase of outer cladding diameter by 1.3%. Eu$_2$O$_3$ swelling ( %) 2.8 - top; 4.1 - bottom. Mechanical testing at 350° and 650° - zero plasticity. |  |
| CR-trap  
1 AE - Eu$_2$O$_3$ sealed | 601121 MWh  
6.4x10$^8$ E$>0$ | AE and ME without defects. Unchanged diameter of ME. 39.52-39.65 mm AE diameter (initial value - 39.5 mm) |  |
| RR-2  
7 AE sealed with gas collector B$_2$C 80%. | 205320 MWh  
4.2x10$^7$ E$>0$  
burnup - 3.1% | Wrapper without defects. AE - residual bending - 20-30 mm. Cross destruction of one peripheral AE. 1-2% increase of cladding diameter in nondestructive AE and 4% - in destructive regions. Four sealed AE, 1-10% gas release. Tight-fitting pellets to cladding. Swelling of pellets: ( %) 4.3-4.9 - bottom; 4.2-4.6 - center; 4.0-5.1 - top. Change of mechanical properties of claddings for all AE zones; the more fluence. the more changes. |  |
| RR-2  
7 AE vented with Na sub-layer | 552000 MWh  
8.7x10$^8$ E$>0$  
max.burnup 9.0% | Wrapper and AE without defects. Increase of AE cladding diameter by 1.4% at the bottom. Increase of pellet diameter by 3-4% at the top and center and 5-6% at the bottom. Mechanical properties of cladding: the more strength, the more fluence and the low irradiation temperature. 3-4% relative elongation for all sections at 20° and 1-2% at 350-550°C. Compatibility: B$_2$C-cladding interaction in the center of peripheral AE - 0.15mm. |  |
| 1 AE vented with Na sub-layer | 518000 MWh  
E$>0$  
10.3x10$^{22}$  
burnup - 7.2% | Wrapper and AE without defects. Increase of cladding diameter by 1.9% at the bottom. B$_2$C pellet swelling ( %) 2.2 - top; 3.6 - center; 6.6 - bottom. Mechanical properties were hold at the high level. |  |
<table>
<thead>
<tr>
<th></th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
</tr>
</thead>
</table>
| RR-1 | 7 AE sealed with gas collector  
B₄C thermally recovered with carbon | 293457.4 MWh  
4.8x10⁻²²E>0  
average burnup - 2.3% | Wrapper without defects. 5 mm sag of AE. Unchanged diameters of claddings. Sealed AE. 1-10% gas release. 0.2 -1.2% increase of cladding diameter. |
|   | 7 AE sealed with gas collector  
B₄C thermally recovered with magnesium | 282700.4 MWh  
5.6x10⁻²² (E>0) | Wrapper and AE without defects.  
Unchanged cladding diameter. 10-20.5% gas release. 2.5-6% (bottom) increase of pellet diameter. |
| SR | 7 sealed AE | 974276 MWh  
1.2x10⁻²²E>0  
max. burnup -4.3% | Wrapper without defects. Severe bending of peripheral AE. Increase of cladding diameter (bottom) up to 1-2%. Gas release 18-24%. Destructed pellets of peripheral AE. 4.5-4.8% (bottom) swelling of pellets ( %). |
| SR | 7 sealed AE | 516300 MWh  
5x10⁻²²  
1-2% average burnup  
18-year storage in pond | Wrapper and AE without defects.  
1.5% max. increase of gas collector and AE diameter. 40% destruction of B₄C pellets. Diameter of B₄C pellets did not exceed 11.0 mm. Mechanical properties of cladding: increased strength, high plasticity. |
| SR | 7 sealed AE  
refabricated B₄C | 549550.4 MWh  
4.2x10⁻²²E>0.1 | Wrapper and AE without defects.  
39.5-39.68 mm (39.5 mm initial) diameter of wrapper. Unchanged diameter of claddings. Shape and integrity of boron carbide pellets were kept up. 5.8-7.7% gas release. 10.85-11.01 mm diameter of pellets. Mechanical properties of cladding: 2.9-6.5% plasticity, 975-1525 MPa ultimate strength within 20-600°C. |
PRODUCTION OF GAMMA-SOURCES, BASED ON EUROPIUM OXIDE IN FAST REACTORS

V.D. RISOVANY, A.V. ZAKHAROV, E.P. KLOCHKOVA,
T.M. GUSEVA
Research Institute of Atomic Reactors,
Dimitrovgrad

V.B. PONOMARENKO, V.M. CHERNYSHOV
Moscow Polymetal Factory,
Moscow

Russian Federation

Abstract

The paper presents the basic results of europium oxide operating properties, its physical efficiency, radiation resistance and compatibility with stainless steel; it considers the structures and characteristics of γ-sources made of europium oxide irradiated of nuclear reactors controls absorbing cores sets.

INTRODUCTION

Conditions of fast reactors operation are characterized by the hard neutron spectrum, high neutron fluences and operating temperature, corrosive medium which makes a number of requirements for controls absorbing materials, the main of such requirements are:

1. High neutron absorbing efficiency and its storage in the process of operation.
2. Resistance to radiation damages that are revealed in change of the form, sizes, structure, physico-mechanical properties in the process of reactor irradiation.
3. Compatibility with structural materials.

In the aggregate of these requirements europium oxide is one of the most promising materials which is confirmed by great positive experience of its usage not only in the BN-reactors controls, but in thermal neutron reactors as well. At the same time in the last years there appeared a tendency on reduction of its usage in Russian nuclear reactors. It is caused first of all by the enhancing requirements for ecology and safe operation of nuclear reactors. Under reactor irradiation in europium oxide accumulated are europium radionuclides $^{152}$Eu, $^{154}$Eu, $^{155}$Eu, which arises serious problems with storage, reprocessing and transportation of radioactive wastes and also can cause catastrophic consequences in case of damage of the nuclear reactor controls and core. Still, complete refusal against usage of europium oxide and materials on its basis with account of their unique properties is not expedient. Their high activity after reactor irradiation can be effectively used in powerful γ-facilities instead of the applied $^{60}$Co and $^{137}$Cs based sources.

The present paper presents the basic results on operating properties of europium oxide, its physical efficiency, radiation resistance, compatibility with stainless steel, and considers the structures and characteristics of γ-sources made of europium oxide irradiated in the set of nuclear reactors controls absorbing cores.
1. Nuclear and physico-mechanical properties

Properties of europium oxide are described in detail in refs. [1-3]. Table 1 presents the most important characteristics from the operation viewpoint.

Eu oxide with the monoclinic structure has the higher nuclear density, so its usage in this modification as an absorbing material is most preferable.

The high structure stability and melting temperature of Eu oxide (2330°C [2]) should be noted. The disadvantage includes its low thermal conduction (2-3 W/m°C)

<table>
<thead>
<tr>
<th>TABLE 1. BASIC PHICIAL CHARACTERISTICS OF EUROPIUM SESQUIOXIDE [2]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Properties</td>
</tr>
<tr>
<td>------------</td>
</tr>
<tr>
<td>1. Melting point, °C</td>
</tr>
<tr>
<td>2. Density, g/cm³</td>
</tr>
<tr>
<td>3. Absorber cation density, atom/cm³</td>
</tr>
<tr>
<td>4. Thermal conductivity coefficient, W/m°K</td>
</tr>
<tr>
<td>- at 25°C</td>
</tr>
<tr>
<td>- at 1000°C</td>
</tr>
<tr>
<td>5. Thermal expansion coefficient, (°C)</td>
</tr>
<tr>
<td>- from 30 to 840°C</td>
</tr>
<tr>
<td>- from 0 to 1000°C</td>
</tr>
<tr>
<td>6. Young’s modulus, N/mm²</td>
</tr>
<tr>
<td>7. Poisson’s ratio</td>
</tr>
<tr>
<td>8. Transformation rate</td>
</tr>
<tr>
<td>- at 1100°C</td>
</tr>
<tr>
<td>- at 1300°C</td>
</tr>
</tbody>
</table>
which sets restrictions for production of the large diameter absorbing core because of its great heating under reactor irradiation.

Eu possesses unique nuclear properties being the best representative in view of this index among all the known neutron absorbers. Natural Eu consists of two isotopes $^{151}$Eu and $^{153}$Eu with the mass content 47.8 and 52.2% [2]. Their thermal neutron-absorption cross-sections make up 9,000 and 420 barns [2], respectively. It is notable that europium effectively absorbs fast neutrons too, inferior only to isotope $^{10}$B. It is this fact that substantiates usage of Eu oxide in fast reactors controls.

During neutrons capture there are produced daughter isotopes $^{155}$Eu, $^{152}$Eu, $^{154}$Eu, $^{156}$Eu, $^{157}$Eu, the fission products of which are gadolinium isotopes (Tab.2 /2/). All these isotopes are also characterized by high neutron-absorption cross-sections. Therefore, physical efficiency of neutron absorption by Eu oxide is not practically decreased under long-term reactor irradiation.

**TABLE 2. SCHEME OF NUCLEAR TRANSFORMATION AND EUROPIUM NUCLIDES DISINTEGRATION [2]**

<table>
<thead>
<tr>
<th>Izotop</th>
<th>Thermal neutron absorption cross section</th>
<th>Half-life</th>
<th>Disintegration products</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{151}$Eu</td>
<td>9000 barn</td>
<td>stable (47,8%)</td>
<td></td>
</tr>
<tr>
<td>$^{152}$Eu</td>
<td>5000 barn $\beta^-$ 13,54 years</td>
<td>$^{152}$Gd</td>
<td></td>
</tr>
<tr>
<td>$^{153}$Eu</td>
<td>420 barn</td>
<td>stable (52,2%)</td>
<td></td>
</tr>
<tr>
<td>$^{154}$Eu</td>
<td>1500 barn $\beta^-$ 8,59 years</td>
<td>$^{154}$Gd</td>
<td></td>
</tr>
<tr>
<td>$^{155}$Eu</td>
<td>14000 barn $\beta^-$ 1,7 years</td>
<td>$^{155}$Gd</td>
<td></td>
</tr>
<tr>
<td>$^{156}$Eu</td>
<td>$\beta^-$ 15,4 days</td>
<td>$^{156}$Gd</td>
<td></td>
</tr>
<tr>
<td>$^{157}$Eu</td>
<td>$\beta^-$ 15,4 hours</td>
<td>$^{157}$Gd</td>
<td></td>
</tr>
</tbody>
</table>
2. Radiation resistance

Numerous investigations showed that the size stability of Eu oxide under reactor irradiation does not practically depend on the spectrum and neutron fluence, but is determined by the irradiation temperature alone (Tab.3) [4]. The samples retain their shape and integrity, swelling does not exceed 1%, if the following conditions are satisfied:
- maximal irradiation temperature should not exceed 1600-1700°C;
- temperature gradient should be less than 45°C/mm.

**TABLE 3. SIZE STABILITY OF EUROPIUM SESQUIOXIDE SPECIMENS UNDER IRRADIATION**

<table>
<thead>
<tr>
<th>Irradiation conditions.</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>fluence</strong> $10^{21}$ cm$^{-2}$ (E&gt;$0.1$MeV)</td>
<td><strong>Temperature, °C</strong></td>
</tr>
<tr>
<td></td>
<td>on the surface</td>
</tr>
<tr>
<td>0.22</td>
<td>890-930</td>
</tr>
<tr>
<td>0.39</td>
<td>760-800</td>
</tr>
<tr>
<td>1.40</td>
<td>920-960</td>
</tr>
<tr>
<td>2.72</td>
<td>1100-14000</td>
</tr>
<tr>
<td>70.0</td>
<td>390-440</td>
</tr>
<tr>
<td>80.0</td>
<td>630-680</td>
</tr>
<tr>
<td>120.0</td>
<td>630-680</td>
</tr>
</tbody>
</table>

Figs. 1,2 present cross-sections and structures of Eu oxide after reactor irradiation. The number of cracks increases with growth of the temperature gradient. The structure of Eu oxide up to 1600°C does not practically differ from the initial one. Increase of the irradiation temperature to 1800-1850°C led to increase of the pellets diameter up to 4-5%. Empty voids were formed in their centre. Three zones are distinctly seen in the structure (Fig.2). Closer to the centre there are column grains
Figure 1. Cross section of Eu₂O₃ pellets after irradiation:

a = \( t_{\text{max}} \approx 900^\circ\text{C} \); \( F = 1.2 \times 10^{23} \text{ cm}^{-3} \)
b = \( t_{\text{max}} \approx 1400^\circ\text{C} \); \( F = 1.4 \times 10^{21} \text{ cm}^{-2} \)
c = \( t_{\text{max}} \approx 1850^\circ\text{C} \); \( F = 2.72 \times 10^{21} \text{ cm}^{-3} \)
Figure 2. Eu$_2$O$_3$ structure after irradiation
($t_{\text{max}} = 1850^\circ\text{C}$; $F=2.72 \times 10^{20}$ n/cm$^2$)

Eu oxide does not interact with austenitic class steels up to 900-1000$^\circ$C. The main components of stainless steels are iron, chromium, nickel, molybdenum, manganese. It is important that full energy of Eu oxide is greater than for these elements oxides [2]. So free energy of the reaction $2\text{Me} + \text{Eu}_2\text{O}_3 \rightarrow \text{Me}_2\text{O}_3 + 2\text{Eu}$ (where Me = Cr, Mn, Fe, Mo, Ni) will be positive and proceeding of these reactions is not expected.
Figure 3. Structure of Eu₂O₃

a - t = 1560°C, τ = 1 h., x 100
b - t = 1700°C, τ = 1 h., x 200
c,d - t = 1800°C, τ = 1 h., x 200
Samples Eu$_2$O$_3$ + Me (where Me = Ni, Cr, W, Nb, Mo, Ta, Re, Cu) were produced for reactor tests. Irradiation was conducted in special capsules in the BOR-60 reactor in helium atmosphere at 400-800°C up to fast neutrons fluence 2×10$^{22}$cm$^{-2}$ (E > 0.1 MeV). The samples retained integrity, the volume changed by a factor of 2-3%, interaction traces were absent (Fig.4) which confirms high compatibility of Eu oxide with these metals.

![Figure 4. Typical structure of Eu$_2$O$_3$ + Me (where Me=Mo,Re,Ta, W, Cu, Cr, Ni) after irradiation (t=400-800°C, F=2×10$^{22}$n/cm$^2$), x200](image)

4. Eu oxide-containing controls serviceability

Typical structures of absorbing elements with Eu oxide at fast nuclear reactors are displayed in Fig.5.

Post-reactor material science investigations showed that the main factor, restricting the operating time of these structures, is low radiation resistance of structural materials. On achieving the fast neutrons fluence 1.2×10$^{22}$ cm$^{-2}$ (E > 0.1 MeV) swelling of the wrapper and claddings up to 10-12% occurred, while all the Eu oxide pellets retained their sizes, shape, integrity (Fig.6). There were no traces of pellets-claddings interaction (OCr18Ni10Ti, Cr15Ni16Mo3Nb). Any noticeable reduction of physical efficiency of neutron absorption by controls was not detected. If the structural materials were replaced by more radiation resistant ones, the Eu oxide-containing controls operating time can be increased.
Figure 5. Typical design of the control element with \( \text{Eu}_2\text{O}_3 \) of the control system rods of the reactors BN-600 and BOR-60.

5. Spectral characteristics of Eu radionuclides

Main characteristics of some Eu radionuclides compared to \( ^{60}\text{Co} \) are given in Table 4. As the Table shows, \(^{152}\text{Eu} \) and \(^{154}\text{Eu} \) have spectral characteristics that are rather attractive for \( \gamma \)-sources. 45% of probability of \( \gamma \)-quanta yield per 1 decay have the energy 0.866...1.21 MeV for \(^{152}\text{Eu} \), and for \(^{154}\text{Eu} \) 65% of decay probability have the energy 0.72...1.0 MeV.

In terms of the values for \( \gamma \)-constant K and radioactivity equivalent 1 mCi/1 mgRa the radionuclides \(^{152}\text{Eu} \) and \(^{154}\text{Eu} \) are 2 fold inferior to the most commonly used industrial \( \gamma \)-source \(^{60}\text{Co} \), but they considerably exceed the analogous values for \(^{137}\text{Cs} \), that is also widely used as \( \gamma \)-sources at the industrial facilities.

The advantages of \(^{152}\text{Eu} \) and \(^{154}\text{Eu} \) radionuclides over \(^{60}\text{Co} \) include the larger half-life period and decrease the costs due to reducing the expenses for their power production (larger neutron-absorption cross-section, irradiation in the controls set).
Figure 6. Pellets of Eu$_2$O$_3$ after irradiation 
(t = 900°C, F = 1, 2x10$^{23}$n/cm$^2$)

TABLE 4. CHARACTERISTICS OF NUCLIDE $^{152}$Eu, $^{154}$Eu, $^{155}$Eu, $^{60}$Co

<table>
<thead>
<tr>
<th>Parameter</th>
<th>$^{152}$Eu</th>
<th>$^{154}$Eu</th>
<th>$^{155}$Eu</th>
<th>$^{60}$Co</th>
</tr>
</thead>
<tbody>
<tr>
<td>Life-time, $T_{1/2}$, year</td>
<td>13.54</td>
<td>8.59</td>
<td>1.7</td>
<td>5.27</td>
</tr>
<tr>
<td>Full $\gamma$-constant, $K_\gamma$, R/h</td>
<td>6.35</td>
<td>6.70</td>
<td>0.33</td>
<td>13.2</td>
</tr>
<tr>
<td>Equivalent of 1$m$Ku isotope per 1 mg Ra</td>
<td>0.60</td>
<td>0.74</td>
<td>0.01</td>
<td>1.54</td>
</tr>
<tr>
<td>Average energy, MeV</td>
<td>0.709</td>
<td>0.745</td>
<td>0.1</td>
<td>1.252</td>
</tr>
</tbody>
</table>

6. Designs and characteristics of $\gamma$-sources

At SSC RIAR (Dimitrovgrad) there were developed the first standard $\gamma$-sources from the Eu oxide-containing absorbing elements spent in the SM-2 reactor. Preliminarily prepared absorbing elements, 4.1 mm in diameter and with the wall thickness 0.3 mm, are located in the cladding, 11 x 1 mm in diameter (Fig.7). The elements are 75-400 mm long. The exposure dose rate (EDR) of the Eu oxide based sources at a distance of 1 m is equal to 0.2...0.6 r/s which is comparable to the $^{60}$Co based sources.

Fig.8 demonstrates the $\gamma$-source design with an insert made of Eu oxide or Eu$_2$O$_3$+Co composition. The inserts are pre-irradiated as an absorbing core of the controls. Characteristics of various source modifications are presented in Tab.5
Figure 7. Gamma-source design based on the control elements with which were worked off in the reactor SM-2

Figure 8. Gamma-source design based on europium and cobalt.
1 - active core Eu, Co+Eu
2 - inner hermetic capsule
3 - external hermetic cover.
Figure 9. External view of the "rod-trap" with Eu$_2$O$_3$ (a) and cross sections (b) after exploitation in the reactor BOR-60
### TABLE 5. BASIC CHARACTERISTICS OF GAMMA-SOURCES

<table>
<thead>
<tr>
<th>Source Type and number</th>
<th>External sizes, mm</th>
<th>Active core sizes, mm</th>
<th>Exposure dose power of gamma-radiation at 1m from the working surface of a source.</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Diamet. D</td>
<td>Height H</td>
<td>Diamet. D</td>
</tr>
<tr>
<td>GSEC-7-2</td>
<td>11.0</td>
<td>81.0</td>
<td>8.0</td>
</tr>
<tr>
<td>GSEC-7-3</td>
<td>11.0</td>
<td>81.0</td>
<td>8.0</td>
</tr>
<tr>
<td>GSEC-7-4</td>
<td>11.0</td>
<td>81.0</td>
<td>8.0</td>
</tr>
<tr>
<td>GSEC-7A-1</td>
<td>12.8</td>
<td>86.0</td>
<td>10.0</td>
</tr>
<tr>
<td>GSEC-7A-2</td>
<td>12.8</td>
<td>86.0</td>
<td>10.0</td>
</tr>
<tr>
<td>GSEC-7A-3</td>
<td></td>
<td></td>
<td>0.485</td>
</tr>
<tr>
<td>GSEC-7A-4</td>
<td></td>
<td></td>
<td>0.775</td>
</tr>
<tr>
<td>GSEC-12A-1</td>
<td>11.0</td>
<td>99.0</td>
<td>8.0</td>
</tr>
<tr>
<td>GSEC-12A-2</td>
<td></td>
<td></td>
<td>0.387</td>
</tr>
<tr>
<td>GSEC-12A-3</td>
<td></td>
<td></td>
<td>0.485</td>
</tr>
<tr>
<td>GSEC-12A-4</td>
<td></td>
<td></td>
<td>0.775</td>
</tr>
</tbody>
</table>

**Conclusion**

Great positive experience has been gained on long-term operation of Eu oxide-containing controls in the fast nuclear reactors BN-600 and BOR-60. Eu oxide has high neutron absorption efficiency which does not practically change under reactor irradiation. At the irradiation temperature up to 1600-1700°C it retains its shape, integrity, swelling does not exceed 1% up to achieving the fast neutrons fluence $1.2 \times 10^{22}$ cm$^{-2}$ (E $> 0.1$ MeV). Eu oxide does not interact with austenitic steels up to the irradiation temperature 900-1000°C. The operating time of Eu oxide-containing controls in the BN-reactors is limited to degradation of the properties of stainless steel elements.

High-active, long-lived Eu radionuclides $^{152}$Eu, $^{154}$Eu, $^{155}$Eu are accumulated in Eu oxide under reactor irradiation which makes it possible to use them as $\gamma$-sources. First designs of sources based on controls spent in nuclear reactors were produced. Two-purpose application of absorbing cores, made of Eu oxide and materials on its
basis, allows reduction of expenses for controls operation, solution of the problem of radioactive wastes utilization, production of cheaper $\gamma$-sources compared to $^{60}$Co and $^{137}$Cs ones with comparable operating characteristics.

REFERENCES

3. Беляев Р.А. и др. Свойства окислов европия. М., Атомиздат, 1974, с.52
5. Кошкин В.М. и др. Аномальная радиационная стойкость рыхлых кристаллических структур// Физика и техника полупроводников, 1984 г., Т.18, вып.8, С.1373-1378.
REPROCESSING OF THE IRRADIATED BORON CARBIDE 
ENRICHED BY BORON-10 ISOTOPE AND ITS REUSE IN THE 
CONTROL RODS OF THE FAST BREEDER REACTORS

V.D. RISOVANY, A.V. ZAKHAROV, E.P. KLOCHKOV, 
A.G. OSIPENKO, N.S. KOSULIN, G.I. MIKHAILICHIENKO 
Research Institute of Atomic Reactors, Dimitrovgrad, 
Russian Federation

Abstract

The present paper discusses the development of technology for reprocessing of 
irradiated boron carbide, which provides complete removal of radionuclides from irradiated 
materials. This technology allows the repeated use of B\textsubscript{10} enriched with B\textsubscript{4}C in fast 
reactors.

As a neutron absorber in the control rods of the fast breeder reactors the boron 
carbide enriched up to 80\% of boron-10 isotope is used. This is the only material 
having the sufficient physical absorbing efficiency in the fast neutron spectrum to 
perform the function of the emergency reactor shutdown. Due to the enrichment this 
material is very expensive. On the other side, according to its in-reactor operating 
conditions the safety rods are the most part of the time in the elevated position and 
the absorbing pin is subject to the neutron flux only in its lower part that is adjacent 
the core. The examination of several safety rods discharged from the BN-600 reactor 
demonstrated that the maximum burnup of the boron-10 isotope in the lower part of 
the pin didn't exceed 5\%. In the upper part of the safety rods the burnup of boron-10 
in the pin was not noticed.

Thus the in-reactor operation of the safety rods is finished when they still have 
a great physical efficiency for neutron absorbtion. The reason for this lies in the 
radiation damage of the steel structural parts and first of all those located in the lower 
part of the pin and those operating in the reactor core (Fig.1). The SSC RIAR has 
performed the examination of 4 safety rods spent in the BN-600 reactor from 100 to 
330 effective days. It was pointed out that the lower extension part of the absorbing 
core, discharged after 330 days of operation, had a substatial bend caused by the non-
uniform material swelling (Fig.2). Fracture toughness of the material in this part (steel 
09Cr18Ni1OTi) is very low (less than 0.3 MJ/m\textsuperscript{2}) and when testing on the impact 
machine the samples are subject to brittle failure. But at the same time the ductility 
of the cladding material of the absorbing pin (steel 08Cr16Ni15Mo3Nb) was 
decreased up to 3\% at 500\°C.

This testifies that the service life potential of the safety rods of this design is 
exhausted. To prolong its service life the substantial modernization is required. First 
of all it is necessary to replace the structural materials for the more radiation resistant 
materials. This procedure is being developed and during the last campaigns the new 
safety rods of more radiation resistant steels have been tried in the BN-600 reactor.

Considerable benefit to the extension of the safety rod service life can be 
achieved if the design of the rod is updated in such a way that it does not have the 
lower extension part. the radiation damage of which limits the service life of the safety 
rods.

These measures can to some extent prolong the service life of safety rods but they 
are not capable to resolve the problem of the rational use of the expensive enriched 
boron carbide.
FIG. 1. SR design of a- reactor BOR-60, b- reactor BN-600.
This problem can be resolved when the pin is reused for manufacturing the new scram rods and this feasibility was tried in the BOR-60 reactor of the SSC RIAR. The safety rods were withdrawn from the storage pool where they had been stored for 18 years. The material science examination demonstrated that the properties of the most part of the pellets (about 40%) of the enriched boron carbide from the absorbing pins were quite satisfactory and they were feasible for manufacturing of the new rods. The only problem we faced at the early stage of this manufacturing process was the presence of the induced pellet γ-activity. Different methods were used for decontamination of pellets in a hot cell. But the activity was reduced only by 8-10 times. Thus all the operations related to the rod fabrication were carried out in shielded boxes. The finished rod also had γ-activity but with certain precaution measures it was suitable for putting it into the reactor.

This rod has been successfully operated for 416 effective days in the BOR-60 reactor and after that it was discharged and examined in hot cells. It did not have any noticeable damages of the absorbing pin cladding, wrapper tube, welds, etc. The pellets in the majority of the absorbing pins were in good state, had no swelling and could be used for fabrication of the similar safety rod.

This was only partial solution of the problem. Therefore the SSC RIAR continued searching for options to use the enriched boron carbide from the spent safety rods. This resulted in the development of the flowsheet involving the use of the boron carbide pellets discharged from the spent safety rods (Fig.3).

The first option provides for the simple use of the boron carbide pellets having the sufficient quality for fabrication of the new safety rods as it was done for the BOR-60.

Depending on the state of the rod the quantity of such pellets may range from 10 to 15%.

The second option provides for the mechanical reprocessing of the part of sub-standard pellets and their fragments. They are milled to powder which is then pressed and sintered into new pellets. This option enables reprocessing practically of the whole core provided that the induced γ-activity is not very high, otherwise it can complicate manufacturing of the rod and further operations with it.
FIG. 3. Technological scheme of the complex remarking of worked off fast neutron reactor safety rods and enriched boron carbide.
The third option provides for full chemical reprocessing of the core. This reprocessing technology has been deeply studied at the SSC RIAR. The boron carbide through several stages is transformed into the boron acid which is then synthesized into the boron carbide powder and sintered into the new pellets for the safety rods.

In this case the material gets completely rid of the radioactive impurities and the finished product has no $\gamma$-activity. During the investigations of the BN-600 safety rods it was established that the main $\gamma$-activity source were the technological microimpurities of Eu. The radiation intensity of the long-lived $^{152}$Eu and $^{154}$Eu isotopes was by several orders greater than that of other impurities (Fig.4). As this takes place, the correlation of the radiation intensity of these isotopes from burnup of $^{10}$B in pellets is practically absent.

It is practically impossible to remove the radioactive Eu impurities providing decontamination by different solutions. In the process of the chemical reprocessing the impurities are completely removed and as a result of this the $\gamma$-activity is lowered up to the natural level.

The SSC RIAR has engineering possibilities for reprocessing of all the enriched boron carbide built up in the storage pools. This makes it possible to empty the storages from spent rods, utilize wastes and to provide safety rods for such reactors as BOR-60 and BN-600. The preliminary works suggest that this technology is economically efficient.

\[ \text{FIG. 4. Gamma-radiation intensity of Eu isotopes in absorbing elements of a lower link of SR.} \]
<table>
<thead>
<tr>
<th>Rod and efpd</th>
<th>Link</th>
<th>Absorb. element region</th>
<th>Test. temp. T°C</th>
<th>Ultimate strength MPa</th>
<th>0.2%Yield strength MPa</th>
<th>Total elongat %</th>
<th>Uniform elongat %</th>
</tr>
</thead>
<tbody>
<tr>
<td>T-08 101 efpd</td>
<td>top</td>
<td>middle</td>
<td>20</td>
<td>670</td>
<td>440</td>
<td>16.2</td>
<td>15.3</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>500</td>
<td>540</td>
<td>340</td>
<td>9.5</td>
<td>8.8</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>600</td>
<td>490</td>
<td>300</td>
<td>10.2</td>
<td>9.8</td>
</tr>
<tr>
<td></td>
<td>bottom</td>
<td>bottom</td>
<td>20</td>
<td>750</td>
<td>480</td>
<td>15.4</td>
<td>14.2</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>500</td>
<td>580</td>
<td>380</td>
<td>9.8</td>
<td>9.2</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>600</td>
<td>550</td>
<td>360</td>
<td>8.8</td>
<td>8.2</td>
</tr>
<tr>
<td>T-02 331 efpd</td>
<td>top</td>
<td>middle</td>
<td>20</td>
<td>670</td>
<td>470</td>
<td>11.7</td>
<td>10.7</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>500</td>
<td>280</td>
<td>260</td>
<td>5.2</td>
<td>3.1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>600</td>
<td>570</td>
<td>410</td>
<td>9.2</td>
<td>7.6</td>
</tr>
<tr>
<td></td>
<td>bottom</td>
<td>bottom</td>
<td>20</td>
<td>880</td>
<td>700</td>
<td>9.0</td>
<td>8.0</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>500</td>
<td>650</td>
<td>540</td>
<td>4.4</td>
<td>3.1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>600</td>
<td>600</td>
<td>510</td>
<td>4.6</td>
<td>3.5</td>
</tr>
</tbody>
</table>
EXPERIENCE IN PRODUCTION OF ARTICLES FROM BORON CARBIDE FOR FAST REACTOR CONTROL RODS

I.A. BAIRAMASHVILI
Institute of Stable Isotopes
Tbilisi, Georgia

Abstract

The paper presents a multi-stage process, including separation of boron-10 and boron-11, stable isotopes, production powder and the various configuration articles from enriched boron carbide.

INTRODUCTION

Since 1964 our institute has intensively been conducting works connected with $^{10}$B$_4$C powder production by various methods. In 1989 the institute started the production of compact articles from boron carbide powder with various configurations produced by us to complete the absorbing rods for the control and protection systems /CPS/ for nuclear power reactors on fast neutrons, (BR-10, BOR-60, BN-350 and BN-600).

Articles production from $^{10}$B$_4$C is a complex and multi-stage process including the separation of B$^{10}$ and B$^{11}$ stable isotopes. All the steps of the technological process are accompanied by respective control.

The scientific and production experience gained allows to draw some conclusions on the advisability of research and development work fulfilment in two main directions: intensification of the technological processes and improvement of the economic index of production.

PRODUCTION OF $^{10}$B$_4$C POWDERS

Currently boron carbide is the main absorber material used in CPS elements for fast reactors. This position has been strengthened as compared to 1983 with the development of nuclear power and relatively constant need in boron carbide containing a highly enriched boron-10 stable isotope. In this connection it is time to analyse the methods for its production. It should be taken into consideration that boron isotopes separation technology includes the production of an elemental material as amorphous powder. Together with this other compounds are produced as well which contain highly enriched boron-10 and boron-11 /the gas BF$_3$, the salt KBF$_4$, boric acid H$_3$BO$_3$, etc./. In all the cases BF$_3$ is the initial material thereby leaving a certain nuance on the technological process of elemental boron extraction or its remaking into other boron-containing compositions.

I.G.Gverdtsiteli Institute of Stable Isotopes in 1964 started to perform works on the production of powder and articles from $^{10}$B$_4$C. The work at the very beginning was carried out in two directions -the production of $^{10}$B$_4$C articles by open hot pressing of an elemental boron-10 and carbon mixture as well as the production of $^{10}$B$_4$C powder by a method for magnesio-thermal reduction of the salt of potassium tetrafluoroborate /K$^{10}$BF$_4$/ in the presence of carbon which is subsequently hot pressed.
In the first case, the process of \(^{10}\text{B}_4\text{C}\) synthesing was combined with the process of the article formation. However the shortcomings of such a technology were soon revealed - the articles formed had an instable chemical content, differed in non-uniform distribution along the boron, boron carbide and carbon samples section, high porosity, a tendency to crack-forming, the complexity of the formation process at high temperatures, etc.

An option of magnesio-thermal method was made on the basis of literature data on the possibility of boron carbide powder production with the nearest stoichiometric composition. The blend thermicity was undoubtedly allowed for. According to the thermodynamic calculations it was necessary to perform reduction at elevated temperatures for normal carrying out of the magnesiothermal process. The briquettes of the reduced blend undergo milling and chemical treatment to wash the products of the reduction reaction off the boron carbide. The basic technological characteristics as well as other matters concerning magnesiothermal boron carbide powder produced were determined in Technical conditions "Boron carbide labelled with boron-10 and boron-11 isotopes".

Further by using the articles from magnesiothermal \(^{10}\text{B}_4\text{C}\) the control rods with a ring geometry for the reactor BN-350 / SR-TC-4 in number, TC - 1 in number/ were fabricated and tested during 250-350 effective days. The SR-TC rod successfully combined the functions of a temperature compensator rod and an emergency shutdown rod. The operation tests passed without any criticism on the construction.

When \(\text{H}_3^{10}\text{BO}_3\) appeared in the range of the institute’s products, a carbothermal method for \(^{10}\text{B}_4\text{C}\) powder production was developed. A method for the blend preparation by the solubility method was also developed. The blend represents a friable powder-like product which subsequently undergoes carbothermal reduction in a tube electric Raman furnace with a graphite heater.

The works are being conducting by using a spontaneous high-temperature synthesis /SHTS/ for the production of elemental boron-10, powder and articles from \(^{10}\text{B}_4\text{C}\), other boron-containing compounds and composites on their basis. This perspective direction of our work is aimed at the development of the energy-saving SHTS technology for the production of boron-containing materials.

Despite the variety of the methods for \(^{10}\text{B}_4\text{C}\) powder production adopted at our institute we currently fabricate the products by means of the synthesis from elements. However some alterations have been made. \(^{10}\text{B}_4\text{C}\) powder synthesis is carried out separately with its subsequent formation under vacuum. This technology was borrowed from Moscow Plant of Polymetals. The synthesis of a preliminarily briquetted blend /\(^{10}\text{B}+\text{C}/ is performed in a vacuum induction furnace. After cooling the briquettes are milled up to the necessary dispersity, and the powder produced is ready for hot pressing.

**PRODUCTION OF \(^{10}\text{B}_4\text{C}\) ARTICLES**

\(^{10}\text{B}_4\text{C}\) powders manufactured by us according to the above technological techniques undergo forming on the plants for vacuum hot pressing. These plants enable to achieve the necessary temperature for forming simultaneously applying press and overall effort up to 40t. The plants are heated with the help of graphite
Resistance heaters consuming considerable amounts of electric power. Graphite is used as a molding tool. Allowing for the high prices of this material and their great consumption it is urgent to prolong the life-time of a graphite molding tool.

The most compactable is the powder from magnesiothermal boron carbide, probably due to the particles shape.

The hot pressed articles are subjected to mechanical treatment which is one of the basic stages of the production. It is provided by using a diamond tool. Thermal treatment is also used for the stress relief and workability improvement. In fig. 1 a pattern for the production of $^{10}$B$_4$C powders and articles is presented.

In 1989 the institute started the production of articles from boron carbide powder to complete the absorbing rods for fast reactor control and protection systems manufactured by Moscow Plant of Polymetals.

**SOME PROPERTIES**

Thus there are various methods for $^{10}$B$_4$C powder production at our institute which differ in the chemical content, dispersity, the particles shape, etc. from one another.

In table 1 the characteristic chemical content of boron carbide powders produced by various methods are presented.

$^a$Total, $^c$total and $^c$free are determined by a chemical analysis. The remaining impurities were determined with the help of an emission-spectral analysis. The B$_4$C phase formation is evaluated by using X-ray structural investigations.

**TABLE 1. CHEMICAL CONTENT OF $^{10}$B$_4$C POWDERS % MAS**

<table>
<thead>
<tr>
<th>Sample designation</th>
<th>B$_{\text{total}}$</th>
<th>C$_{\text{total}}$</th>
<th>C$_{\text{free}}$</th>
<th>Mg</th>
<th>Si</th>
<th>Fe</th>
<th>Ni</th>
<th>Al</th>
<th>Ca</th>
<th>Cu</th>
<th>Cr</th>
<th>$^{10}$B$_4$C enrichment % at</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Magnesiothermal</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>80/1-H</td>
<td>77.60</td>
<td>29.9</td>
<td>0.6</td>
<td>0.030</td>
<td>0.016</td>
<td>0.46</td>
<td>tr.</td>
<td>tr.</td>
<td>tr.</td>
<td>tr.</td>
<td>-</td>
<td>natural</td>
</tr>
<tr>
<td>82/1</td>
<td>77.70</td>
<td>22.6</td>
<td>0.5</td>
<td>0.026</td>
<td>0.110</td>
<td>0.23</td>
<td>0.06</td>
<td>tr.</td>
<td>tr.</td>
<td>tr.</td>
<td>84.5</td>
<td></td>
</tr>
<tr>
<td>83/1</td>
<td>77.75</td>
<td>22.9</td>
<td>0.8</td>
<td>0.038</td>
<td>0.020</td>
<td>0.23</td>
<td>tr.</td>
<td>tr.</td>
<td>0.07</td>
<td>0.006</td>
<td>61.7</td>
<td></td>
</tr>
<tr>
<td>Carbothermal</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>8/K</td>
<td>76.0</td>
<td>23.4</td>
<td>0.6</td>
<td>0.040</td>
<td>0.018</td>
<td>0.25</td>
<td>0.12</td>
<td>0.03</td>
<td>tr.</td>
<td>0.02</td>
<td>0.15</td>
<td>natural</td>
</tr>
<tr>
<td>56/1</td>
<td>76.22</td>
<td>22.3</td>
<td>0.5</td>
<td>0.035</td>
<td>0.030</td>
<td>0.35</td>
<td>0.03</td>
<td>0.05</td>
<td>tr.</td>
<td>0.001</td>
<td>0.03</td>
<td>82.6</td>
</tr>
<tr>
<td>8/K-56-1</td>
<td>76.04</td>
<td>22.4</td>
<td>0.7</td>
<td>0.035</td>
<td>0.04</td>
<td>0.40</td>
<td>0.04</td>
<td>0.06</td>
<td>0.03</td>
<td>0.002</td>
<td>0.06</td>
<td>60.0</td>
</tr>
<tr>
<td>Synthesized</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>CK5-234</td>
<td>76.98</td>
<td>22.5</td>
<td>0.35</td>
<td>0.012</td>
<td>0.018</td>
<td>0.04</td>
<td>tr.</td>
<td>tr.</td>
<td>tr.</td>
<td>0.01</td>
<td>-</td>
<td>natural</td>
</tr>
<tr>
<td>CK5-235</td>
<td>76.45</td>
<td>21.6</td>
<td>0.45</td>
<td>0.003</td>
<td>0.010</td>
<td>0.01</td>
<td>tr.</td>
<td>tr.</td>
<td>0.03</td>
<td>tr.</td>
<td>59.7</td>
<td></td>
</tr>
<tr>
<td>CK5-270</td>
<td>76.8</td>
<td>21.8</td>
<td>0.25</td>
<td>0.003</td>
<td>0.010</td>
<td>0.6</td>
<td>0.2</td>
<td>tr.</td>
<td>tr.</td>
<td>0.18</td>
<td>80.0</td>
<td></td>
</tr>
</tbody>
</table>
By means of the SHTS we managed to produce elemental boron with the base material's content 98-99% mas with impurities: K-0.3% mas and Mg-0.1% mas, boron carbide powder with the base material's content 98-99% mas at $^7\text{B}$ total 77% mas and $^6\text{B}$ total 22% mas.

The powder from magnesiothermal boron carbide differs in the most profitable shape-spherical. The average dispersity of the powder when using such a method for the production is 1.0µm. The carbothermal boron carbide powder is characterized by a needle-like /elongated/ shape while the synthesized one has a faceted shape near to equiaxial. In both cases the average dispersity of the powders increases more than up to 55µm.

WORKS IN THE NEAREST FUTURE

The scientific and production experience gained on the manufacture of powder and articles from $^{10}\text{B}_4\text{C}$ allows to come to the conclusion on advisability of fulfilling our future work in the two basic directions — intensification of the technological processes and improvement of the economical index of the production.

To execute the first direction it is necessary to mechanize and automatize the production process for improvement of the quality and reliability of the products. Particularly: control of $^{10}\text{B}_4\text{C}$ molding powder granulometric composition; working out of the technological conditions of hot pressing under vacuum and thermal treatment with the purpose of producing an equilibrium structure in the articles free from internal stresses; the study on the mechanism of appearance and behaviour of these stresses during the formation and thermal treatment of the articles, etc.; an increase in the output and mechanization of certain technological operations /forming of the articles, fabrication of a molding tool, assembling of molds, etc./.

To execute the second direction the following problems are of great importance: the development and introduction into the production permitting its functionation at relatively small expenses of power resources, a molding tool and other auxiliary materials. In this connection a wide use of high-frequency facilities, deepening of the investigations on the SHTS, etc. are of great interest. The SHTS is distinguished as well by the possibility of boron carbide alloying with titanium, zirconium and aluminium if necessary.

OUR PERSPECTIVE

It is known that boron carbide possesses unique properties for application in various fields of the industry and especially for nuclear engineering. It seems quite resistivc to use this material in separate constructions of nuclear reactors which differ in function from one another using the attractive properties of $^{10}\text{B}$ and $^{11}\text{B}$ stable isotopes. As an example such important parts of fast reactors can be named as heat-releasing elements, control and emergency shutdown rods, a side water wall /a reflector/, steel for a reactor body, etc. Apparently the usage of $^{10}\text{B}_4\text{C}$ in control elements does not settle the matter. Widening and deepening the knowledge of boron carbide will promote the soonest and most effective solution of these problems.
We should try to attain to produce $^{10}$B$_4$C free from impurities, molding powder with an optimal granulometric composition with maximally uniaxial particle size.

The rich experience with various metallurgical fields shows that we can achieve striking properties of materials by means of alloying. In particular the examples of favourable influence of alloying on serviceability under operating conditions of fast reactors are known relative to $^{10}$B$_4$C articles. There have been such results.

A special place in our future work occupies the problem of regeneration /refabrication/ of the worked off $^{10}$B$_4$C. The matter is attractive, but it is connected with certain difficulties.

We suppose that the basic difficulty lies in the worsening of the refabricated boron carbide quality, as $^{10}$B nucleus burn-up leads to $^{10}$B /$^{11}$B nuclei ratio disturbance, a chemical content alteration - in particular an increase in carbon content, i.e. a change in the B/C ratio and therefore in stoichiometry. To achieve practically the necessary dispersity both in dimension and in the shape seems doubtful. The residual $\gamma$-activity of the refabricated material for its remaking into a high-quality $^{10}$B$_4$C article /in a chemical content, grain size, density, etc./ seems very serious. Economic profit is also controversial because the addition of the corresponding amount of fresh $^{10}$B into the fabricated material is necessary. If the $\gamma$-activity of the samples of the refabricated $^{10}$B$_4$C is acceptable we are ready to participate in such a work with the purpose of high-quality articles production.

<table>
<thead>
<tr>
<th>Technologic pattern of powders and articles production from $^{10}$B and $^{10}$B$_4$C</th>
</tr>
</thead>
<tbody>
<tr>
<td>Separation of $^{10}$BF$_3$-$^{11}$BF$_3$ Isotopes</td>
</tr>
<tr>
<td>Hydrolysis of $^{10}$BF$_3$</td>
</tr>
<tr>
<td>Synthesis of salt $^{10}$BF$_4$+KF $\rightarrow$ K$^{10}$BF$_4$</td>
</tr>
<tr>
<td>Reduction of KBF$_4$+Mg+C</td>
</tr>
<tr>
<td>Evaporation of H$_3$BF$_3$</td>
</tr>
<tr>
<td>Electrolysis of molten salts $^{10}$KB$F$$_4$+KF</td>
</tr>
<tr>
<td>Chemical treatment and refining, 1800°C</td>
</tr>
<tr>
<td>Reduction of H$_3$BF$_3$+7C</td>
</tr>
<tr>
<td>Production of elemental boron $^{10}$B$_{amorph}$</td>
</tr>
<tr>
<td>Hot pressing $^{10}$B$_{amorph}$, 2150°C</td>
</tr>
<tr>
<td>Synthesis 1850°C $^{10}$SB+C $^{10}$B$_3$C powder</td>
</tr>
<tr>
<td>Diamond treatment, annealing</td>
</tr>
<tr>
<td>$^{10}$B$_3$C articles</td>
</tr>
<tr>
<td>$^{10}$B$_3$C powder 2150°C</td>
</tr>
<tr>
<td>Hot pressing $^{10}$B$_3$C articles</td>
</tr>
<tr>
<td>Diamond treatment, annealing</td>
</tr>
<tr>
<td>$^{10}$B$_3$C articles</td>
</tr>
<tr>
<td>$^{10}$B$_4$C articles</td>
</tr>
</tbody>
</table>

229
DIVING-BELL AND DOUBLE-VENTED B₄C CONTROL ROD PIN

LI SHIKUN, Z. CHANGSHAN, XU YINGXIAN
CIAE, China

Presented by R. Voznesenski

Abstract

In this paper a conceptual design of vented device for absorber pins with sodium and gas bonding is described. A related analytical calculations and experimental programme are presented.

I. Introduction

For fast reactors, up to the present only B₄C is extensively used as an actual and potential neutron absorber of the control rod pins. But the awkward problems accompanied with the large yields of helium and lithium due to the burnup of boron-10 (3.72 STD cc helium per 10²₀ atomic burnup of boron-10) are the significant B₄C pellet swelling and helium release. In order to attain high burnup and long-life pin, the B₄C pellet swelling can be accommodated by making use of wet pin, and the helium release can best be rid of by means of vented device. Therefore, vented B₄C pin is known as an accepted orientation. This orientation is pursued in China from the very beginning of the China Experiment Fast Reactor (CEFR).

A lot of reports on the conception of vented devices can be seen in the literatures. In this paper, the diving-bell and double-vented device (DBDVD) which can be applied to wet pin (the filler of the pellet/cladding gap is sodium) as well as dry pin (the filler of the pellet/cladding gap is helium) is researched emphatically on the basic theoretical models and mechanisms, for the referential detail information or theoretical/experimental model of the DBDVD are nowhere to be found in the public literatures. The vent model, when the pin is in work, is set up basing on the phenomenastic hydraulic theory of surface tension-capillarity and gas bubble growth/detachment from the vented end surface of pore immersing in flowing liquid; and the hermetical model, when the pin is out-of-work, is established basing on the phenomenological models of the solubility/diffusibility of inert gases in high temperature liquid metals (here being helium and sodium) and the hydraulic fluctuation of coolant sodium. Then the conceptual design of vented device and the test rig for the hermetical characteristic demonstration of the bell are developed.

I. Vented Phenomenalized Model

In our conceptual structure (see section IV), the helium released into the bell when control rod pin is in work can only be discharged by the vented pore in individual bubbles eventually. The phenomenalized model governing this process may be described as follows.
1) Surface Tension-Capillarity Phenomena — The Necessary Condition for The Initial Bubble Formation

According to the phenomena of surface tension, it is well known that for a bubble with radius \( r_1 \) in the liquid (or the same, droplet in the steam), the forces' balance equation at the interface is

\[
4\pi r_1^2 (p_1 - p_2) = -\frac{d}{dr_1} (4\pi r_1^2 \sigma)
\]

or

\[
p_1 - p_2 = \frac{2\sigma}{r_1}
\] (1)

where \( p_1, p_2 \) are inner pressure of bubble (or droplet) and the outer pressure of liquid (or steam) respectively; \( \sigma \) is the surface tension of the liquid.

For a circular capillary with radius \( r_1 \) (or diameter \( d_1 \)), the free interface in static state between two immiscible fluid media within the capillary (such as helium and sodium) is a spherical meniscus with curvature radius \( r_1 / \cos \theta \), and the balance equation relative to the forces at the interface becomes

\[
\left( p_B - p_1 \right) \left( r_1 / \cos \theta \right)^2 = \frac{\sigma}{d \left( r_1 / \cos \theta \right)} \left( r_1 / \cos \theta \right)^2
\]

or

\[
p_B - p_1 = \frac{2\sigma \cos \theta}{r_1}
\] (2)

where \( p_B, p_1 \) are gas and sodium static pressures, \( \theta \) is the wet angle of the liquid to the wall in capillary, an awkward and delicate parameter, which is very difficult to be measured and used.

If the meniscus is not a spherical, this equation is written as

\[
p_B - p_1 = \left( 1 / r_1 + 1 / r_2 \right) \sigma \cos \theta
\] (2')

with \( r_1, r_2 \) being the two principal radii of curvature of the meniscus. When vent condition is reached, the bubble is formed suddenly on the vented end surface of the capillary with initial diameter \( d_0 > d_1 \) in general. So, comparing eqs. (1) and (2), we can understand easily that the necessary condition (the minimum pressure difference) for the bubble formation is

\[
p_B - p_1 > \frac{2\sigma}{r_1}
\] (3)

This condition is independent of the wet angle \( \theta \) obviously.

2) The Bubble Growth — Dynamic Balance of The Forces

For this phase of bubble growth, the phenomenallized model shown in Fig. 1 is used.
a) Kinetic Energy (Expansive Work) \( K \) and Power \( K' \) of the Bubble

In light of the definition of kinetic energy and the law of momentum conservation as well as noting the conception of virtual mass \( \text{(a)} \) \( M = 1.5M_0 \) when \( d_b \gg d_i \), \( M = 1.688M_0 \) when \( d_b \ll d_i \) that can only be encountered while the bubble jet is in very high frequency, the following equations can be written from Fig 1, they are

\[
K = (3 \pi / 32) \rho_1 F_1 d_b^2 d_b'
\]

\[
K' = (3 \pi / 16) \rho_1 F_1 d_b^2 d_b' (d_b d_b' + 1.5d_b' d_b^2)
\]

\[
K' = (\pi / 2) (p_0 - p_1) [1 - F_2 (d_b')] d_b^2 d_b'
\]

where \( \rho_1 \) is the density of sodium; \( d_b = \frac{d}{dt} (d_b) \), \( d_b' = \frac{d'^2}{dt^2} (d_b) \); \( F_1 \) is the modification for non-fully free motion of the bubble, \( F_2 < 1 \); \( p_0 \) is the pressure on the outer interface of the bubble; \( F_2 (d_b') = (3/4) (d_b')^{-2} \tan (0.5 \sin^{-1} \left((d_b')^{-1})\right)) (1 - (1/3) (d_b')^{-1} \tan (0.5 \sin^{-1} (d_b')^{-1})) \) with \( d_b = d_b/d_0 \) and \( d_0 > d_i \).

From eqs. (5) and (6), we have

\[
p_0 = p_1 + 0.372 \rho_1 F_1 [1 - F_2 (d_b')]^{-1} (d_b d_b' + 1.5d_b' d_b^2)
\]

while \( p_b = p_0 + \frac{4 \sigma}{d_b} \), \( p_b \) is the pressure within the bubble (or the pressure on inner interface of bubble with pressure field within the bubble being neglected).

b) The Relationship for Dynamic Balance of The Forces

At last, the dynamic balance equation of the forces is

\[
p_a = p_1 + 0.372 \rho_1 F_1 [1 - F_2 (d_b')]^{-1} (d_b d_b' + 1.5d_b' d_b^2) + 4 \sigma / d_b + \sum \Delta p
\]

where \( \sum \Delta p \) is the total pressure loss for the gas flow, and the following dynamic relation is satisfied:

\[
p_a > p_b > p_o > p_1
\]

Equations (8) and (9) are in accordance with equation (3).

3) Bubble Detachment —— Critical Bubble Diameter \( d_{bc} \)

Here the diffusion loss is neglected. Then a dynamic bubble just previous to detach the wall encounters a variety of forces including surface tension force, buoyancy force, drag force, drag force due to the wake of the previous bubble, momentum flux of
the gas phase, added mass inertial force\(^{(a)}\). The treatment of balance relation for these forces is more complication, but in our interest range, merely the balance of the former three terms is approximately used, i.e.

\[
\begin{align*}
[1 - F_2 (d\xi_c)] d\xi_c + (0.75 / g) C_d v^2 (1 - F_3 (d\xi_c)) d\xi_c - (\delta / \rho \cdot g) d_c &= 0 \\
\end{align*}
\]

where \(F_2 (d\xi_c) = F_2 (d_c)\) but \(d\xi_c = d_c / d_c\) is substituted for \(d_c\). \(g = 9.81 (m/s^2)\), \(C_d\) is viscous drag coefficient equal to \(24 / Re\) \((Re < 0.4)\), \(18.4 / Re^{0.6}\) \((0.4 < Re < 500)\) and \(0.44\) \((Re > 500)\). \(Re = \nu d_{bc} / \nu\) and \(\nu\) are velocity and viscosity of sodium respectively, \(F_3 (d\xi_c) = 0.5 (d\xi_c)^{-1} \tan(0.5 \sin^{-1} ((d\xi_c)^{-1}))\) \(d_{bc}\) is defined as critical bubble diameter for detachment from wall surface.

II. Analysis of Hermetical Characteristics — Model of Helium Leakage / Sodium Permeability.

The vital function of double vented device is hermetization of the pore through a sealing sodium level in the bell while helium source is absent. Helium / sodium reversal leakage is caused by solubility / diffusibility of helium in sodium and the hydraulic fluctuation of sodium coolant.

1) Solubility of Helium in High Temperature Sodium

There are lots of experimental and theoretical researchs on the Henry's law for the helium in sodium, but only the typical three are cited here.

a) The Expressions of Henry's law Determined by Experiment

Up to present, the most accepted expression of Henry's law for helium in sodium is\(^{(a)}\)

\[
\ln K_H = 4.247 - 6523 T^{-1} \tag{11}
\]

in which \(K_H = (N m^{-2})^{-1} = c_0 / \rho_{H_2}\). \(c_0\) is the solubility of helium in sodium, \(\rho_{H_2}\) is partial pressure of helium in gas phase, \(T = 605K - 850K\)

The other set of data comes from \([12]\) and is expressed as

\[
\log K_H [mol mol^{-1} atm^{-1}] = -2.34 - 35957 T^{-1} \tag{12}
\]

where \(T = 573 - 873K\)

The agreement (particularly at high temperature, see table 1) between the determinations of these two sets of data is surprisingly good in view of the disparate techniques adopted.

b) Phenomenalized Theoretical Model

Henry's law can also be derived from statistical thermodynamics. It is well known that the vital procedure for the method of thermodynamics is always to find the Helmholtz free energy \(F\) first. To find \(K_H\) makes no exception. One of the results is\(^{(12)}\).
\[ K_H = (V_r / RT) \exp(\Delta F_{er} / RT) \]  \hspace{1cm} (13)

By the definition of free energy and further by the assumption that the entire process of dissolution consist of three individual steps: digging hole, diving gas atom into the hole, and raising the gas atom to oscitation as liquid atom in the solution, we have

\[ \Delta F_{er} = \Delta U_{er} - T \Delta S_{er} \]

\[ = U_{er.c} + U_{er.w} + U_{er.a} - T S_{er.a} \]

and

\[ K_H = (V_r / RT) \exp[- (U_{er.c} + U_{er.w} + U_{er.a}) / RT + T S_{er.a}] \]  \hspace{1cm} (14)

in which \( V_r \) is molar volume of sodium in solution; \( \Delta F_{er} \) is the molar Helmholtz free energy of helium in solution, the reversible molar work required to introduce the atoms of gas into the solution of concentration \( c_0 \) (\( c_0 \) is so small, the solution can be regarded as pure sodium); \( U_{er.c} \) is the molar internal energy required for forming a hole in the liquid of the size of the gas atom, its value is plus and is the largest one in magnitude, the accompanied entropy is known as a very little plus value but it is very difficult to calculate, so is neglected; \( U_{er.w} \) is the molar inner energy required to raise the "quasi-rigid sphere" introduced to necessary potential (i.e. introduced to the hole), its value is minus and intermediate in magnitude; \( U_{er.a} \) is the molar internal oscillation energy of the gas atom after introduced into the hole, its value is plus and intermediate in magnitude; \( \Delta S_{er}=S_{er.a} \) is the molar entropy of oscillation of the gas in solution, the only entropy being considered, it is a significant plus value. \( V_r, U_{er.c}, U_{er.w}, U_{er.a}, S_{er.a} \), hence \( \Delta F_{er} \) are temperature-dependent. This model is more easily understood than others\(^1\), but not yet very precise in its parameters, there are a certain margins to be improved.

The quantity comparison between these three models is shown in table 1.

<table>
<thead>
<tr>
<th>T (K)</th>
<th>623</th>
<th>673</th>
<th>723</th>
<th>773</th>
<th>823</th>
<th>873</th>
</tr>
</thead>
<tbody>
<tr>
<td>( K_H = c_0 / p_{\text{hel}} )</td>
<td>eq.(11)</td>
<td>1.96 \times 10^{-8}</td>
<td>4.27 \times 10^{-9}</td>
<td>8.35 \times 10^{-9}</td>
<td>1.50 \times 10^{-7}</td>
<td>2.50 \times 10^{-7}</td>
</tr>
<tr>
<td>( p_{\text{hel}} ) (atm)</td>
<td>eq.(12)</td>
<td>7.75 \times 10^{-9}</td>
<td>2.08 \times 10^{-9}</td>
<td>4.87 \times 10^{-9}</td>
<td>1.02 \times 10^{-7}</td>
<td>1.96 \times 10^{-7}</td>
</tr>
<tr>
<td></td>
<td>eq.(14)</td>
<td>2.48 \times 10^{-8}</td>
<td></td>
<td></td>
<td>1.50 \times 10^{-7}</td>
<td></td>
</tr>
</tbody>
</table>

2) Diffusion Coefficient of Helium in High Temperature Sodium

235
There are several different arguments on the diffusibility in liquid solutions, but the rigorous and generalized model is difficult to find. Here two typical models are quoted.

a) The Classical Model —— A Hydraulic Model

With potential flow as its prerequisite, this model is first derived by Stokes—Einstein for the movement of a not too small spherical particle through the continuum (liquid).

- Stokes — Einstein Formation

When a spherical particle moves in fluid by unit force, the original irregular Brownian motion becomes an oriented one named as mobility. This mobility is expressed as

\[ u_i = \frac{1}{A \times r \eta} \]  

in which \( A = 6 \), \( r \) is the particle radius, \( \eta \) is the fluid viscosity.

- Use in The Regular Diffusion Field of Solutions

When diffusion law is applied to the field of chemical potential/concentration of the solution, the diffusion coefficient \( D \) in one dimensional \( X \) can be derived as

\[
\begin{align*}
\mu &= \mu^0 + kT \ln c \\
u &= u_i (-d\mu/dX) = -u_i (kT/c) (dc/dX) \\
J &= -D (dc/dX) = cu_i = -u_i kT (dc/dX)
\end{align*}
\]

At last we obtain

\[ D = \frac{kT}{A \times r \eta} \]  

where \( A \) is approximately constant (\( A = 1.4—6 \) dependent on the relative size of the particle to the continuum's one and the slip degree or the chemical activity (or fugacity) for the diffusion, etc); \( r \) is particle radius; \( \eta \) is viscosity of solvent.

\( k = 1.381 \times 10^{-23} \text{J/K} \), \( \mu \) (and \( \mu^0 \)) is chemical potential of the particle in solution.

Eqs. (15) and (16) (that we may call as the law of first-order power of temperature for diffusion) are extensively used in many fields with fine accuracy. Some data of diffusion/self-diffusion for sodium, lead and sodium/lead system, which may be useful in the technology of fast reactors, are available \( ^{14, 17} \), we can find that the eq. (17) is applicable in engineering sense even for the self-diffusion of sodium in which the pair potential is broad and shallow. But for the component with intense volatility (the intensely spontaneous and irreversible process) in the solution such as helium in sodium, its diffusion (the particle velocity) is evidently so fast, that this model of potential flow should not be used.

b) The Relationship From [12]

This is an experimental relationship having no parallel anywhere and perhaps
agreeable to the thermodynamic principle, so it is the one that can only be used as an expedient at present, and is approximately expressed as

\[ \log D = 3.77 - 3051 \times T^{-1} \]  

(17)

Compare and analyze eqs. (12) through (18) as well the diffusions of the components of solids and solutions, one can learn a good deal of interest instructions such as the analogy between the vacancy theory of solids and the hole theory at liquids, the irreversible / reversible relation between dissolution and diffusion, etc. Fig. 2 shows the significant difference between the processes of dissolution and degassification of the helium / sodium system as well the Argon / sodium system imitated from the report [12].

3) Hydraulic Fluctuation of Sodium Coolant

The hydraulic fluctuation (V or p) within a channel of the core assembly, which in turn is feedback to the pore and cause sodium flow in it, may be caused by the periodic suction of a venduri flow, by the wake of a vortex, by the impulse of an impinging jet (such jet in turn may be caused by pumping pulse or vortex formation / shedding), or by the motion of control-rod-self, etc and their secondary effects. The periodic suction of the venduri flow comes from random displacement or vibration of the control rod or rod pin. The impulse of the vortex formation / shedding and wake is the characteristics of the turbulent flow, and the maximal one with \( \Delta V \) equal to the velocity of the largest edding and \( f \) equaling to \( V_0/1 \) (\( V_0 \) is the average velocity of the coolant sodium flow, \( I \) corresponds to the characteristic dimension of this edding). The pumping pulse comes from the rotation of pump blades with \( f \) equal to \( n \times \text{rpm} \), \( n \) being the number of blades.

Letting \( V = V_0 + V_3 \), \( p = p_0 + p_3 \), the Navier-Stokes equation and the continuity equation can be written into two counterparts, one presents the average flow, the other presents the fluctuation (perturbation) flow. The average flow is

\[ (V_0 \text{grad}) V_0 = -\text{grad}(p_0 / \rho_0) + \nu \Delta V_0 \]

\[ \text{div} V_0 = 0 \]  

(18)
and the fluctuation flow is
\[
\left( \frac{dV_a}{dt} \right) + (V_0 \text{grad}) V_a + (V_1 \text{grad}) V_0 = -\text{grad} \left( \frac{p_1}{p_2} \right) + \nu \Delta V_a
\]
\[
div V_a = 0
\]

(19)

The boundary condition is that \( V_0 \) and \( V_a \) vanish on fixed solid surface. Evidently, \( V_a, p_1 \) in eq. (19) cannot be resolved because they have complex spectrums, and the method of noise analysis or measurement must be used. In our question, what we most concern is the effective amplitude and the frequency can be regardless. This effective amplitude \( |V_a| \) is expressed as
\[
|V_a| = \sigma = \sqrt{(V - V_0)^2} = \sqrt{(V^2 - V_0^2)}
\]

(20)
i.e. \( |V_a| \) is the standard deviation or rms of the \( V_0 \). Then the velocity \( u_s \) in the pore is resolved from following equation:
\[
u L^2 = \frac{42.67}{d^2} u_s = 0.67 V_x^2
\]

(21)

where \( \nu \) is dynamic viscosity of sodium, \( L \) is capillary length.

4) Helium loss Rate in Plenum (Bell) by Diffusion and Hydraulic Fluctuation

This can be found by resolving the continuity equation of the diffusion component of the solution. This equation is (here the reversal diffusion of argon being neglected conservatively for the K_H and D of argon in sodium being much less than those of helium in sodium)

\[
\frac{d}{dt} \int \rho \cdot c dv = \int \rho \cdot c u_s df - \int J df
\]

or

\[
\frac{d}{dt} (\rho \cdot c) = -\text{div}(\rho \cdot c u_s) - \text{div} J
\]

where \( \rho \) is sodium density in at./cc, \( J \) is diffusion flux at./cm^2·s).

In our question, following conditions are satisfied:
- Incompressive fluid, in which \( \frac{d}{dt} \rho + \text{div}(\rho \cdot u_s) = 0 \).
- \( J = -\rho \cdot D[\text{grad}c + k_c \text{grad} - k_p \text{grad}p] = -\rho \cdot D \text{grad}c \).
- One dimension and only steady state being considered.
- \( u_s = \text{const.} \) and having the \( J \) direction be presumed conservatively, and letting \( D(c) = \text{const.} \).

Then, this equation and its solutions can be written easily as:
\[ u \frac{dc}{dx} + D \frac{d^2c}{dx^2} = 0 \] (22)

\( x = 0, \ c = c_0; \ x = L_{err} \ (L_{err} < L), \ c = 0 \)

\[ c = (1 - \exp(-u_t (L_{err} - x) / D)) \{ 1 - \exp(-u_t L_{err} / D) \} c_0 \] (23)

\[ J = [1 - \exp(-u_t L_{err} / D)]^{-1} \rho \ u_t c_0 \] \[ \text{[at (He) \cdot cm}^{-2} \cdot \text{s}^{-1}] \] (24)

\[ Q = \pi d J / 4 = A \rho \] \[ \text{[at (He) \cdot s}^{-1}] \] (25)

in which \( L \) is length of the pore.

It can be seen easily that when \( D \to 0 \), \( Q \to \rho \ u_t A c_0 \), the loss rate caused by pure hydraulic fluctuation; when \( u_t \to 0 \), \( Q \to D \rho \ A c_0 / L_{err} \), the loss rate caused by pure diffusion of \( c \).

IV. Conceptual Design

1) Design Parameters

As a conceptual design, following parameters are considered: cladding diameters \( D_0 / D_1 = 15 / 13 \text{ (mm)} \), diameter of \( \text{B}_4\text{C} \) pellet \( D = 12.3 \text{ mm} \), pellet stack 51cm, average burnup of \( \text{U}^{235} \) \( 4 \times 10^{21} \text{cm}^{-3} \), volumetric swelling rate of \( \text{B}_4\text{C} \) 1.5% per \( 10^{21} \text{cm}^{-3} \) burnup of \( \text{U}^{235} \), operation temperature 800K, minimal shutdown temperature 500K, pressure 0.2MPa / 0.14MPa, pins pitch 16-16.2mm (with wire wraps as spacer).

2) Vent Calculation

In this area, only the detached bubble diameter is concerned. So, the calculation is carried out according to eq. (11). The results are listed in Table 2. The dissolution is

<table>
<thead>
<tr>
<th>( \nu ) (m/s)</th>
<th>( d_i = 1.6 \mu m )</th>
<th>( d_i = 48.1 \mu m )</th>
<th>( d_i = 0.1 \text{mm} )</th>
<th>( d_i = 0.2 \text{mm} )</th>
<th>( d_i = 0.3 \text{mm} )</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>( d_{be} ) mm</td>
<td>( f ) ( s^{-1} )</td>
<td>( d_{be} ) mm</td>
<td>( f ) ( s^{-1} )</td>
<td>( d_{be} ) mm</td>
</tr>
<tr>
<td>0</td>
<td>&gt;1.16</td>
<td>0.023</td>
<td>&gt;1.76</td>
<td>0.057</td>
<td>&gt;2.25</td>
</tr>
<tr>
<td>0.10</td>
<td>&gt;1.03</td>
<td>0.3</td>
<td>&gt;1.65</td>
<td>0.07</td>
<td>&gt;2.14</td>
</tr>
<tr>
<td>0.25</td>
<td>&gt;0.74</td>
<td>0.8</td>
<td>&gt;1.28</td>
<td>0.15</td>
<td>&gt;1.72</td>
</tr>
<tr>
<td>0.50</td>
<td>&gt;0.42</td>
<td>4.3</td>
<td>&gt;0.77</td>
<td>0.67</td>
<td>&gt;1.1</td>
</tr>
<tr>
<td>0.75</td>
<td>/</td>
<td>/</td>
<td>&gt;0.53</td>
<td>2.2</td>
<td>&gt;0.76</td>
</tr>
<tr>
<td>1.00</td>
<td>/</td>
<td>/</td>
<td>&gt;0.39</td>
<td>5.4</td>
<td>&gt;0.58</td>
</tr>
<tr>
<td>1.25</td>
<td>/</td>
<td>/</td>
<td>&gt;0.32</td>
<td>9.7</td>
<td>&gt;0.46</td>
</tr>
</tbody>
</table>
conservately neglected in the calculation of bubble diameter for this process is relatively slower than the total process of bubble formation-growth-detachment in general. The maximal bubble diameter that the subchannel of the control rod can accommodate but can not be in contact with neighbour pins is determined by the pitch of the pins. So, the \( d_{bc} \) must be less than 3.25 mm, and this size is the design limitation.

3) Evaluation of Hermetical Characteristics

Using equations (23) to (25) and the relevant parameters, the hermetical function (the helium loss rate by diffusion/hydraulic fluctuation while source is absent) can be executed. The results are listed in table 3.

<table>
<thead>
<tr>
<th>( T ) (K)</th>
<th>533</th>
<th>623</th>
<th>673</th>
<th>723</th>
<th>773</th>
<th>800</th>
</tr>
</thead>
<tbody>
<tr>
<td>( \rho P_{He} ) at ( ml^{-1} )</td>
<td>2.62 \times 10^{19}</td>
<td>2.33 \times 10^{19}</td>
<td>2.15 \times 10^{19}</td>
<td>2.00 \times 10^{19}</td>
<td>1.875 \times 10^{19}</td>
<td>1.811 \times 10^{19}</td>
</tr>
<tr>
<td>( \rho P_{He} ) at ( \cdot ml^{-1} )</td>
<td>2.30 \times 10^{22}</td>
<td>2.245 \times 10^{22}</td>
<td>2.20 \times 10^{22}</td>
<td>2.16 \times 10^{22}</td>
<td>2.11 \times 10^{22}</td>
<td>2.08 \times 10^{22}</td>
</tr>
<tr>
<td>( c_0 ) mol \cdot mol^{-1}</td>
<td>1.03 \times 10^{-6}</td>
<td>3.87 \times 10^{-6}</td>
<td>8.44 \times 10^{-6}</td>
<td>1.65 \times 10^{-7}</td>
<td>2.97 \times 10^{-7}</td>
<td>3.93 \times 10^{-7}</td>
</tr>
<tr>
<td>( u_0 ) cm \cdot s^{-1}</td>
<td>2</td>
<td>2</td>
<td>2</td>
<td>2</td>
<td>2</td>
<td>2</td>
</tr>
<tr>
<td>( Q ) at ( s^{-1} )</td>
<td>3.45 \times 10^{-11}</td>
<td>1.40 \times 10^{-11}</td>
<td>4.60 \times 10^{-12}</td>
<td>1.38 \times 10^{-12}</td>
<td>4.10 \times 10^{-13}</td>
<td>7.10 \times 10^{-13}</td>
</tr>
<tr>
<td>( Q' ) at ( s^{-1} )</td>
<td>3.25 \times 10^{-11}</td>
<td>1.23 \times 10^{-12}</td>
<td>2.63 \times 10^{-12}</td>
<td>5.03 \times 10^{-13}</td>
<td>8.90 \times 10^{-13}</td>
<td>1.20 \times 10^{-13}</td>
</tr>
<tr>
<td>( Q'' ) at ( s^{-1} )</td>
<td>4.00 \times 10^{-10}</td>
<td>5.70 \times 10^{-11}</td>
<td>3.00 \times 10^{-12}</td>
<td>1.10 \times 10^{-12}</td>
<td>3.65 \times 10^{-13}</td>
<td>5.52 \times 10^{-13}</td>
</tr>
<tr>
<td>( t ) day \cdot ml^{-1}</td>
<td>/</td>
<td>197</td>
<td>55.5</td>
<td>16.8</td>
<td>5.29</td>
<td>2.95</td>
</tr>
<tr>
<td>( t' ) day \cdot ml^{-1}</td>
<td>/</td>
<td>/</td>
<td>94.7</td>
<td>45.9</td>
<td>24.4</td>
<td>17.5</td>
</tr>
<tr>
<td>( t'' ) day \cdot ml^{-1}</td>
<td>/</td>
<td>/</td>
<td>84</td>
<td>21</td>
<td>5.94</td>
<td>3.21</td>
</tr>
</tbody>
</table>

* \( Q \) , \( t \) corresponds to diffusion/hydraulic effect, \( Q' \) , \( t' \) corresponds to pure hydraulic effect, \( Q'' \) , \( t'' \) corresponds to pure diffusion effect

From this table one can see that to protect the helium from leakage in high sodium temperature is an awkward matter. In this case, \( Q \approx A / L_{e_{rr}} \), the more effective method to decrease leakage is the reduction of the \( d_{bc} \).

4) Conceptual Design of The Diving-Bell

The sketchy structure of the DBDVB is shown in Fig 3. From this Fig, it can be seen that the gas plenum consists of two organic parts with a capillary link between them. The filters in the bell is used to protect the particle from escape and to calm the sodium level in the plenum for protect the capillary from sodium sneaking in. Before commissioning, the plenum was filled with argon in advance and the vented pore was plugged with eutectic alloy. After commissioning, the eutectic alloy melt out automatically and the device is put into operation.
dutiful design, the capillary of the plenum must be protected from the surreptitious intrusion by sodium of inner or outer bell. Therefore, for the volume \( V_i \) of the inner bell, the effective accumulated swelling of the \( \text{B}_4\text{C} \) pellet stack and the difference of the thermal expansions of the \( \text{B}_4\text{C}/\text{Na}/\text{S.S} \) (cladding) system must be determined, besides, a necessary allowance must be stipulated; for the volume \( V_o \) of the outer bell, in addition to the volume \( V_{o1} \) of the accumulated helium leakage (or coolant sodium sneak-in) during the designed duration with no source present, a design margin \( V_{o2} \) for the helium volume shrinkage due to maximal / minimal temperature / pressure differences between the regimes of rate and reload operation must be stipulated.

Basing on these requirements, for our conceptual design, the following relations of the organic plenum components should be satisfied:

\[
V_i = V_i + V_o
= V_i + V_{o1} + V_{o2} \tag{26}
\]

\[
V_{o2} > 0.35 V_i \tag{27}
\]

Here the referential values are arbitrarily taken: \( V_i = 5 \text{ cc} \), \( V_{o1} = 4 \text{ cc} \), \( V_{o2} = 7.5 \text{ cc} \), \( V_i = 16.5 \text{ cc} \). The Sketchy structure of DBDVD is shown in Fig. 3.

V. The Hermetical Test Rig

As mentioned above, the referential detail information or theoretical/experimental model of the DBDVD are nowhere to be found in the public literatures, all of the arguments in this paper can not but be based merely on the expedient scientific principles. Obviously, there are lots of works to be done to convert this conceptual design into the real one, but the badly necessary affair at present is the demonstration of the hermetical characteristics of the vented pore, especially the diffusion coefficient \( D \) of the helium in sodium solution at our particular conditions. Now, the test rig is designed and fabricated. This rig is shown in Fig. 4 diagrammatically. Here only the main points are illustrated as follows:
Test objects: Verifying the hermetical characteristics of the vented pore, hence the conceptual design, some referential results be pinned hopes on.

Test arrangements: $T = 230-600^\circ C$ (at steady state) with 50-100$^\circ C$ steps, $p = 0.06-0.1$MPa;

Main technologic points:
- The treatments of cleanliness/degassification and sealing degree of the system satisfying test requirements;
- The purification of sodium/Ar satisfying test requirements ($O < 10$ ppm, $H < 10$ ppm, initially);
- $T/p$ of the sodium tank/Ar tank adjustable;
- Several pairs of thermocouple to measure the sodium level in the helium can with a siren to alarm the sodium lever.

The shortcoming of this design which will affect the applicability of the test results and which should be taken into account is that it can not imitate the sodium coolant flow, for this imitation is not yet possessed at present.

![Diagrammatical Rig Scheme for The Hermetical Test](image)

Fig. 4 The Diagrammatical Rig Scheme for The Hermetical Test

This paper and the arguments are prepared by Li shikun, the design of the rig and test is entrusted to Zhou changshan with Xu yingxian participation. We would like to acknowledge Wang Xiaorong for his elaborate editing/ printing and Ju zaixiang for her calculation of table 2.
REFERENCES

1 CEA-CONF-6655, Apr 1983
2 CEFR Primary Design (in Chinese), 1995
3 IAEA Report IWGFR/48, OBNINSK, 7-10, June, 1983
4 HEDL-SA-1995-PP, 1980
8 王竹溪, 热力学 (in Chinese), 1955
9 S C Chuang & W Goldschmidt, J of Basic Engg, Dec 1970, p705-711
10 E Veleckis et al, ANL-7802, 1971
11 H U Borgsteld and Mathews, Applied Chemistry of The Alkali Metals, 1987, ch 8
12 KFK-1166 or EURTRN-785, 1970
14 Sylvan Z Beer, ed, Liquid Metals (chemistry and physics), 1972 p250-1, 514-525
15 Tyrrell, Diffusion and Heat Flow in Liquids, 1961
17 Natrium (in Russian), 1992

243
LIST OF PARTICIPANTS

Alexandrov, I.  
Experimental Machine Building Bureau (OKBM)  
Burnakovski Proyezd 15  
603 603 Nizhny Novgorod - 74,  
Russian Federation

Baldov, A.  
MAEC, NPP BN-350  
466200 Aktau  
Kazakhstan

Bairamashvili, I.  
Institute of Stable Isotopes  
21, Kavtaradze str.  
380086, Tbilisi,  
Georgia

Babu, R.  
Indira Gandhi Centre For Atomic Research  
Department of Atomic Energy  
Kalpakkam 603 102, Tamil Nadu,  
India

Buksha, Yu  
Institute of Physics and Power Engineering, IPPE  
Bondarenko Sq. 1  
Obninsk, Kaluga Region 249020  
Russian Federation

Chernyshov, V.  
Moscow Factory Polymetals  
Kashirshoy Strasse - 49  
Moscow 115409,  
Russian Federation

Edelmann, N.  
Forschungszentrum Karlsruhe  
Postfach 3640  
76021 Karlsruhe,  
Germany

Favet, D.  
FRAMATOME Direction NOVATOME  
10 rue Juliette Récamier, BP 3087  
69456 Lyon Cedex 06  
France

Ivanov, A.  
Institute of Physics and Power Engineering, IPPE  
1 Bondarenko Sq., Obninsk  
Kaluga Region, 249020  
Russian Federation
Kaito, T. O-arai-Engineering Center
Power Reactor and Nuclear Fuel Development
Corporation (PNC)
4002, Narita-cho, O-arai-machi
Higashi-ibaraki-gun, Ibaraki-ken 311-13
Japan

Kryger, B. CEA, Centre d’Etudes de Saclay
AMT/SEMI/LEMA
91191 Gif-sur-Yvette Cedex
France

Matveev, V. Institute of Physics and Power Engineering, IPPE
1 Bondarenko Sq.,
Obninsk, Kaluga Region 249020
Russian Federation

Okada, K. Mitsubishi Heavy Industries, Ltd.
Advanced Reactor Engineering Department
3-1 Minatomirai 3-chome
Nishi-ku, Yokohama 220-84
Japan

Poplavski, V. Institute of Physics and Power Engineering, IPPE
Bondarenko Sq. 1, Obninsk
Kaluga Region 249020
Russian Federation

Ponomarenko, V. Moscow Factory Polymetals
Kahshirskoy Strasse - 49
Moscow 115409
Russian Federation

Rudenko, V. Institute of Physics and Power Engineering, IPPE
Bondarenko Sq. 1
Obninsk, Kaluga Region 249020
Russian Federation

Rogov, V. Experimental machine Building Bureau (OKBM)
Burnakovski Proyezd 15
603 603 Nizhny Novgorod - 74
Russian Federation

Roslyakov, V. NPP BN-600
Beloyarsk 624 051
Russian Federation
Rinejski, A. (Scientific Secretary)  
IAEA, Division of Nuclear Power  
Wagramerstrasse - 15  
P.O. Box 100  
A-1400 Vienna,  
Austria

Risovany, V.  
V.I. Lenin Research Institute of Atomic Reactors (RIAR)  
Dimitrovgrad, 43 35 10  
Russian Federation

Shabalin, A.  
Experimental Machine Building Bureau (OKBM)  
Burnakovski Proyezd 15  
603 603 Nizhny Novgorod - 74  
Russian Federation

Tarasikov, V.  
Institute of Physics and Power Engineering, IPPE  
Bondarenko Sq. 1, Obninsk  
Kaluga Region 249020  
Russian Federation

Truffert, J.  
CEA/DRN/DEX/SDC  
Bat 315, CE-Cadarache  
13108 Saint Paul les Durance  
France

Voznesenski, R.  
Institute of Physics and Power Engineering, IPPE  
Bondarenko Sq. 1, Obninsk  
Kaluga Region 249020  
Russian Federation

Zakharov, A.  
V.I. Lenin Research Institute of Atomic Reactors (RIAR)  
Dimitrovgrad, 433510  
Russian Federation