SPECIALIST MEETING ON NUCLEAR FUEL AND CONTROL RODS: OPERATING EXPERIENCE, DESIGN EVOLUTION AND SAFETY ASPECTS

Madrid, Spain, 5-7 November 1996

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Principal Working Group 1 of the Committe on the Safety Nuclear Installations (CSNI)

and hosted by:

Consejo de Seguridad Nuclear (CSN)
Unidad Eléctrica S.A. (UNESA)
Empresa Nacional del Uranio S.A. (ENUSA)

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article I of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

— to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;

— to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and

— to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994) the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996) and the Republic of Korea (12th December 1996). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its f rst nonEuropean full Member. NEA membership today consists of all OECD Member countries, except New Zealand and Poland. The Commission of the European Communities takes part in the work of the Agency.

The primary objective of the NEA is to promote co-operation among the governments of its participating countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source. This is achieved by:

— *encouraging harmonization of national regulatory polieies and practiees, with partieular referenees to the safety of nuelear installations, proteetion of man against ionising radiation and preservation of the environment, radioaetive waste management, and nuelear third party liability and insuranee:*

— *assessing the eontribution of nuelear power to the overall energy supply by keeping under review the teehnieal and economie aspeets of nuelear power growth and foreeasting demand and supply for the different phases of the nuel ear fuel eyele;*

— *developing exehanges of seientifie and teehnieal information partieularly through partieipation in common services*

— *setting up international research and development programmes and joint undertakings.*

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

The opinions expressed and arguments employed in this publication are the sole responsibility of the author and do not necessarily reflect those of the OECD or of the governments of its Member countries.

Committee on the safety of nuclear installations

The Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA) is an intemational committee made up of senior scientists and engineers. It was set up in 1973 to develop, and co-ordinate the activities of the Nuclear Energy Agency conceming the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster intemational co-operation in nuclear safety among the OECD Member countries.

The CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of the programme of work. It also reviews the state of knowledge on selected topics on nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach intemational consensus on technical issues of common interest. It promotes the co-ordination of work in different Member countries including the establishment of co-operative research projects and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as infommation exchanges, establishment of wo~ing groups, and organisation of conferences and specialist meetings.

The greater part of the CSNI's current programme is concemed with the technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment perfommance, risk assessment, and severe accidents. The Committee also studies the safety of the nuclear fuel cycle, conducts periodic surveys of the reactor safety research programmes and operates an intemational mechanism for exchanging reports on safety related nudear power plant accidents.

In implementing its programme, the CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency conceming the regulation, licensing and inspection of nuclear installations with regard to safety. It also cooperates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

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Foreword

Design and management of nuclear fuel has undergone a strong evolution process during the past years. The increase of the operating cycle length and of the discharge burnup has led to the use of more advanced fuel designs, as well as to the adoption of fuel-efficient operational strategies. The analysis of recent operational experience highlighted a number of issues related to nuclear fuel and control rod events raising concerns about the safety aspects of these new designs and operational strategies.

Considering the situation, the CSNI at its November 1995 meeting, approved a proposal made by the Principal Working Group No. 1 on Operating Experience and Human Factors (PWG1) to Organise a Specialists Meeting on fuel and control rod issues. The meeting was intended to provide a forum for the exchange of information on lessons learned and safety concern related to operating experience with fuel and control rods (degradation, reliability, experience with high burnup fuel, and others).

The meeting took place in Madrid, Spain, in November 1996, at the invitation of the Spanish Authorities, who hosted it and published the Proceedings. Sponsor organisations were the NEA, the Consejo de Seguridad Nuclear (CSN), Unidad Eléctrica Española (UNESA), who hosted the meeting, and Empresa Nacional del Uranio (ENUSA).

The Specialists Meeting was organised by a Programme Committee composed of PWG1 experts from different countries, and representatives of the sponsor organisations. The members of the Committee were the following:

José M. Conde (CSN). Chairman of the Programme Committee
J. P. Clausner (OECD/NEA)
B. Gautier (EDF)
J. Serrano (ENUSA)
J. In de Betou (SKI)
N. Tricot (IPSN)
J. Rosenthal (USNRC)
D. Molina (Iberdrola)
T. Itaki (NUPEC)

These proceedings contain the papers presented at the meeting and an account of the main discussions and conclusions that could be obtained. The opinions presented are those of the speakers and do not necessarily express the official views of countries or international organisations concerned.

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Session Chairman: *D. Molina* (Iberdrola) and *T. Itaki* (NUPEC)

TECHNICAL SUB-SESSION IV.B

"Improvement on Fuel Fabrication and Core Management"

Session Chairman: *J. Rosenthal* (USNRC) and *J. Segarra* (GENUSA)

Preliminary conclusions of the meeting

(As transmitted to the CSNI)

The material presented at the Meeting, and the discussions that followed, have identified the following potential safety concerns:

The operational space in which the fuel is used is being expanded, by increasing the duration of the operation cycle, the discharge burnup or the power density, sometimes without a sound experimental base. As a consequence, the adequate fuel behaviour under those more demanding conditions cannot be guaranteed beforehand.

Some adverse effects have been listed which seem to be related with extended fuel burnup, or perhaps with the increased in-reactor residence time. Some of the effects identified are excessive guide thimble tube distortion, excessive crud accumulation and long-term control rod material behaviour.

Incomplete control rod insertion events have occurred at several PWR plants, involving fuel from at least three different suppliers up to now. Some preliminary root cause analyses have been forwarded, but the information available does not seem to allow for the adoption of a long-term solution at this stage.

The current fuel demonstration programs (Lead Test Assemblies) seem to not be providing the desired insights on the new design's behaviour and performance. A thorough definition of the objectives pursued by each demonstration program should be performed, and the core conditions under which the LTA are operated (power and burnup history and discharge values, temperature,...) should be representative of the conditions will have in future reload cores.

The introduction of "minor" fuel and/or core design changes, and the evolution towards more efficient fuel management and plant operation strategies (including the adoption of mixed cores), have finally resulted in adverse effects in some cases, due to the lack of an in-depth safety analysis of the individual and cumulative consequences of the different changes by the licensees. The need for an enhanced oversight of these processes by the regulators was identified.

The core related issues are becoming more and more plant-specific, due to the flexibility of the fuel and core design and to the wide variety of management strategies available to the licensee. The validity of generic safety demonstrations for an specific case should be carefully revised.

Technical session conclusions by the session chairman

SESSION I

The objective of this first session was to review the operational experience regarding core performance, to try to identify the safety concerns related to fuel and control rods. The operational experience of different countries was presented, basically in terms of the fuel failures detected (leakers) during the last years. In general terms, the experience described was remarkably good when measured in terms of the number of failed fuel rods.

Some presentations referred also the operational experience with control rods, and some material problems were described. This issue, together with the incomplete control rod insertion events, was extensively treated during Technical Session 3 and will be described there.

Some relevant events involving leakers were described, and an analysis of the root cause was provided for those cases. Main causes for the rod failures seem to have been the inadequate chemical conditions of the reactor coolant and debris induced damage. Excessive corrosion of the fuel structure while stored in the spent fuel pool was reported in one case, and some design and manufacturing errors were mentioned.

Most papers highlighted steady improvements in fuel and core components in response to economical pressure and competitive markets. That seems to indicate that the present fuel designs, when subject to normal operating conditions, are adequate for the burnup levels currently in use.

However, such competition also results in extended fuel cycles and higher burnup fuel, and leads to new types of deficiencies attributed to inadequate preservice testing or inadequate evaluation of the component's capability to withstand more demanding service conditions.

Most presentations described the operational experience in detail, but did not address the potential safety concerns that could be raised from that experience. Some concerns were however expressed in a few papers, and in the discussions that followed each presentation:

- Impact of high burnup and longer fuel cycles on fuel and control rods performance: thimble bowing, crud accumulation, long term behaviour of control rod materials,... As a result, an increase of the current burnup limit values has been denied in most countries.
- Adequacy of the lead test assembly programs to demonstrate the correct fuel behaviour during commercial operation.
- Safety analysis of mixed cores. Interfaces between competing organisations: proprietary data availability, quality assurance, thermal hydraulic compatibility,...
- Assessment of the cumulative impact of minor design changes. It was pointed out that the licensees and the regulators should ensure that new design features have been assessed prior to implementation.
- MOX fuel issues. The increasing trend of using MOX fuel was identified as a concern by the French regulatory body.

SESSION II

Seven papers were presented in this session: two on fuel failures, two on the development and performance of debris filters, two related to fuel behaviour under transient conditions and one describing on-site measurement equipment.

Fuel failures were reported in Lovisä 2 and Hamaoka I. The failures in Lovisä 2 were attributed to crud formation and build-up in the spacers, leading to coolant flow reduction, temperature increase, fuel rod vibration and, finally, grid fretting failure.

The fuel failure in Hamaoka I was caused by excessive corrosion, attributed to a combination of an abnormal water chemistry and high corrosion susceptibility of the cladding material.

The papers dealing with debris filters show large improvements on debris induced fuel failure rates when the filters are introduced. As the earlier filter designs did not completely eliminate this type of failures, new improvements have been introduced aimed to reach the 100% protection level. In addition, these new designs should reduce or suppress grid fretting in the bottom part of the fuel assembly.

In the case of the automatical reactor trip during the earthquake, the consequences of the change of the width of the water gaps between the fuel assemblies led to a net reactivity increase, but only in the case of the D-lattices because of the specific enrichment distribution. Specific vibration tests and subsequent 3D dynamics analysis have confirmed these conclusions.

Power ramping performed on commercial fuel rods in the frame of the TRANSRAMP project, indicate that incipient cracks induced by a first ramp decrease the time to failure when a second ramp is experienced.

The last presentation was dealing with the on-site measurement equipment. This tool provides a large and accurate capability for geometrical and corrosion measurements in getting information on the fuel behaviour.

The main conclusions that can be drown from this session can be summarised as follows:

The external environment (water chemistry, debris, etc.) has to be taken into account when designing the fuel. The objective of reducing debris-induced defects seems to be close to achieved today. Water chemistry is still an important issue, especially when increasing the cycle duration. Efforts on understanding crud formation and the impact of water chemistry on the corrosion rate should be maintained.

On-site measurement can be considered accurate enough to be used by the designers with profit. However, the possibilities of measuring other parameters, such as rod diameter or FGR should be considered. And, even more important, the time needed for the fuel measurements should be reduced, the utilities being reluctant to allow extra time for such mesurements.

SESSION III

In session 3 we had papers dealing with operational experience and design of control rods. We also got information on experience feedback related to compartment of control rods and fuel behaviour in the reactor.

French experience on RCCAs was presented, including a RCCA management strategy and design improvement of RCCA. A research program is in progress, with the aim to build a set of physical models to introduce them in a computer code in order to be able to predict RCCA wear.

A German RCCA management concept was presented. It will help to ensure that the RCCAs do not reach their thinning limit, and that they will not be limited by interaction of cladding creepdown and absorber swelling during design lifetime. A report from Japan on an investigation of RCCA to establish a criterion for degradation of RCCA due to swelling of the rod tip was provided.

A report from Russia was given that describes the design work for control rods in WWER reactors. They have performed a study of different absorber materials.

An optimisation study of AP-600 grey control rod design showed two designs to improve load following.

A report was given of control rod cluster drop time anomaly at Guandong Nuclear Power station and EDF 1450 MWe PWRs. In the paper the test programs are described that were employed to identify the cause of the problem and to justify the continued operation of the reactors.

A description of the problem with sticking control rods in Ringhals 3 and 4 was provided as well as an explanation for how this complex problem was solved. We got information on the problems with sticking control rods in the US. The solving of the problem is in progress. We were also provided with a report on the problem with sticking control rods in Belgium.

In this session we saw the importance of inspection program for the RCCAs to detect problems in an early stage. A management plan for RCCAs are necessary. The plan should include testing and inspection. The basic life limiting factors are known for current design of RCCA, for example control rod cracking and wear. But it is an incentive to extend the lifetime of RCCAs and new designs are under development. When introducing new designs of fuel and RCCA it is important to verify the performance in the reactor. The design work for grey control rods highlights the connection between safety considerations and operational flexibility.

From the presentations given on the topic of experience feedback on the control rods, it appears that similar problem has been observed concerning stuck control rods in Sweden, USA and Belgium.

The events that have been reported highlight how to conduct an investigation to find the root cause and how to justify restart of the reactor when the information of the problem is incomplete. The events also raise the question whether it is acceptable to operate a reactor with degraded safety function, in this case the scram system. This also raises the question if new fuel types should be tested more thoroughly and if you should have longer lead test programs for new designs before commercial reloads.

Investigations should be made on RCCAs and fuel after the irradiation time and conditions should be compared with the design calculation results. Representative validation testing must be performed before irradiation in the reactor. As far as test programs for new designs are considered, like new fuel types, new core loading pattern, new types of RCCAs etc. they should be reviewed. One should ensure that the safety functions are always fulfilled.

SESSION IV.A

The evolution of ABB fuel designs both for BWR and PWR was presented, focusing on improvements on fuel performance and fuel behaviour. The capabilities of the SVEA 10x10 BWR fuel in terms of thermal margin performance were highlighted. The Frigg'95 test facility was described, as well as the current fuel failure statistics that show debris as the single known failure cause for this fuel design due to the reliability of clad with Zirconium liner.

Cladding corrosion improvements for PWR fuel (developed through the NFIR project) were described, together with the updated fuel failure statistics.

The current 8x8 BWR fuel generation in Japan was described next, as well as the development of 9x9 lattices for the next generation, with a burnup target of 45-50 Gwd/MtU. The approach followed to design this new fuel for D-lattice plants was described in detail.

Next was the only MOX fuel related presentation of the meeting. It described the Advanced Thermal Reactor Concept, and the experience obtained so far from the prototype "Fugen". A demonstration program for high burnup MOX fuel is being carried out in this prototype and a fuel performance analysis code is being validated. The coming licensing process of the code was outlined.

The last presentation covered the segmented fuel rod irradiation program, a research activity aimed to test new clad materials for PWR reactors. This international effort involves partners from Japan, Spain and the USA. The objectives and the current status of the program were described, and some preliminary results showing an improved behaviour in the new materials were provided.

The future phases of the program, namely ramp testing of selected segments and hot cell examination, were also outlined.

SESSION IV.B

The papers presented in this technical session describe the demanding nature of the manufacturing process to produce defect-free fuel, the care and engineering expertise needed to ensure compatibility of fuel from multiple vendors in mixed cores, illustrated the opportunities for the continued optimisation of fuel and fuel management strategies, and reinforced the idea that the operating experience continues to present new problems and opportunities to improve safety.

Activities aimed to reduce the number of rod failures were covered in two papers. Fabrication improvements and total quality initiatives at ENUSA were described in detail in one paper, while the other listed the advantages of ultrasonic inspection of the rod welds when compared with the traditional X-ray inspection.

The issues associated to mixed cores were also treated in two papers. ENUSA described the general trend to operate mixed cores, due to economic pressures and performance optimisation, and stressed the need for a comprehensive compatibility assessment to ensure that safety margins are adequately assessed and maintained. The paper from AVN described the current licensing requirements for mixed cores in Belgium, and also stressed the same need, highlighting that only knowledgeable suppliers and licensees could be able to fulfil it adequately.

ENUSA described in a paper the current design process for modern BWR fuel. Multiple design trade-offs were described, including energy, burnup, stability and thermal and reactivity margins, and how changes in some design features could improve behaviour in each of those fields while reducing the fuel cycle cost. The need for knowledge and sophistication of both the supplier and the licensee was again stressed.

Illustrating how fuel design and management improvements can effect other areas, ENUSA described a low leakage loading pattern strategy aimed to reduce the neutron fluence at the reactor vessel.

The last presentation dealt with the operating experience with control rods in Belgium, highlighting the incomplete rod insertion issue, and describing the related regulatory requirements.

Taken together, the papers presented during this session remind us of the need for careful design and manufacturing, and shows that the operating experience continues to provide very valuable lessons. The experience also suggests the continued need for lead test assembly programs and for careful adoption of increased burnup limits and, in general, of other new features.

OPENING SESSION

Meeting objetives

A. Alonso
Counsellor CSN and CSNI Vice-Chairman

It is accepted that the core of the nuclear reactor in a nuclear power plant it is where the basic events take place. The phenomena associated to the physics of fission occurs inside the fuel while the control rods are the key components to keep the chain reaction stable. Once mass is converted into kinetic energy of particles and into radiation, the rest, up to having the faithful electrons active into the electrical grid, it is based on mostly conventional, well known technologies. It is then no strange that the Committee on the Safety of Nuclear Installation (CSNI) have considered and shown concern, since its creation, on the behavior of fuel and control rods in operating power plants, as well as on its design evolution and regulatory implications.

Well before CSNI acquired its present organizational scheme comprising *five* Principal Working Groups (PWG's), already in the old days of the Committee on Reactor'Safety Technology (CREST), the predecessor of CSNI, during its annual meeting, in the sixties and early seventies, it was costumary to include a session on *operating experiences,* what is now the responsibility of PWG num. 1. If you were to look at the record of these days you will find many referents'to fuel and control rod induced incidents, as well as descriptions of design and development efforts and of safety concerns. I am obliged to recall now the excellent information provided by tbe pioneers of the time: Dr. Cliford K. Beck, from the US Atomic Energy Commission; Dr. Jean Bourgeois, from the French Commissariàt a l'Energie Atomique, and Prof. Reginald F. Farmer, from the UK Atomic Energy Authority, among others.

Although such early efforts are now very far from the problems of 1996, they show an unabated preoccupation for the problem, despite the significant breakthroughs and improvements in the design, fabrication and operation of the reactor core. The worldwide research and development effort which has led to the present cladding materials and neutron absorbents, together with the modern handling and manufacturing of the uranium oxide pellets, all that constitute a brilliant chapter in the *Book of Knowledge* written in the second half of our twentieth century.

The safety record of the core in light water reactors has been generally well within the safety limits imposed; there have been leaking fuel rods, stuck control rods, crud induced localized corrosion, hydriding and even mechanical induced failures, but all of that has been investigated and solved. Nevertheless there are now more constrains than in the past. *Society* is today demanding higher levels of safety and is intervening in the regulatory process of our advanced democratic states. Failures in fuel rods and control elements, however small and insignificant, catch instant public attention and have to be explained thoroughly. The *decision makers* are more and more demanding competitiveness from the plant owner/operators, while non jeopardizing safety. The plant *owner/operators* themselves are responding to such demands by lengthening the fuel cycle and by considerably increasing the burnup of the nuclear fuel, what brings clear economical advantages as well as safety concerns. The great availability of plutonium from fuel reprocessing and military sources is prompting many *fuel manufacturers* to substitute uranium-235 for plutonium-239 through mixed oxide pellets, with the safeguards implications brought about by using a strategic material.

All these facts combined, of a sociological, political, economical and technological nature, shape our present and' their time evolution will certainly determine our future. The management of CSNI included all these aspects among the objectives of the Specialists Meeting and it expects from you, as participants, hard work, complete and deep discussions and enlightment through your suggestions and recommendations, all that to better serve our terms of reference which are mainly aimed at serving the needs of the member countries.

As you well know, Spain has in operation *nine* light water reactors, *two* of them more than *twenty* years old, and has accumulated more than a century of operating experience. Spain also manufactures fuel in the present competitive market. These facts prompted Spain to offer hosting the meeting, what was readily accepted by the CSNI.

The efforts put by our three institutions: UNESA, the consortium of Spanish electrical companies; ENUSA the fuel manufacturer, and the CSN, the nuclear regulatory authority, under the expert direction of the Nuclear Energy Agency, all they have made it possible the Specialists Meeting on *Nuclear Fuel and Control Rods: Operating Experience, Design Evolution and Safety Aspects,* you are going soon to initiate.

On behalf of the Chairman of the Spanish Nuclear Regulatory Council and of the others Counsellors, I welcome you to Madrid and wish you a profitable and pleasant meeting. Thank you for your kind attention.

CSNI/PWG1 Specialists Meeting on Nuclear Fuel and Control Rods: Operating Experience, Design Evolution and Safely Aspects.

Madrid, Spain, November 5th-7th, 1996
NEA Welcome Remarks

Good morning ladies and gentlemen. My name is Jean-Pierre Clausner and, on behalf of the OECD Nuclear Energy Agency (NEA), I am very pleased to address a warm welcome to all of you to this CSNI Specialists meeting on Nuclear Fuel and Control rod Issues. I would like also to express my thanks to our hosts, the Spanish organisations ENUSA, UNESA and CSN for their strong support and the greal deal of efforts they put in organising this meeting. Laoking at the large number of participants gathered in this room today, it is already a big success.

As you know, the meeting has been organised in the framework of the Committee on the Safety of Nuclear Installations of the OECD Nuclear Energy Agency. For those of you who are not familiar with the organisation, I should briefly explain the role of this Agency.

The NEA is one of the fifteen bodies that make up the Organisation for Economic Cooperation and Development (OECD) which is located in Paris. Since most of the activities of this organisation are oriented towards economics, the role of the Nuclear Energy Agency is less known and may be sometimes confused with that of the International Atomic Energy Agency (IAEA) in Vienna.

In connection with safety, the charter of the NEA calls for "... the promotion of the safety of nuclear installations...", "... the establishment of joint services for the prevention of accidents...", and for "... the dissemination of information... on the safety and regulation of nuclear activities...".

With some 80% of the 430 reactors opded around the world, Members of the NEA have set up and developed, over the years, a close operational feedback process in order to meet the primary objective of the Agency which aims to promote co-operation between regulatory authorities of its Member countries, contribute to nuclear safety and regulations of nuclear activities and, ultimately, to help in the development of nuclear energy as a safe environmentally acceptable energy source.

In the vast domain of nuclear safety and regulation, the NEA activities encompass: operating experience and human factors, primary coolant system behaviour, reactor component integrity, probabilistic safety assessment, severe accident management, nuclear fuel cycle, exchange of regulatory activities including inspections practices, and some specific projects like the OECD Halden Project and the RASPLAV Project. Most of these activities are driven by the two standing committees dealing with safety matters: the Committee on the Safety of Nuclear Installations (CSNI), and the Committee on Nuclear Regulatory Activities (CNRA) both consist of delegates representing regulators, technical support and researchers from the NEA Member countries[1].

[1] The OCDE Nuclear Agency (NEA) was established on 1st February 1958 under the name of the OEEC, European Nuclear Energy Agency. Its received its present designation on 20th April 1972 when Japan Became its first non-European full member. NEA membership today consists of all European Member countries of the OECD as well as Australia, Canada, Japan, Mexico, Republic of Korea and the United States.

While the CSNI has been responsible for more than 20 years for the exchange of information on safety research programmes enabling participating countries to develop common approach on identified safety issues, the CNRA, created in 1989, aims to exchange information and developments which could affect national regulatory practices, and achieve a common understanding on possible new regulatory requirements.

In brief, the role of these committees is neither to impose nor judge but, rather, through exchanging and understanding, to reach a common technical opinion and develop international consensus based on a better knowledge of technical issues and regulatory practices.

With respect to the subject which brings us here in Madrid today, the decision to organise a specialists meeting was taken in September 1995 by the Principal Working Group N°l on Operating Experience and Human Factor (PWGl) as a follow up of several fuel and control rods deficiencies reported to the NEA. Incident Reporting System. The Group expressed its concern regarding potential safety aspects which might arise from such events. Since it was decided to organise that meeting, several other events involving fuel and control rods occurred in the United States, in Europe too, confirming not only the relevance but also the timeliness of this meeting.

Of course, you are all aware that the current weakness of the nuclear market, is one factor which drives the utilities to look for better availability of their plants, in shortening outages for refuelling, in using more efficient fuels during longer cycles, in increasing the power of their units and their lifetime as well. Fuel assemblies and control rods are among these components subjected to these changes. Some of them are important and may have significant impact on the safely of both existing plants and new plant designs, and they are a subject of concern for regulators which are carefully following these issues. Currently many questions, regarding fuel optimisation, changes in plant parameters and their consequences on safety margins, and on fuel and control rod reliability, have been raised from the operating experiences and recent incidents.

I hope that this meeting will give you the opportunity to address those topics openly, to exchange your views and experiences, to discuss regulatory practices and identify research needs, and, ultimately to develop conclusions and proposals for areas which would require more attention. I am convinced that with the group of experts gathered here today the meeting will meet its objective and hopefully your expectations.

In closing, I would like to reiterate my tbanks to our hosts as well as to you all for your contribution in the success of the meeting. I recognise that the Programme Cormmittee set up a very tight schedule, but I do hope that you will have the chance to enjoy the beautiful City of Madrid. I wish you a very interesing and productive meeting.

Opening Remarks

J. M. Conde
Chairman of the Programme Committee

Good morning, ladies and gentlemen.

My name is José Conde. I work for the CSN and I'm the Chair of the Programme Committee for this meeting on nuclear fuel and control rod behaviour.

As Professor Alonso and Mr. Clausner have already described the objectives of the meeting, I will very briefly describe the technical contents and the structure of the meeting.

In the programme you have received together with the meeting documentation you can see that four technical sessions have been prepared. The first one is intended to describe the general perspective of the problem in different countries, and will provide the basis for the more detailed descriptions to be presented in later sessions.

Sessions 2 and 3 focus on the fuel and on the control rod issues respectively. I want to draw your attention to the fact that an effort has been made to build a comprehensive session on control rod behaviour, with papers describing all the recent incomplete control rod insertion events.

I have to point out that a paper from this session, identified as 3.9 has been included in the last session 4.b. We have been asked to do so by the Belgian representatives, because they want to talk one after the other.

Session 4 has been subdivided into two parts, both dealing with the improvements on fuel design and fabrication and core management.

In addition to these sessions, the Programme Committee deemed appropriate to have some invited speakers treating some general subjects that could prepare the audience for the technical sessions.

The first paper, presented by Mr. Hanevik, will review the status of the Halden Project to describe the fuel issues being investigated there, specially those that could havean impact on the future licensing perspective.

The second paper deals with the impact on fuel behaviour of the BWR stability issue. As you might know, the PWG2 of the CSNI has recently finished a State of the Art report on BWR Stability. Mr. Félix Castrillo from Iberdrola has participated in this effort and will describe for us the aspects that have been found to have a potential impact on fuel behaviour.

Mr. F. Schmitz, who will update for us the status of the high burnup fuel issues, will present the last invited paper. Many of you were present at the very interesting meeting held at Cadarache one year ago, at which these problems were described in detail. The Programme Committee thought that an update on the status of this issue was very convenient, although high burnup is not an issue to be covered at this meeting. The presentation by Mr. Schmitz will be complemented by some remarks offered by Mr. Larry Phillips, from the US-NRC, who will add the licensing perspective of this issue to Mr. Schmitz's talk.

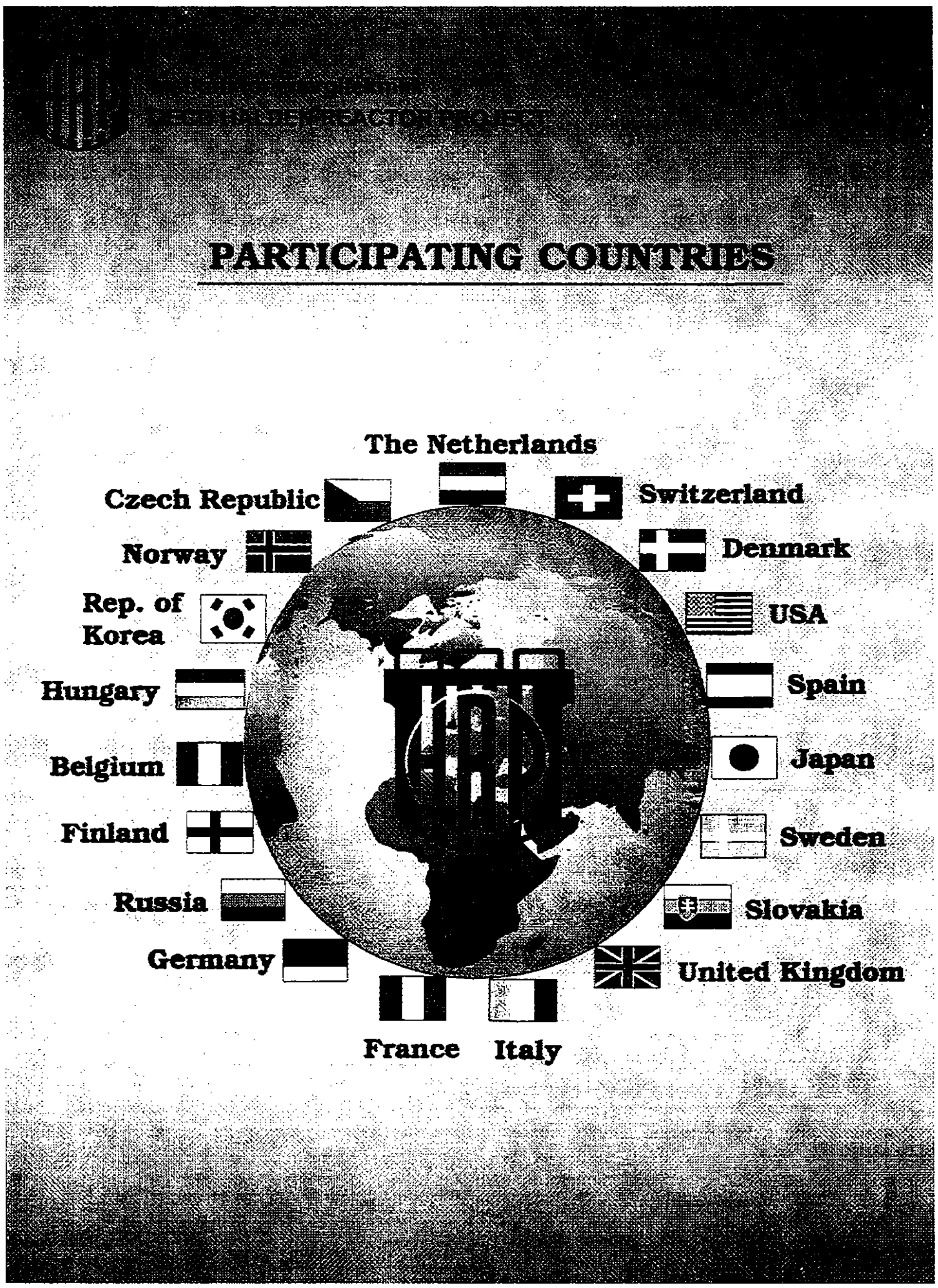
Only two more comments about the meeting organisation. The meeting documentation given to you includes a draft compilation of the papers received as the authors sent them to us. They are not the meeting proceedings, formal proceedings will be edited on due course after the meeting, but the Program Committee thought that it would be very convenient for the participants to have a copy of the papers during the technical sessions. I want to bring your attention to the fact that the paper by Mr. H. Wilson in Session 3 is not included in the compilation. Given the present status of the issue, there will be no formal paper submitted for the proceedings. However, the slides used for the presentation will be made available during the meeting. Also, the title included in the program for this paper was tentative. The correct one is "Incomplete RCCA Insertion Observations in Westinghouse Fueled Plants".

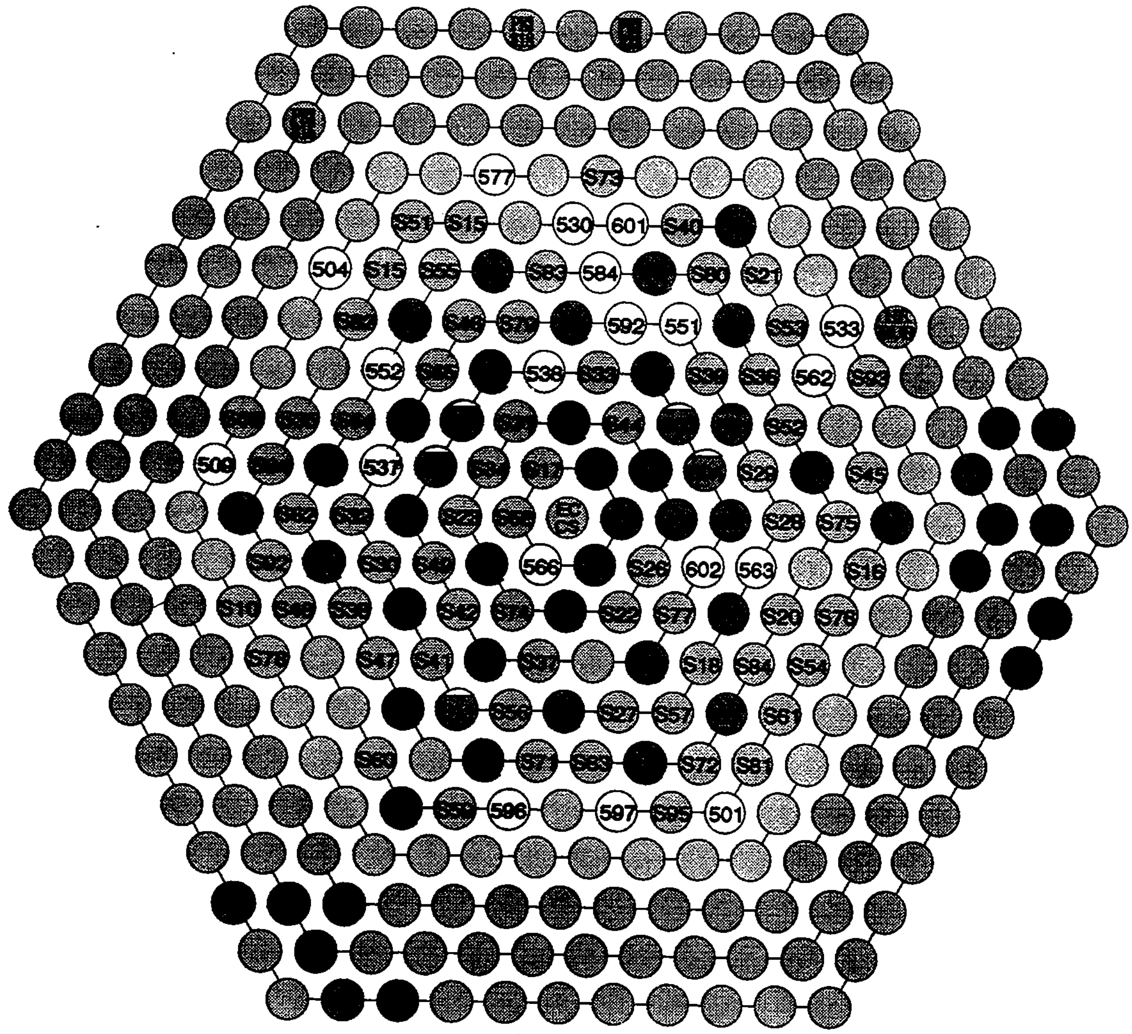
I guess that is all I have to say by now. Apart from the technical quality of the presentations that will take place, the most valuable result that will be extracted from this meeting will come from the discussions between the participants. I encourage you all, the speakers and the audience, to actively participate in the discussions that will follow each presentation and at the final discussion panel.

I wish you all a pleasant and fruitful meeting. Thanks very much for your attention.

Invited Paper:
The OECD Halden Project

A. Hanevik





HBWR CORE LOADING No. 94, December, 1994

Control Station (CS 19)	30
Neutron Shield Bolt	14
570 Instrumented Fuel Assembly(IFA-570)	26
S41 Spike Assembly	66
Pressure Flask	1
Driver Assembly	4
Fuel Flask Assembly	5

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OECD HALDEN REACTOR PROJECT

Recommendations 1997 - 1999 - Fuel Issues

High Burnup Performance

- Burnup range 40 - 70 MWd/kg UO_2
- Representative materials and designs

Fuel Types

- Standard UO_2
- Gadolinia
- WWER
- MOX
- Advanced

Performance Issues

- Thermal
- Fissions gas/overpressure
- PCMI

Operational Issues

- Power changes
- Dry - out
- LOCA
- Degraded fuel

Testing Objectives

- Separate effects - mechanisms
- Integral tests - verification

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Recommendations 1997 - 1999
Materials Issues

Cladding Corrosion and Hydriding

- Separate effects - mechanisms
- Standard-advanced alloys
- Water chemistry - crud
- Pre-irradiated materials

Cladding Creep

- Modern materials
- High fluence materials

Irradiation Assisted Stress Corrosion

- Separate effects - water chemistry, fluence, stress intensity, materials composition
- Electrochemical sensor development
- Electrochemical noise measurements
- Miniaturized fracture mechanics specimens
- High fluence materials
- Extend to PWR-conditions
- Mechanistics understanding - modelling
- Mitigating measures

Pressure vessel Embrittlement

- Cooperative work arrangements

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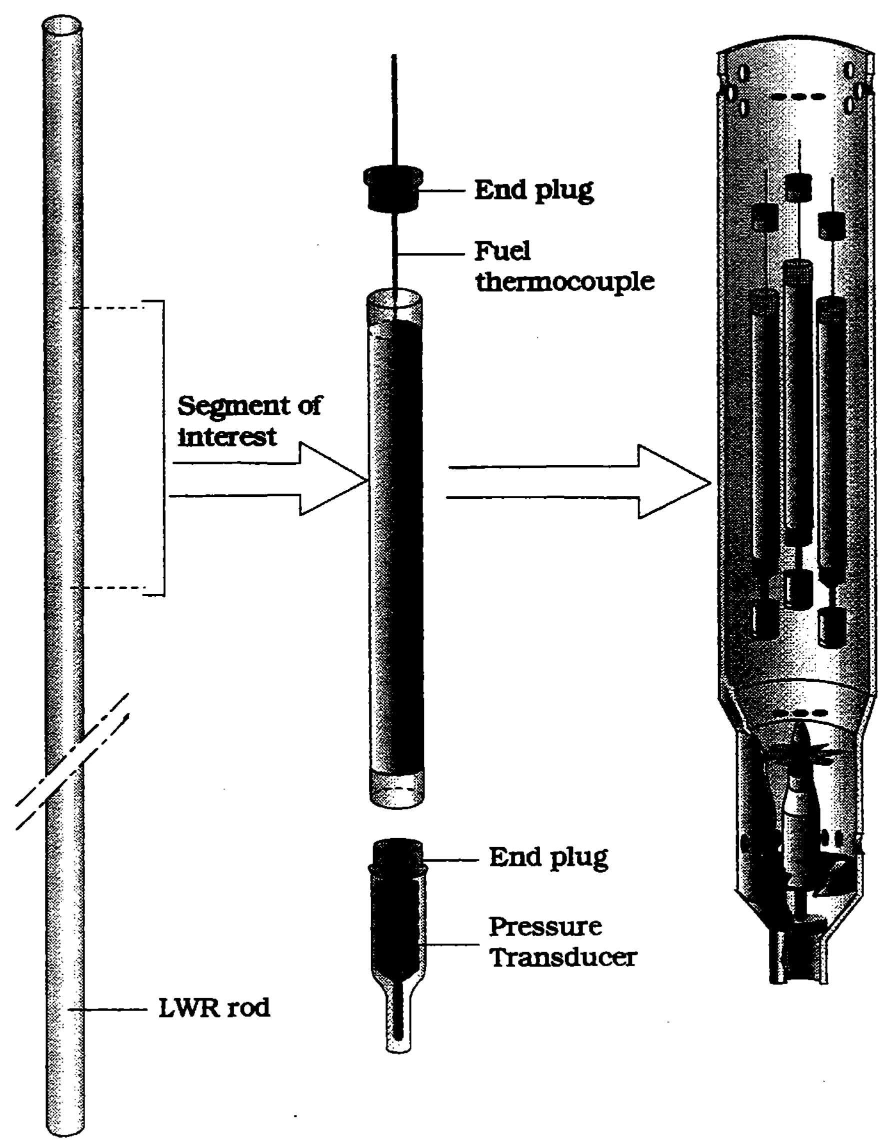
SCHEMATIC OF THE UTILISATION OF COMMERCIAL FUEL, REINSTRUMENTED WITH FUEL THERMOCOUPLE AND PRESSURE TRANSDUCER



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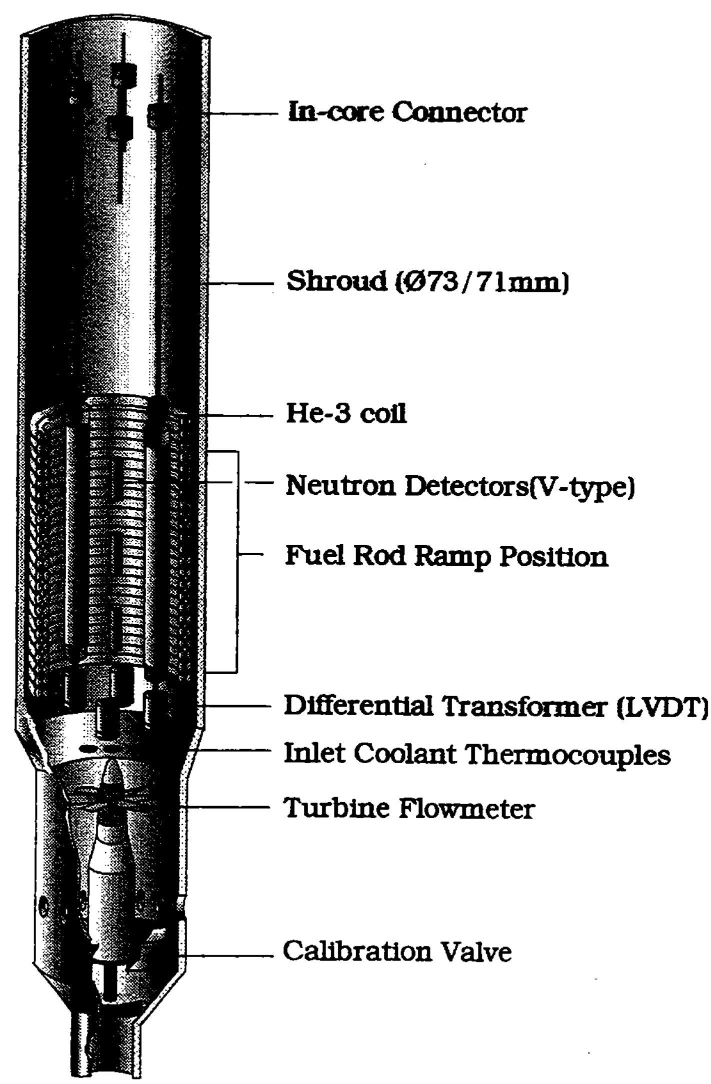
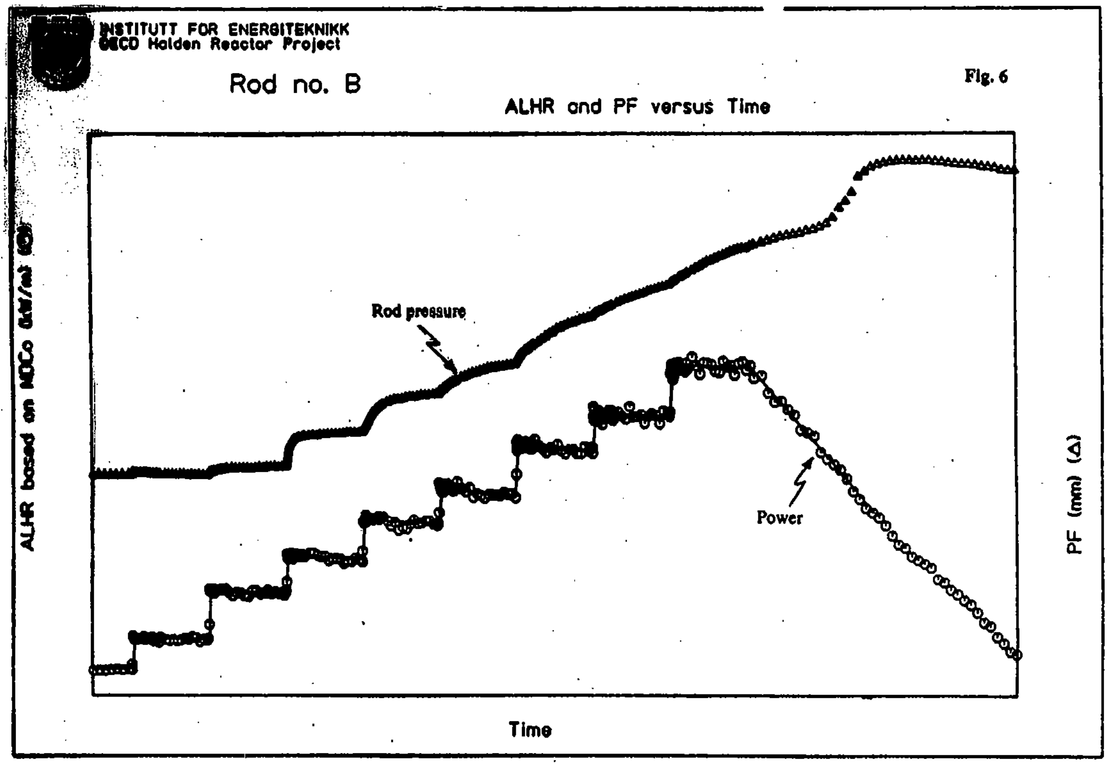
OECD HALDEN REACTOR PROJECT

TEST RIG FOR POWER BUMPS OF HIGH BURN-UP COMMERCIAL FUEL, REINSTRUMENTED SEGMENTS





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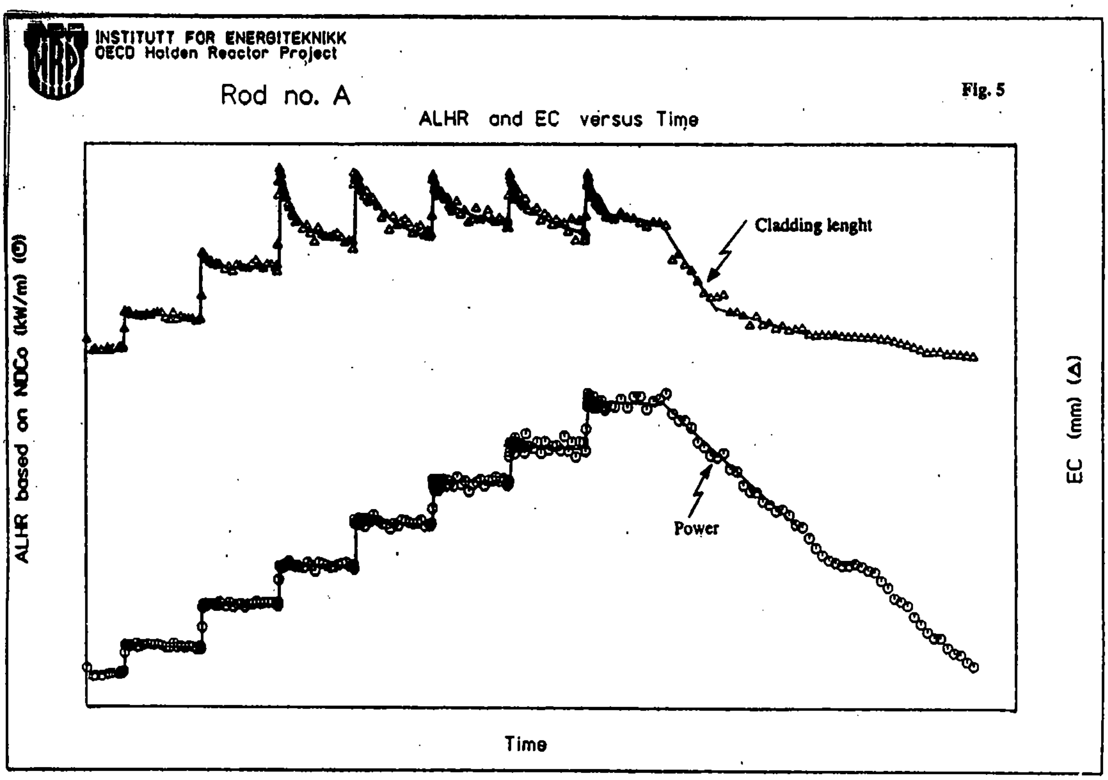
OECD Halden Reactor Project
Rod no. B
Fig. 6
ALHR and PF versus Time
Rod pressure
Power
ALHR based on NDCo (kW/m) (O)
PF (mm) (Δ)
Time



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OECD Halden Reactor Project
Rod no. A
Fig. 5
ALHR and EC versus Time
Cladding lenght
Power
ALHR based on NDCo (kW/m) (O)
EC (mm) (Δ)
Time

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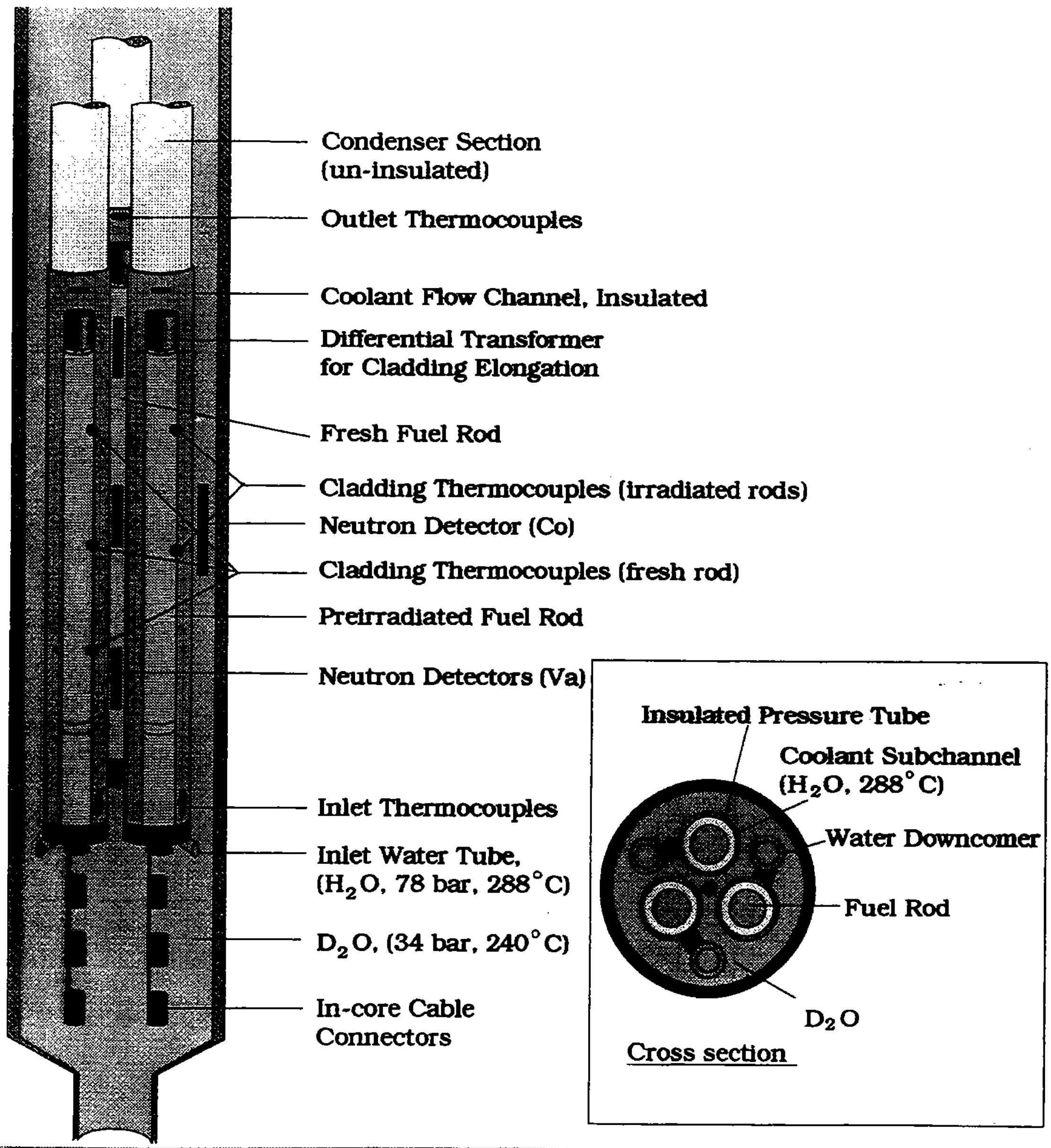
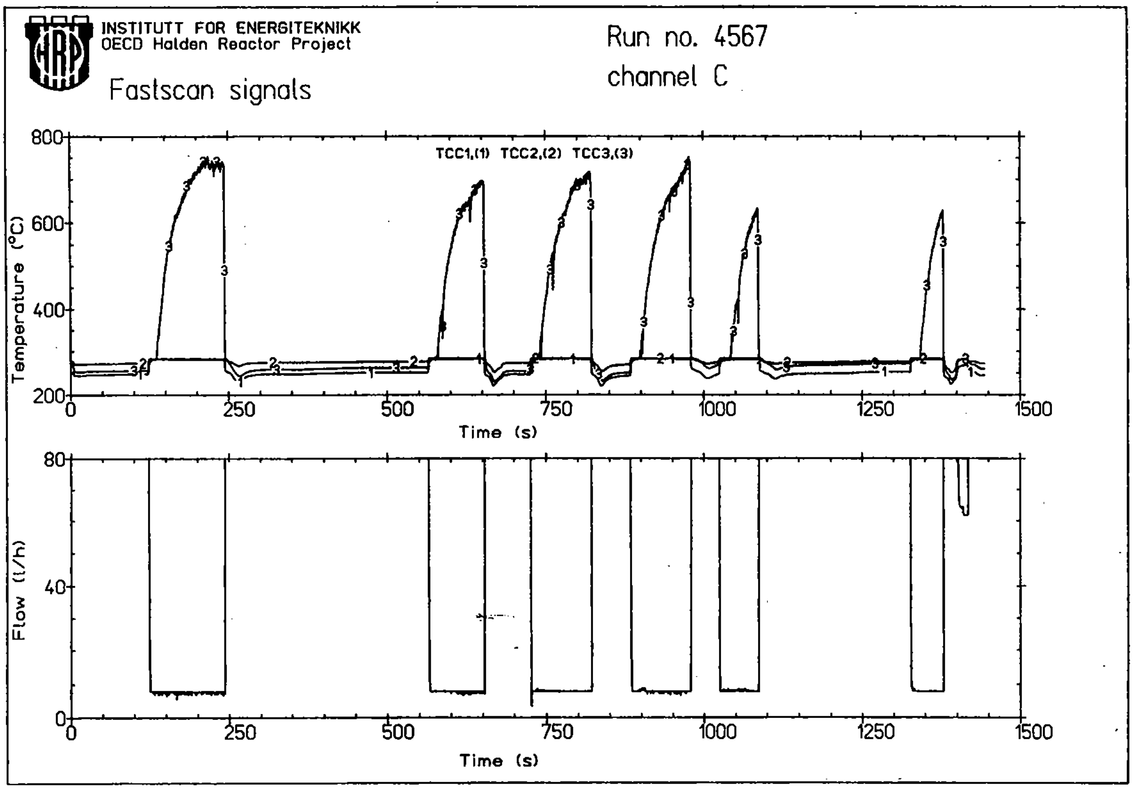
Dry-out Test. Schematic of Test Rig IFA-613

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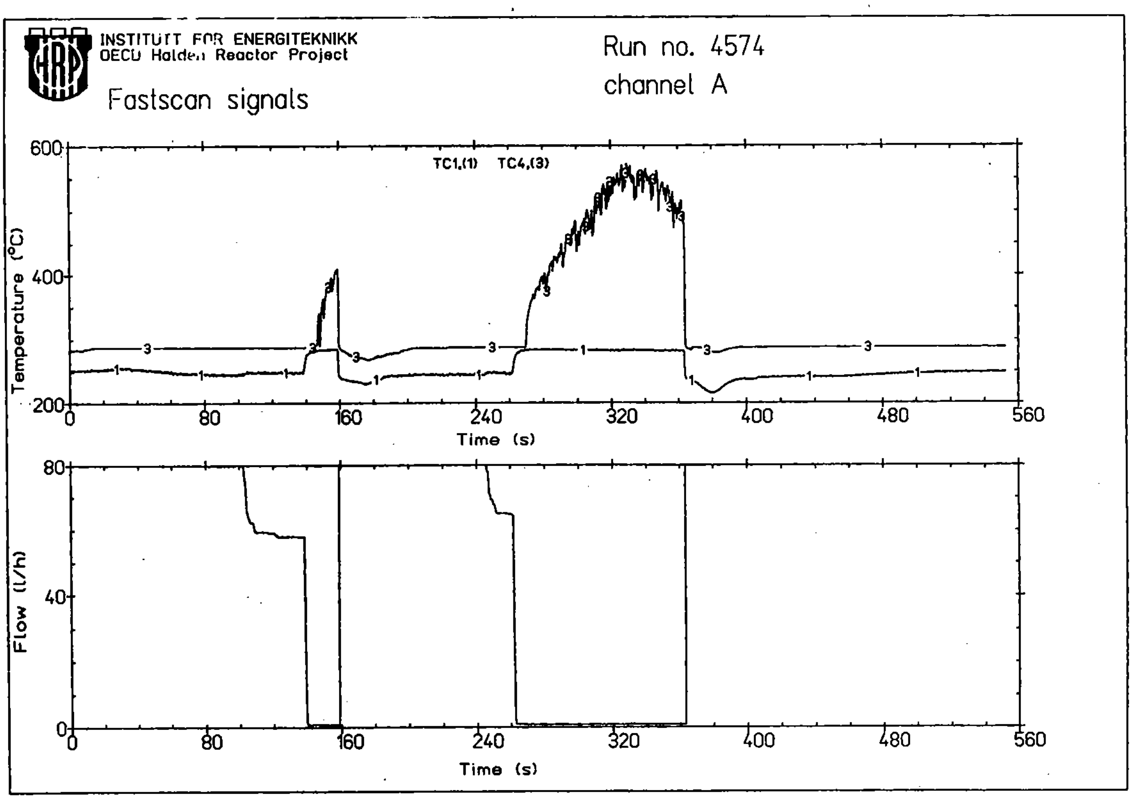
OECD Halden Reactor Project
Fastscan signals
Run no. 4567
channel C
TCC1,(1) TCC2,(2) TCC3,(3)
Temperature (°C)
Time (s)
Flow (l/h)
Time (s)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OECD Halden Reactor Project
Fastscan signals
Run no. 4574
channel A
TC1,(1) TC4,(3)
Temperature (°C)
Time (s)
Flow (l/h)
Time (s)

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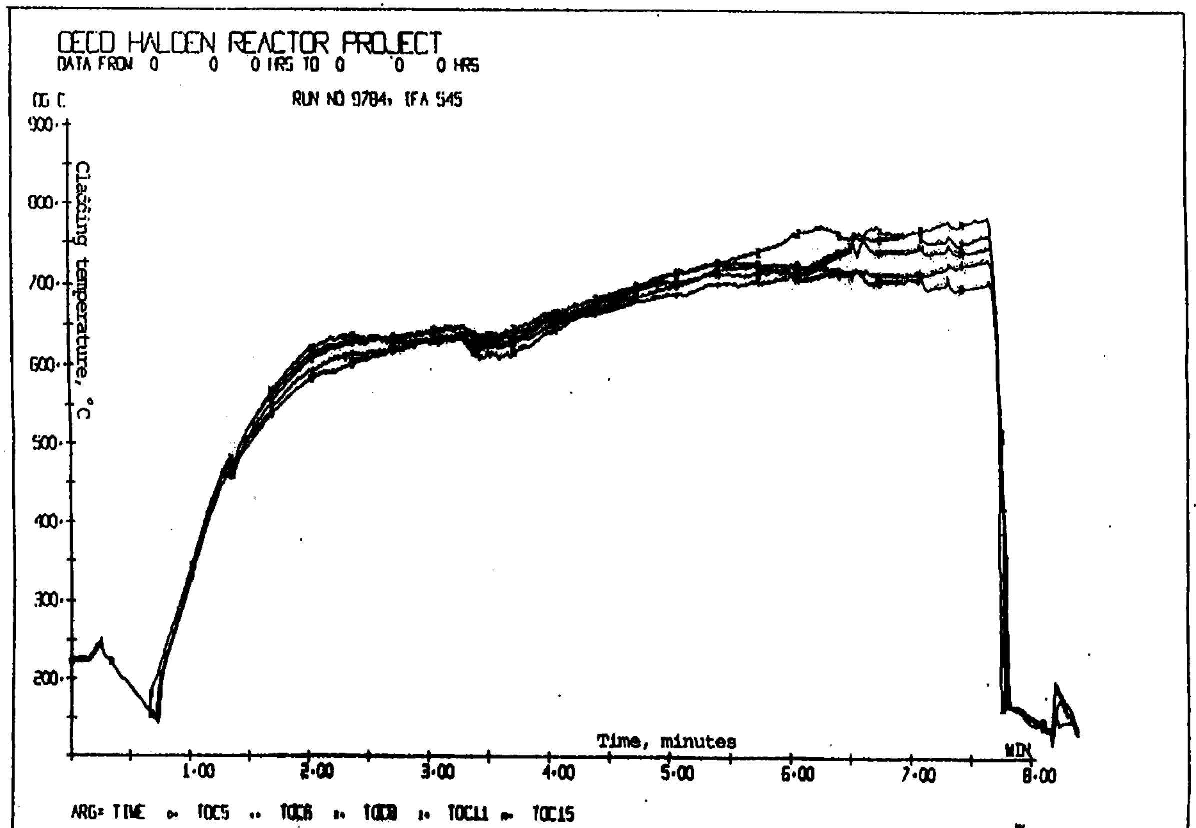
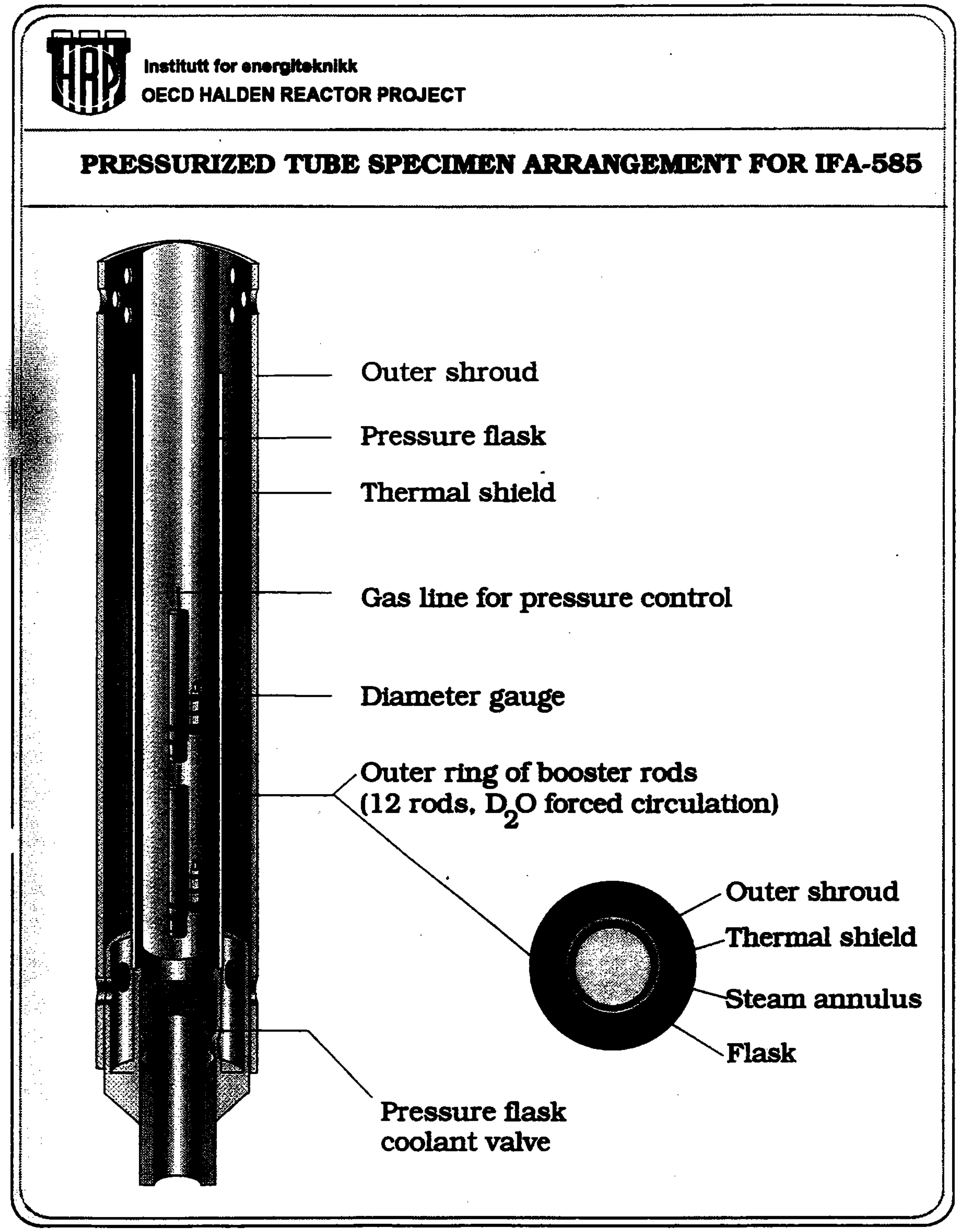
CLADDING TEMPERATURE TRANSIENTS IN PREVIOUS BELLOWING TESTS AT HALDEN

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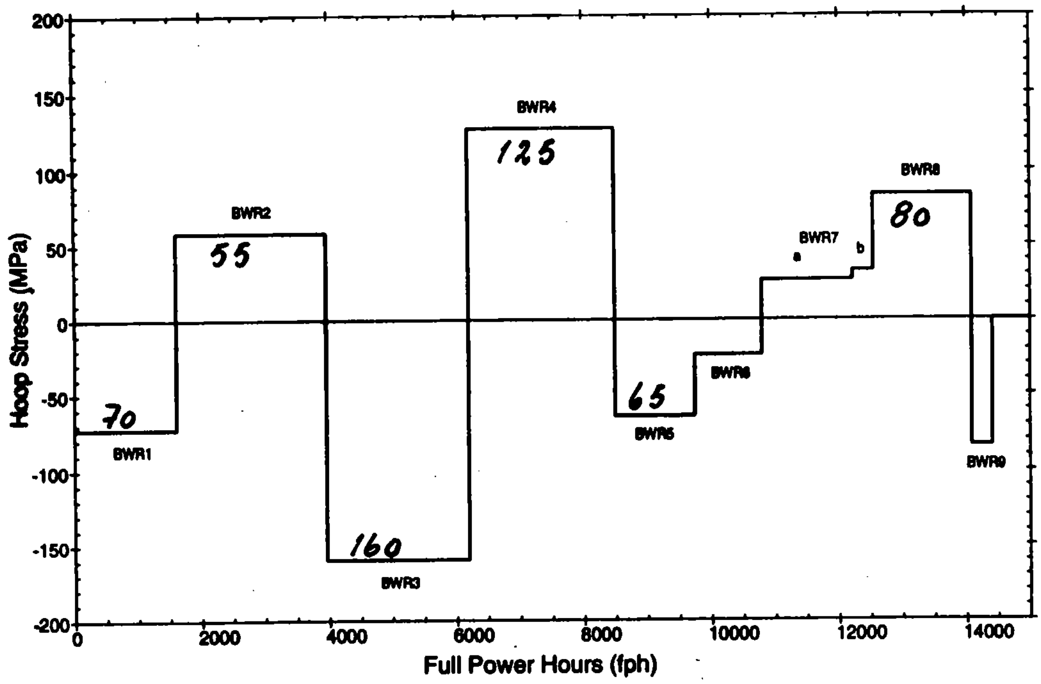
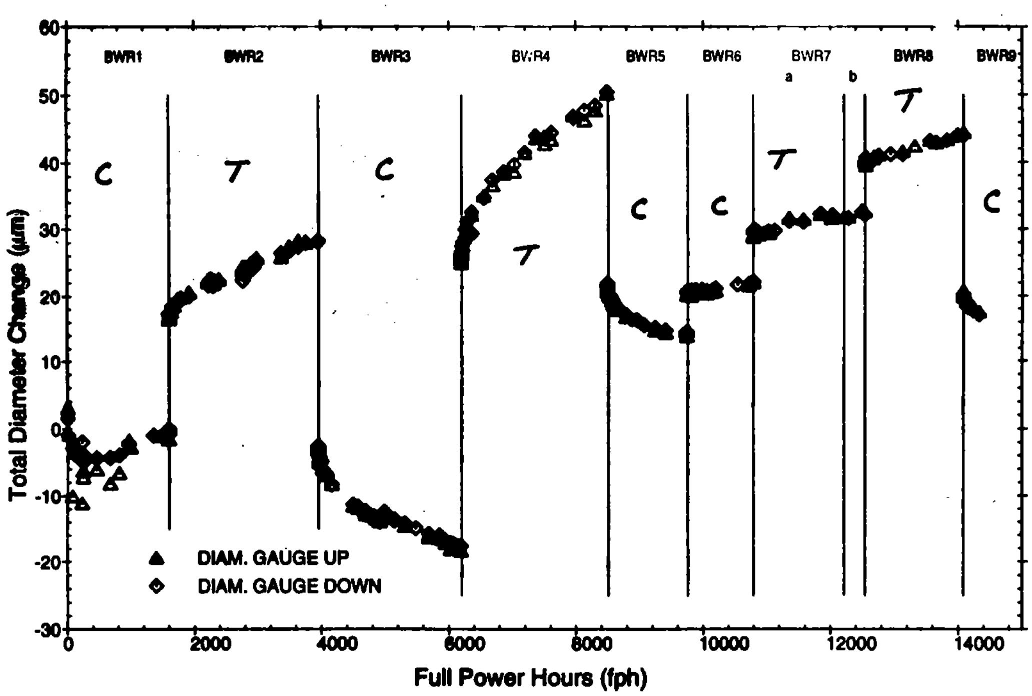
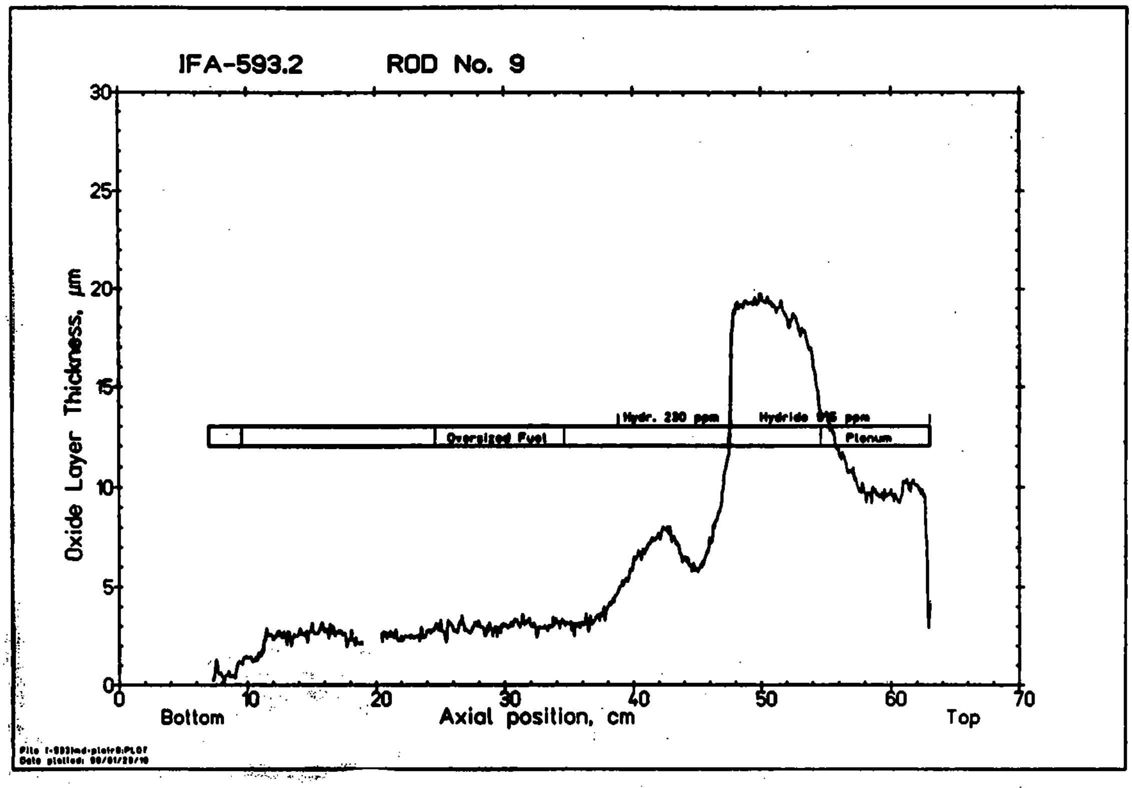
OECD HALDEN REACTOR PROJECT
PRESSURIZED TUBE SPECIMEN ARRANGEMENT FOR IFA-585
Outer shroud
Pressure flask
Thermal shield
Gas line for pressure control
Diameter gauge
Outer ring of booster rods
(12 rods, D_2O forced circulation)
Outer shroud
Thermal shield
Steam annulus
Flask
Pressure flask
coolant valve

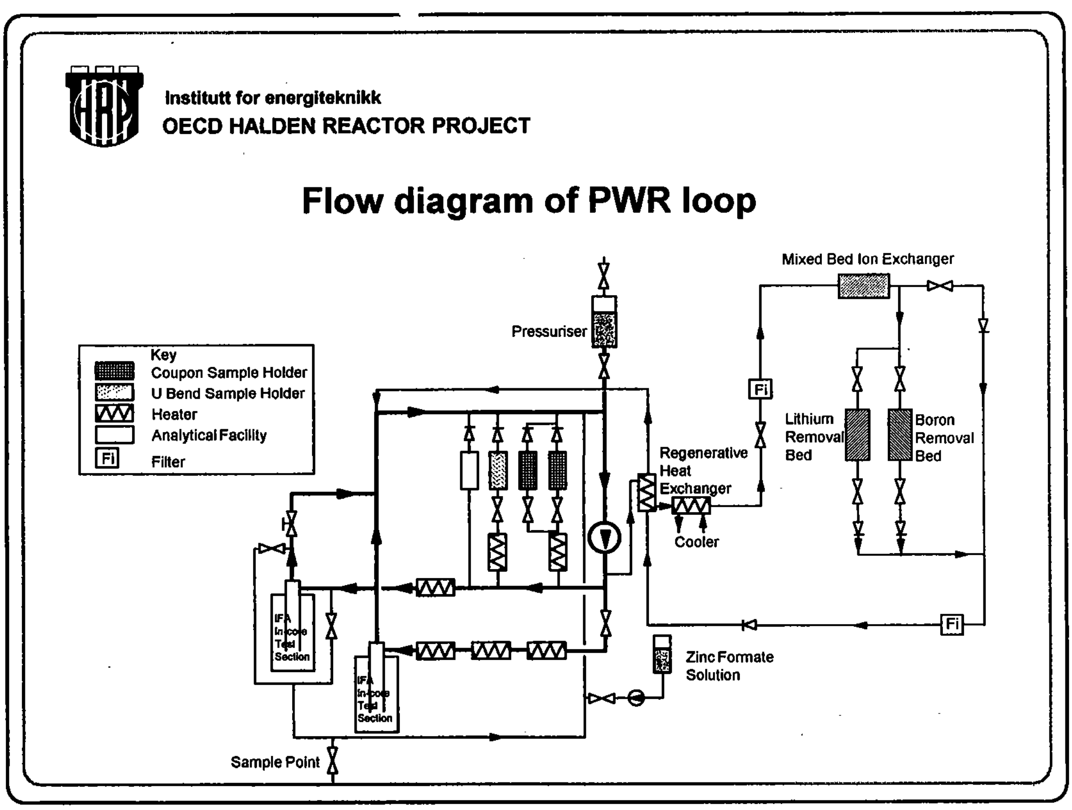
CREEP TEST.BWR ROD.STRESS HISTORY



CREEP TEST.BWR ROD.TOTAL DEFORMATIONS.




IFA-593.2
ROD No. 9
Oxide Layer Thickness, µm
Hydr. 230 ppm
Hydride
ppm
Oversized Fuel
Plenum
Bottom
Axial position, cm
Top
0
5
10
15
20
25
30
40
50
60
70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Flow diagram of PWR loop
Mixed Bed Ion Exchanger
Pressuriser
Key
Coupon Sample Holder
U Bend Sample Holder
Heater
Analytical Facility
Fi
Filter
Lithium Removal Bed
Boron Removal Bed
Regenerative Heat Exchanger
Cooler
IFA In-core Test Section
Zinc Formate Solution
Sample Point

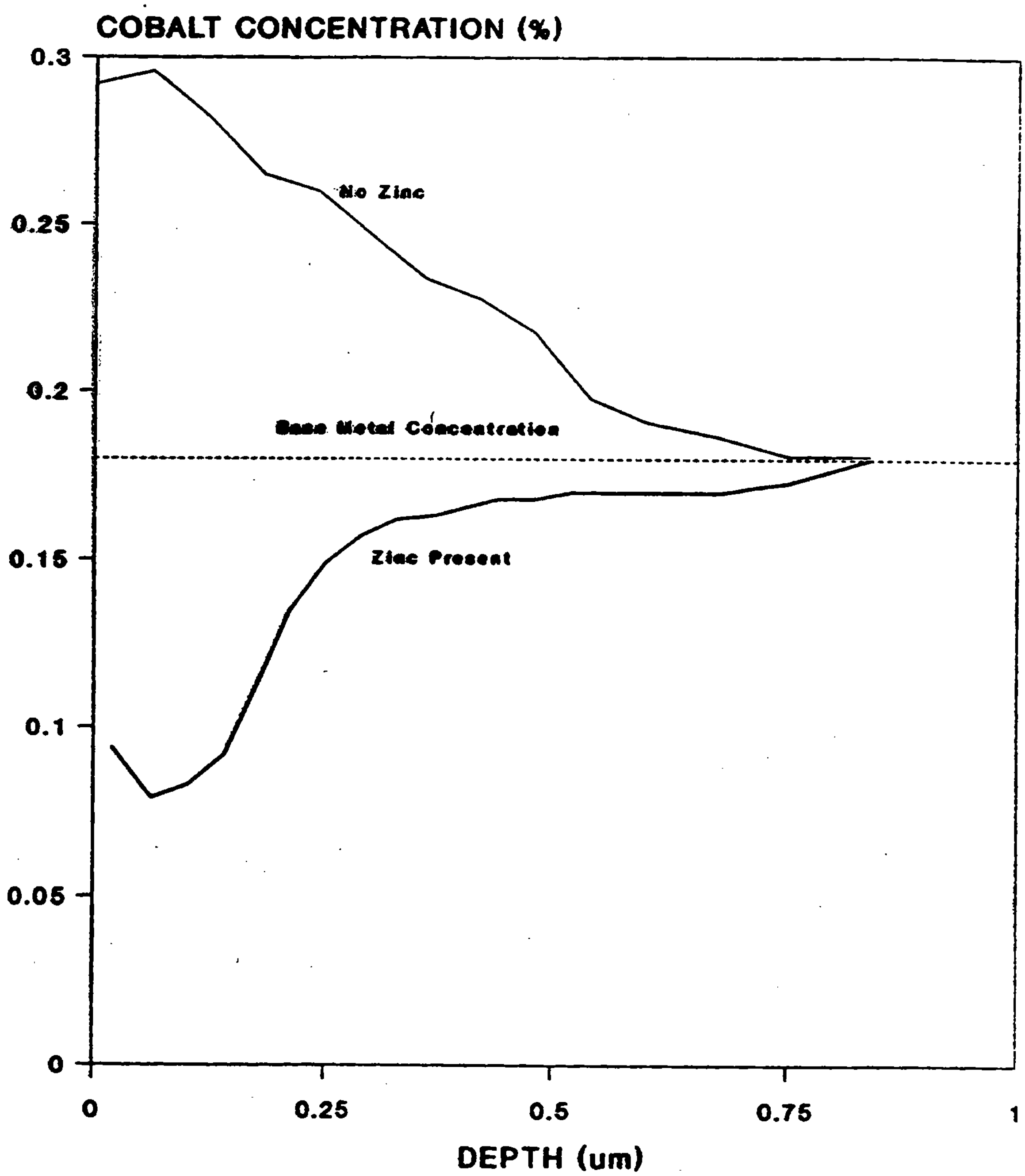

FIGURE 25. COMPARISON OF COBALT PICK UP BY NEW 304L S/STEEL COUPONS WITH AND WITHOUT THE ADDITION OF ZINC TO THE COOLANT

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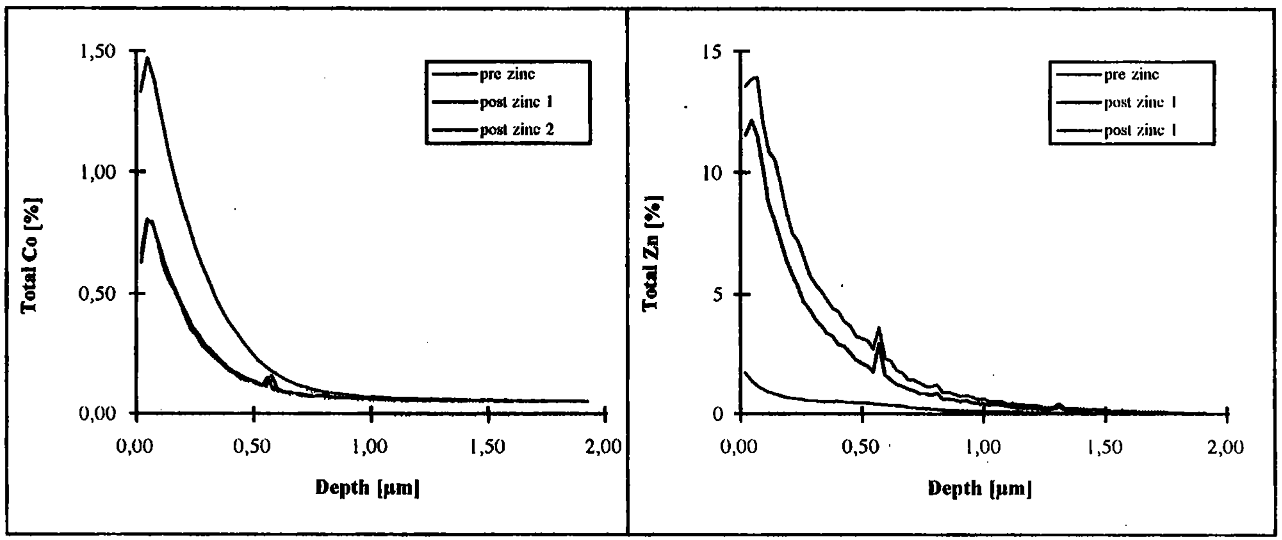
Cobalt and zinc profiles through pre-oxidised Inconel-690 coupon



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Main experimental facilities

- Halden Reactor
- Fuel inspection compartment
- Hot cells at Kjeller (near Oslo)
- Chemical laboratories (Halden & Kjeller)
- Fuel fabrication laboratory (Kjeller)
- Instrument / rig workshop (Halden)

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Effects of Water Chemistry on Co Deposition in BWRs

Results

- In NWC, thicker oxide layer
- higher Co-60 deposition + activity

Next step

expose NWC coupons to HWC + vice versa

BWR Stabilitv. Impact on fuel behaviour

Felix Castrillo
IBERDROLA

Abstract

The issue of BWR stability has deserved special attention in the last years, since the instability events reported from several reactors. Different approaches are followed in plants to avoid potential unstable conditions and to detect and suppress core power oscillations. Fuel behavior during power oscillations shows that its integrity would be endangered only in the case of extremely high oscillation amplitudes. The characteristics of new fuel designs that can affect to stability are reviewed, as well as the actual trends in the technology related to this subject.

Some historic background

Looking back at the beginning of the Boiling Water Reactor Technology, in the 50s and 60s, when this type of reactors were on their experimental phase (BORAX and Experimental Boiling Water reactors), the subject of neutronic stability was one of the concerns for these reactors to become commercial.

Eventually, the BWR technology was adopted for commercial use. The design incorporated some sort of penalty to assure the stable operation of the core, i.e.: an extra pressure drop was added in the core flow path, by means of tight orifices located at the inlet of the fuel elements, with the objective of increase the ratio of the single phase to the total core pressure drop.

Extensive test programs were established in the startup phase of the early commercial reactors (Dresden, Big Rock Point, Garigliano) to verify their stable operation.

In the 70s, additional test were performed (Peach Bottom, Vermont Yankee) mainly with the objectives of validating the new analytical tools under development at that time and to verify that power oscillations, in case of appearing, would be terminated by reactor scram. No oscillation incident was reported in several years of BWRs operation

During the 80s, a number of situations of neutronic oscillations were observed in several plants, either during programmed tests or as operational incidents: Caorso, TVO, Leibstadt and La Salle,. It was the La Salle event, and the accumulated experience from the other previous events, the point that marked some change in the way the instability issue was being addressed, especially in USA. The conclusions from this event showed that there was a large error in calculated decay ratio (which is the figure of merit used to evaluate the stability margin), that the power oscillations were greater than expected and that plant instrumentation to detect the unstable situation may not be adequate. Based on these conclusions, the focus was changed to operational recommendations and training to avoid potential instability regions, and to assure the capability to detect and suppress oscillations. Decay ratio calculations, lost their credit, and the search for a solid and long-term solution to this issue was initiated upon request of USNRC, while interim operational recommendations, oriented to avoid potential unstable conditions and to recognize and suppress oscillations, were issued and implemented in US plants.

Many European BWRs have adopted a different policy, in close connection with their regulatory requirements, oriented to perform plant measurements on a regular basis, to implement automatic actions to exit unstable regions and to use stability monitors to control on-line the stability margin. Especial emphasis is given to operators training to recognize oscillations with existing plant instrumentation

The approach followed by Japanese plants include the definition of potential unstable regions, the automatic insertion of selected rods to exit from potential instabilities, and the use of stability monitors.

Some other events in European and American reactors have been reported in recent years, and have induced updates to these recommendations, with special emphasis on operator training and requesting drastic suppression of oscillations once there is evidence of any unstable situation. Figure 1 shows the trace of average power during an instability event.

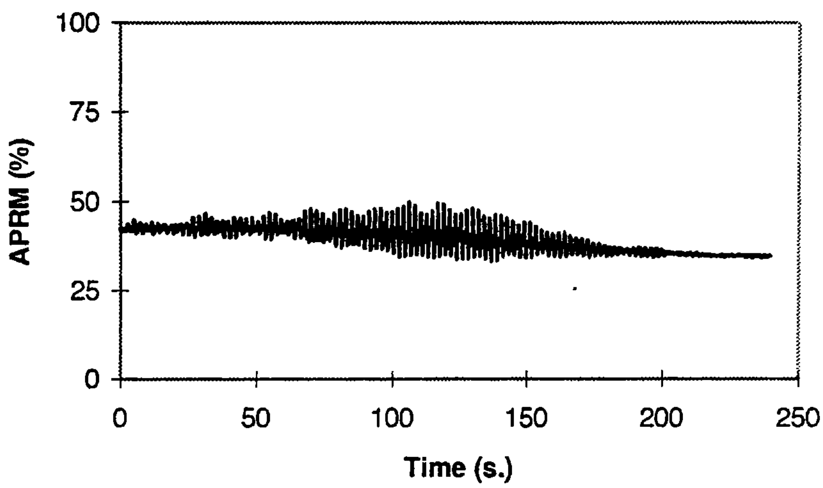

Figure 1.—Average power oscillation in an out of phase instability

Stability Scenarios

Generally three types of stability situations are considered in the analysis of BWRs:

- Plant stability: can affect to the complete NSSS and is mainly related to the control systems.
- Core stability: related to the coupled neutronic-thermohydraulics phenomena in the core.
- Channel stability: affects to individual fuel elements, related to the hydrodynamic stability of a two phase flow in a channel with constant pressure boundaries

Abnormal behavior of control systems can induce forced oscillations in the core. Control system settings are well adjusted during the startup period, and these type of incidents are typically originated by unexpected systems malfunctions. Oscillations will disappear once the exciting source is stabilized.

Hydrodynamic stability of fuel designs are verified by separate effect test on appropriate facilities. Some case of channel instability has been reported that was originated by an incorrect positioning of the fuel element.

Core stability, also referred to as neutronic, reactivity or power stability, is the one weare discussing in this paper. The physical mechanism leading to this type of instability has been widely described, basically, it is a coupling of density wave oscillations and neutronic feedback that, when their phase delays are properly tuned can lead to self sustained power and flow oscillations Experience in commercial BWRs shows that two modes of core instabilities have been observed:

In-phase instability, also called global or core-wide instability, corresponds to the case where neutronic oscillations are in phase along the whole core. APRM signals reflect the neutronic behavior of the core and the reactor protection system will automatically suppress the oscillations in the case that the scram setpoint is reached (typically 1 18~). Total core ftow oscillates with the same frequency (around 0,5 Hz.).

Out of phase instability, also called regional instability, in this case half of the core oscillates with a phase shift of 1 80Q from the other half. APRMs will average core power response and therefore they will only reflect the not canceled part of the local (LPRM) signals (due to the not perfectly symmetric location of LPRMs) and the frequency components that are in phase, once de oscillations are sufficiently developed to show up their non linearities. This is the situation where the capability of detection has been questioned.

The neutronic response in one or other of these modes is related to a) the separation between the corresponding (fundamental and subcritical) neutronic eigenvalues, and b) the dynamics of the Recirculation loop. It is also possible to have oscillations with both mode types coexisting at the same time, with different decay ratios.

It is usual to represent BWR operation, in terms of core power and flow, by means of the power/flow map (Figure 2). Although it is necessary to recognize that not all the parameters affecting core stability are represented in this map, the empirical and analytical experience shows that the stability conditions across this map can be represented by lines of constant decay ratio, with the shape of those represented in Figure 2, that shows how the region of potential instability is limited to the left corner of the map, far away from the region of normal operation (right corner).

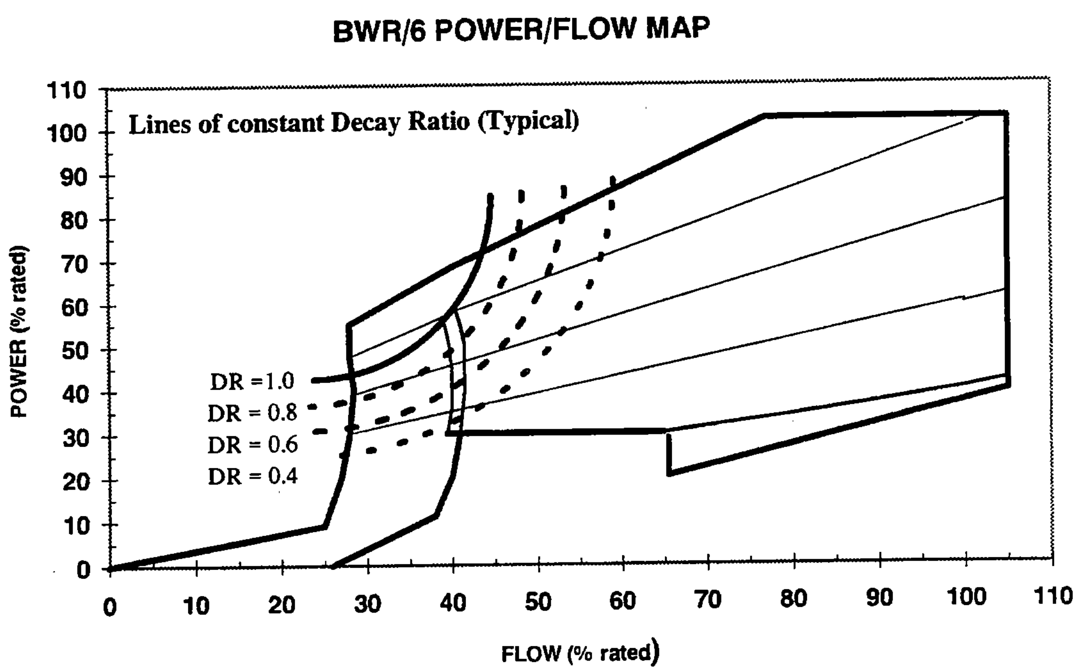

Figure 2.—BWR Power/Flow operating map

The region of unstable condition can be reached typically during maneuvering for power ascensions, being the shiR from low to high pump speed, one of the critical moments for developing oscillations, specially if feedwater temperature is low. Operators are aware of this situation and special attention is paid to APRM signals and associated noise.

Another situation where unstable region can be entered is as a result of an operational transient, for example the trip of Recirculation pumps,that will drive the core power down along the corresponding rod line in the map, towards the unstable region.

Finally, in the unlikely situation where scram is not available to resolve a transient, i.e.: an ATWS accident, the automatic actions to mitigate the consequences include the trip of Recirculation pumps, in order to decrease the core power to the minimum level, this action will lead to core conditions in the unstable region, where power oscillations may develop. Some calculations have shown very large amplitudes in this kind of accidents. Emergency procedures actions are oriented to anticipate the boron injection and to decrease the subcooling at the core inlet to minimize the oscillations.

Fuel behavior under core power oscillations

When neutronic oscillations come up, in a reactor core operating under unstable conditions, the fuel rods are subjected to an oscillatory behavior, intemally the power generated in the fuel pellets responds to the fluctuations of reactivity, oscillating with frequencies in the range of 0.4 to 0.5 Hz., whose amplitudes may eventually diverge until a limit cycle is reached due to the nonlinearities in the system dynamics. Externally, the rod conditions are defined by the heat flux through the cladding and the coolant conditions.

The heat flux response to power oscillations is governed by the fuel dynamics, which has a relatively slow response, therefore acting as a filter with a time constant in the range of 4 to 10 seconds, this value depending on the fuel rod thermal characteristics, (pellet and clad size, gap conductance, etc.).

Analyses of the dynamic response of fuel temperature have shown that cyclic power variations of +100% around 20 Kw/m, will result in heat flux oscillations of +10% and temperature oscillations of +5%, around the steady-state values.

Figure 3 shows the calculated relationship between power oscillations and the heat flux to the moderator, it can be seen that even if the power spikes may reach relatively high values, the fuel response, in terms of heat to the moderator, is considerably damped.

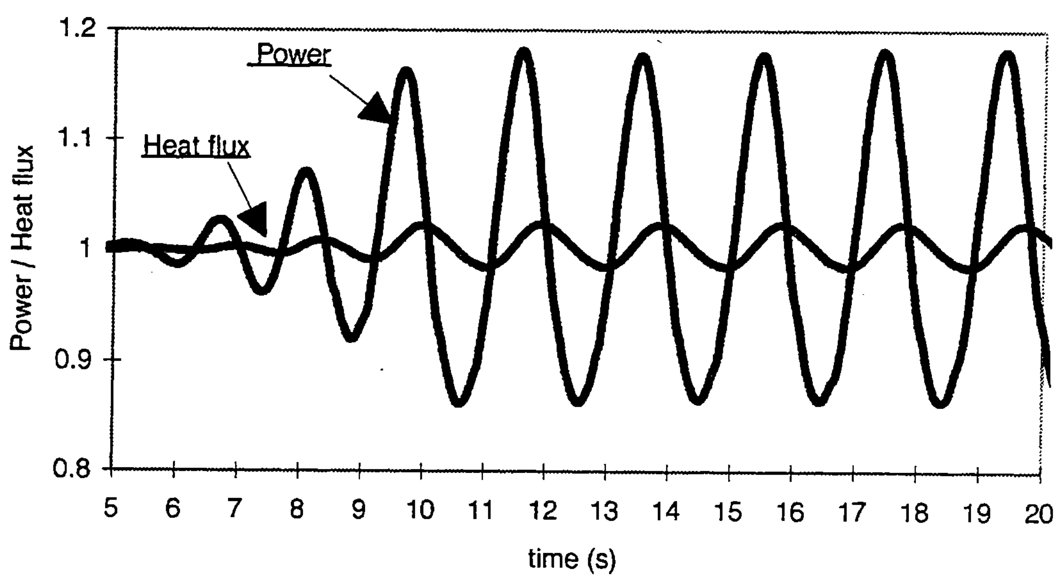

Figure 3.—Response of fuel to power oscillations

We are interested on the possible failure modes of BWR fuel under neutronicthermohydraulic oscillations. The main concerns are related to cladding temperature excursions as a consequence of potential dry-out, degradation of cladding mechanical properties, and cladding stresses induced by PCI and fission gas release.

The available experience of power cycling in fuel rods is related to operation in the Load Follow or Automatic Frequency Control modes. No evidence of increase in failure rates, in plants operating in the above modes, has been observed.

Investigations in Halden reactor, where irradiated fuel rods were subjected to power cycling, show no significant increment in gas release and that clad strains remain in the elastic range for power oscillations of the same magnitude of those observed during instability events. In the case of irradiated fuel with small gap, strong PCI effects were observed during test of ramp and cyclic power.

Fuel behavior related to dry-out is of primary concern to fuel integrity. Margin to dry-out is measured in terms of Critical Power Ratio (CPR), which is calculated using a boilinglength type correlation.

Evaluations of CPR performance during global power oscillations show that enough margin exist for oscillations in the range of 10%. It is necessary that neutron flux reaches a level in the order of 200% (well above the scram setpoint at 118%),to have the CPR on the hot fuel element below the Safety Limit

In the case of out of phase oscillations, the peak power level to reach the SLMCPR is much lower due to the strong effect of flow, that in this type of oscillations can have large fluctuations on an individual channel basis. Results of specific plant calculations show that power oscillations in the range of 170% (peak-minimum)/average, for the hot channel, may result in violation of the Safety Limit, while it would be reflected on APRM fluctuations in the range of 15% over the initial value. These type of oscillations would be detected in the plant by the unusual increase of APRM noise and/or LPRM upscale alarms.

But, for BWR fuel conditions, short dry-out does not necessarily mean damage. Due to the nature of this transient, passing below de Safety Limit of CPR means that in some location of the bundle transition boiling has occurred, and a sudden increase of the fuel cladding temperature will start, but considering the cyclic behavior of the thermalhydraulic variables, this situation will only be maintained during shorts periods of time, at the 'valleys" of the CPR evolution versus time, and thus the boiling transition will only last less than one second, and will be followed by the rewetting corresponding at the cycle portion where the nucleate boiling is recovered (CPR above Safety Limit). The embrittlement of the cladding is a function of the amount of overheating (clad temperature) and its duration, the change in mechanical properties will be small if clad temperature remains below 800 QC. It has been suggested, in different countries, that a failure criterion based on a maximum value for clad temperature for a limited period of time, could be more appropriate, specially for this type of transients.

In summary, no fuel damage is expected for power oscillations of moderate amplitude, like those that can develop in core global (in-phase) instabilities, as long as neutron flux scram will eventually terminate the transient.

Local (out-of-phase) oscillations may be undetected for some period of time and power oscillations may induce dry-out conditions in some specific fuel elements, even if dry-out conditions are reached, this will lead to short periods of rod surface temperature excursions with no fuel damage, unless rewetting is precluded by the large amplitude of oscillations. For high bumup fuel (small gap), significant PCI can be induced if operation is continued for long time.

Finally, the worse scenario is when oscillations are induced by an ATWS accident, calculations seem to indicate that if no action is taken, oscillations might grow to a level resulting in dry-out of some fuel rods and subsequent rewetting could be in jeopardy, some fuel damage could be expected.

Fuel design evolution and core stability

There are a number of fuel characteristics that affect directly to the core stability:

number of fuel rods
fuel rod diameter
2-phase vs. 1-phase pressure drops
gap conductance
void reactivity coefficient

Looking for a better neutronic efficiency and relaxation of thermo-mechanical duty, the natural tendency in new fuel designs, is towards increasing the number of fuel rods per assembly ($7 \times 7 \rightarrow 8 \times \times 8 \rightarrow 9 \times 9 \rightarrow 10 \times 10$) with smaller rod diameter.

These changes induce an increase in the two-phase pressure drop and a shorter fuel time constant, that can affect adversely to channel and core stability.

To overcome these unfavorable characteristics and preserve stability margins, fuel vendors need to include additional features with the objective of reducing two phase pressure drop, by new designs of spacers and upper tie plate or eliminating part of the fuel rod in the upper portion of the bundle (partial length rods), while increasing the single phase drop, by adjusting the lower tie plate design.

Fuel rods with smaller diameters, when under power oscillations, will have quicker response and therefore larger amplitudes in temperature and heat flux.

Advanced fuels allow more aggressive core designs, with higher peaking factors, that could affect adversely to core stability margin. Core kinetics parameters like void reactivity, which plays a major role in core stability, need to be considered at the specific cycle and core conditions.

Tests made in experimental facilities (Karlstein, FRIGG, ATLAS, NFI, BEST) have been used to analyze the hydraulic stability evolution of different fuel types, and to investigate the influence of specific design features, such as the flow communication openings in the ABB Atom SVEA design (assessed in the FRIGG loop) and the partial length rods, introduced in the GE new products (BEST facility). However, these separate effect test serves only to study the hydraulic stability performance of the fuel elements. Plant measurements or calculations (with well qualified codes and models), where core state parameters, spatial effects, and specially neutronic feedback (that conforms the real link to the hydraulics of the channels) are taken into account, would be the proper approach to verify the tendency of stability margins when a new fuel design is introduced in the core.

Actual and Future trends

Many organizations and investigators are active in this field, with an important amount of experience accumulated, and significant advances can be expected in the near future within the different approaches followed in the areas of analysis, monitoring and design. The followii~g topics are among those with greater interest at the present time:

3D kinetics codes

Efforts are being made by developers and users of codes to implement 3D kinetics capabilities for the analysis of the response of subcritical modes in the cases of out-ofphase oscillations in large reactors

Codes qualification

Confidence is coming back to analytical simulations as long as population of code benchmark with plant data is increasing. There is no doubt that programs like the one launched by the OECDE/NEA Nuclear Science Committee, to benchmark codes against stability data from Rinhals 1, as well as the initiative of a group of European utilities to quantify the uncertainty in predictive calculations, provide a valuable contribution to support this confidence.

Stability Margin predictors

Predictive capabilities are being linked to the on-line plant monitoring computers which, based on frequency domain codes, can help the operator to know the expected evolution of the core stability margin resulting from any anticipated changes in core conditions Some core designers include stability predictions as part of the core design practice to verify that it caabe operated with adequate stability margin.

Stability Monitors and operator aids

Probably the weak points of stability monitors, base don noise analysis, are the response time and the capability to identify an incipient out-of-phase instability when may be hidden (same frequency) under the core response to the fundamental mode.

Instability detecting instrumentation, based on direct LPRM check, capable of detecting global and regional oscillations and provide alarm, or even automatic action, has been developed as part of the BWO-OG Long term solutions.

Potential of large oscillations

Care should be taken when extrapolating to extreme severe conditions, where some models can predict large power oscillations that could prevent clad rewetting and the consequent sustained dry out. Qualification of the models involved and quantification of uncertainties are a difficult task due to the lack of experimental data.

Performance of Long Term Solutions

Long term Solutions developed by the BWR Owners Group are close to be implemented in the plants. The future experience in the plants that have selected the different options will show their real performance and the capability of avoiding instability events while maintaining operational flexibility and plant availability.

Experience of new advanced fuel designs

The challenge of advances in fuel designs,while maintaining stability margins,needs to be assessed and verified by plant measurements.

Core management strategies

Increase in discharge bumup and core loading strategies can lead to core design with hard zones where significant gradients in neutronic characteristics can be found. Analysis of the expected behavior along the cycle of these cores, with respect to stability, can avoid operational surprises. Mixed cores, where different fuel designs coexist, must be carefully designed to assure neutronic and hydraulic compatibility

References

Additional information related to this subject can be found in the recent State of the Art Report (SOAR) on BWR Stability, that was conceived inside the OCDE/CSNI THSB Task Group, and is close to its official publication. This report is probably the best compendium on the subject of BWR Stability and has been prepared by a group of Lead Authors, among the most relevant scientists active in this field, under the leadership of the Editor Prof. F. D'Auria from University of Pisa, Italy.

Update of High Burnup Fuel Issues

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1. Introduction

Up to the end of the eighties the cores of the Light Water Reactors PWR and BWR appeared to be well protected against the potential consequences resulting from the design basis reactivity accidents (RIAs). The knowledge of the fuel state at burn-ups beyond 50 Gwd/t and recent results from RIA experiments with high-burnup fuel seem to indicate that the safety criteria which have been formulated on basis of fresh or low-burnup-fuel experiments may not be applicable to high burnup fuel rods.

First alarming results were obtained in the NSRR experiments JM4 and JM5 with failures well below the regulatory level and with clear indications for a PCMI failure mode, characteristic for high-burnup fuel and not observed in the earlier fresh and low burnup tests. These results have been confirmed in the tests with industrial fuel, REP-Na 1 in CABRI and HBO-l and HB0-5 in NSRR. In all these tests a direct correlation with high-burnup phenomena could be established and in addition, the previously not well understood test PBF-CDC 859 could be integrated into this family of failure tests.

The internationally agreed failure mode is the hydride assisted PCMI failure. Hydride accumulations occur either as local blisters or as azimuthally regular high radial concentrations on the outer rim of the cladding.They might be accidental and untypical for real fuel as in JM4, JM5 and CDC859 or "natural" as in HB0-1, HB0-5 and REP-Na 1.

Other CABRI experiments at lower corrosion state, in particular REP-Na 4, 5 and 6 (see table l) and also HBO-3 demonstrate a remarkable mechanical resistance against the severe loading resulting from the rapid power excursions.

All the CABRI or NSRR experiments however suffer from unrepresentativity in the experimental conditions. Uncorrect clad temperatures and untypical system pressure and, in the case of CABRI, sodium cooling instead of water, do not allow a direct transposition to the reactor case.

The performance of computer simulations with the aim to realize this transposition is limited and finally not valid due to the unsufficient knowledge of behaviour models and material properties.

It is to be concluded therefore that representative inpile experiments are needed in order to investigate the behaviour of high-burnup fuel submitted to fast power transients, to determine precisely the safety margins and to evaluate the potential consequences resulting from post failure events.

2. Present state of Knoluledge

The original data base of RIA test results has been enriched by the recent results from NSRR and CABRI.All the available relevant data are plotted in the well known diagram of the maximum mean enthalpy reached before failure as a function of the burnup of the tested fuel rod (Fig. l).

Table 1: CABRI REP-Na test characteristics and major results

Test	Test-Rod	Pulse (ms)	Energy at End of Pulse(cal/g)	Corrosion (μ)	RIM (μ)	Results and Remarks
Na-1 (11/93)	Grav5c span 5 4.5%U 64GWd/t	9.5	110 (at0.4s)	80 **important initial spalling**	200	- **brittle failure** at30cal/g - hydride accumulations - **fuel dispersion:6g** including particles other than RIM - **pressure peaks** in sodium
Na-2 (6/94)	BR3 6.85%U 33GWd/t	9.5	211 (at0.4s)	4	-	- **no rupture** - $\Delta\phi/\phi$(max): 3.5% mean value - FGR/ 5.5%
Na-3 (10/94)	EDF 4.5%U 53GWd/t	9.5	120 (at0.4s)	40	100	- **no rupture** - $\Delta\phi/\phi$(max): 2% max - FGR/ 13.4%
Na-4 (7/95)	Grav5c span 5 4.5%U 62GWd/t	#60	95 (at1.2s)	80 no initial spalling	200	- **no rupture** - **transient spalling** - $\Delta\phi/\phi$(max): 0.4% mean value - FGR/ 8.3%
Na-5 (5/95)	Grav5c span 2 4.5%U 64 Gwd/t	9.5	105 (at0.4s)	20	200	- **no rupture** - $\Delta\phi/\phi$(max): 1% max - FGR: 15.1%
Na-6 (3/96)	**MOX**,3c span 5 47 GWd/t	35	126 at .66s 160 at 1.2s	40	-	- **no rupture** - $\Delta\phi/\phi$(max): 3.2% max - FGR: 22%

Remaining REP-Na tests to be performed

Na-7 (11.96)	**MOX**,4c span 5 56 Gwd/t	35	ca 130	-	-	to be performed
Na-8 (06/97)	Grav 5c span 5 4.5at% >60GWd/t	60	ca 100	-	-	cladding presenting spalling to be performed
Na-9 (06/97)	**MOX**,2c span 5	35	ca 150	-	-	to be performed

This diagram has been used in the past for delimiting the safe area from the failure region and to define safety criteria which represented guidelines for the definition of control rod worth and for rod design.

At high burnup this procedure is not longer valid due to the irradiation induced changes which are not sufficiently defined by the burnup only and which can be different over a wide range depending on material choices and irradiation conditions:

- gap closure
- clad corrosion phenomena (hydrogen pickup and oxide spalling)
- fission gas release and retention
- plutonium built up and redistribution (RIM formation).

These transformations of the fuel produce in addition a strong sensitivity to the parameter of the power rise rate, determined by the pulse width of the reactivity excursion.This parameter is changing over a wide range for the various experiments performed in different test facilities.

Therefore the diagram presented in fig.l has to be considered with crititical attention and a large uncertainty range must be envisaged for the enthalpy to failure at a given burnup due to different test fuel and different test conditions.

2.1. Fuel behaviour

Fast transient heating of high burnup fuel produces new and significantly different effects compared to fresh or low burnup fuel.

The closure of the initial fuel/clad gap, resulting from the creep-down of the clad material under the effect of the PWR system pressure, leads to the immediate built up of high contact pressures when the fuel temperature rises rapidly. In addition to the thermal expansion, transient fuel swelling occurs when retained fission gas rises to high pressures inside the bubbles and porosities. Grain boundary gas produces fuel fragmentation. The working gas increases finally the internal pin pressure when it is released into the free volume of the fuel rod.

Beyond 45Gwd/t burnup a specific structure is built up progressively at the outer rim of the fuel. Neutron resonance capture phenomena of U-238 are the reason for the locally high plutonium production which increases the fission rate and leads to locally high burnup and associated high fission product concentrations. In this region a large temperature peaking can occur when the transient heating is close to adiabatic due to a very narrow power pulse. When the pulse width inreases, the thermal disequilibrium disappears as a result from heat conduction and because the RIM is very narrow (ca 100 microns).

The clad temperature and the system pressure have a strong influence on these high burnup phenomena which determine the global response of the fuel rod to the accident conditions.

2.2. Cladding behauiour

High burnup PWR cladding is characterized by severe and typical corrosion phenomena. Waterside corrosion produces the deposition of a steadily increasing oxide-layer on the cladding surface and simultaneously hydrogen dissolution in the metallic part with a steep concentration gradient over the clad thickness. These corrosion effects lead to a brittle mechanical behaviour of the cladding material and, in detail, to the shift of the brittle to ductile transition to higher temperatures.

The incomplete data base of transient mechanical properties of high exposure, severely corroded LWR cladding and the difficulty to define mechanistic failure criteria is one of the reasons which make test interpretation and behaviour prognostics difficult. An important test programme named PROMETRA is presently performed in France, in the frame of the CABRI programme, with the aim to improve the state of knowledge.

Another aspect of high burnup clad behaviour is resulting from the observation that increasing oxide thickness beyond around 80 microns or more may lead to the phenomenon of oxide spallation e.g. local scaling-away of the oxide layer.Under nominal operation conditions spallation produces cold spots and, when this fuel undergoes a still prolongated operation, hydrogen accumulates in the cold region and hydride blisters are formed.

The mechanism of spallation is not understood in detail, load-follow operation might stimulate this evolution. The present state of knowledge leads to postulate that at high burnup an unknown population of fuel rods of the core presents more or less advanced spallation and eventually associated hydride blisters (or sun-burst).

Under the rapid transient loading of the RIA scenario the blisters act as crack initiators as observed in CDC859,JM4,JM5 and REP-Na 1.

In the tests HBO-1 and HBO-5 no initial spallation was observed and the clad failure is to be attributed to a steep hydrogen gradient in the clad wall and to the general embrittlement of the cladding which in addition is exacerbated because of the untypical low clad temperatures in NSRR.

Furthermore the narrow power-pulse of NSRR (FWHM ca 5 ms) might be penalizing and unrealistically conservative compared to the broader pulse (FWHM ca 35 ms) which would occur in a reactor accident.This hypothesis however has no experimental confirmation at present time.

As a conclusion it is to be stated that the major risk for the occurrence of low enthalpy failures is related to the hydrogen embrittlement of the cladding and in particular to the presence of hydride accumulations.

Here again more realistic experimental conditions close to the PWR conditions might change significantly the experimental results which are presently obtained under untypical conditions.

2.3. Transient thermohydraulics

The evolution of the cladding temperature history following a given, fast RIA transient, is essentially determined by the clad-to-coolant heat-transfer behaviour. This behaviour is depending on the nature, the pressure and the flow of the coolant, it is also sensitive to the channel geometry. Furthermore it might be significantly influenced by the state of the cladding surface during the transient sequence.

Transient spalling phenomena have been observed in several CABRI tests,they would influence most probably the heat exchange close to the critical heat flux.

Neither in CABRI nor in NSRR the thermalhydraulics can simulate correctly the reactor situation presently.

In the sodium loop configuration of CABRI, only during the very short, close to adiabatic time period at the beginning of the transient, the clad temperature is correct. If a clad failure occurs in this short time interval, the result is rather pertinent even if eventual post-failure events can not be turned to account. This was the case for REP-Na 1. The absence of failure in all other CABRI tests is not conclusive, because overcooling under sodium after the first adiabatic PCMI loading phase and the impossibility to reach DNB, prevents the clad temperature from increasing like in PWR conditions.The survival of the test pins in some of the tests,in particular REP-Na 4 might be explained by the high resistance of the clad due to its too low temperature.

Representative global experiments under PWR test conditions are urgently needed.

Analytical experiments are performed in the PATRICIA loop in France with the aim to determine the transient heat transfer correlation.It is expected that the results of these tests will give a better physical understanding of the heat exchange during rapidly changing high heat fluxes.The influence of the clad surface, typical for high burnup fuel, originally one of the test goals, cannot be simulated in these tests.

3. Conclusion

When high burnup fuel is subjected to the rapid power transients which are typical for the conditions of the design basis reactivity accidents of LWRs, a higher risk for failure and post failure fuel dispersion must be expected compared to fresh or low burnup fuel.

Fission gas driven fuel swelling and the RIM effect increase the transient loading of the cladding while corrosion phenomena decrease its mechanical resistance.

The recent results from experiments in CABRI and in NSRR give indications that beyond the burnup level of 50 GWD/t the present fuel rod design is extremely vulnerable to the rapid power pulses of the RIA accident scenario. The hydrogen embrittlement of the ZIRCALOY cladding, in particular the risk of local hydride accumulations (blisters), seems to be the most preoccupying aspect.

These experiments however suffer from lack of representativity. Clad temperatures which are typical for the reactor situation cannot be achieved. The too low temperatures can be used as arguments both for postulating excessive conservatism as for the opposite. Indeed, cold cladding produces higher contact pressures but at high temperature the cladding looses rapidly its mechanical strength.

The high system pressure in the reactor might influence the fission gas behaviour and most probably the failure mode and the post failure events which include the risk of energetic fuel coolant thermodynamic interaction.

The available test facilities NSRR and CABRI will not allow to resolve the open questions.

A project study has been performed in order to investigate the possibility to install into CABRI a pressurized-water-loop and to quantify the achievable performance characteristics of this new facility.

The result of this study allows to demonstrate that all requested technical requirements can be achieved:

— representative thermal-hydraulics (pressure,temperature,flow)

— large flexibility for the power ramp-rate (adjustable pulse-width)

— high energy deposition at high burnup (more than 100 cal/g at 65 Mwd/t).

In this facility it will be possible to perform experiments which will allow to define new safety criteria and to test improved fuel at high burnup and under representative conditions.

IPSN

RIA data base as a function of burnup showing fuel rod failures by high temperature and PCMI mechanisms.

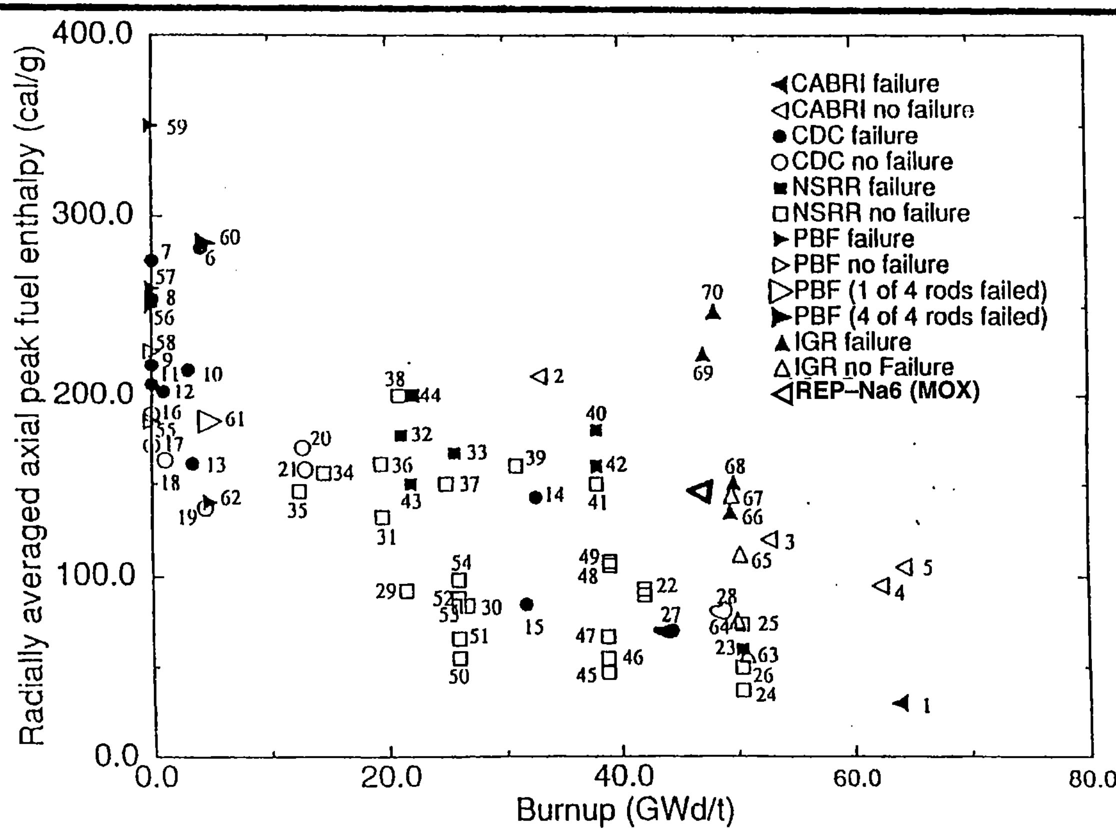

FIGURE 1	The apparently broad RIA data base is confusing because listed experiments differ in important parameters which are not accounted (rod design, t_o state, pulse-width

From R.K. McCardell and R.H. Hobbins (INEL)
ACRS Meeting 10/4/96

TECHNICAL SESSION I

Operating Experience Identified Core Performance Issues

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Introduction

The economic incentive for optimizing core design and the highly competitive markets for supply of fuel and control rods have resulted in extended-length fuel cycles and extended-burnup fuel applications that place increasing demands on the design performance of fuel and control rods. Design changes to gain these advantages have contributed to operating performance events that may have been avoidable. Steady improvement in the operating performance of fuel and core components has perhaps lulled the industry into complacency that has permitted design changes to be implemented with incomplete fuel testing or developmental testing that would have delayed the marketing of new fuel and control rod designs. This paper cites examples of apparent design deficiencies revealed by operating experience and discusses other design proposals that could lead to similar problems. The U.S. NRC Core Performance Action Plan to improve regulatory oversight and contribute to continued excellence in core performance behavior is described.

Fuel operating experience

Operating experience with Zircaloy clad fuel in both boiling-water reactors (BWRs) and pressurized-water reactors (PWRs) in the United States has been generally excellent. The incidence of fuel failures, defined by loss of cladding integrity, has been very low and has gone down steadily as the root causes of generic problems are identified and corrected. The number of defective fuel assemblies per gigawatt electric installed nuclear power capacity has declined from a range of 15-30 in the 1970s to 5-10 through the mid-80s, and only slightly more than one BWR assembly and three PWR assemblies failed per year, on average, from 1986 to 1996. Some of the more recent failure mechanisms that were mitigated by corrective actions were:

1. Undetected Manufacturing Defects

   Leaker fuel rods were placed in service in BWRs when inspection equipment failed to detect defects in the end cap welds (R-1). In some cases, secondary hydriding caused by coolant ingress led to severe damage to the fuel rod and loss of fuel material.

2. Crud-induced Local Corrosion (CILC)

   BWR water chemistry problems associated with the condenser tubing alloys lead to corrosion relating to crud deposition on the fuel cladding surface (R-2).

3. Debris Fretting Failures of PWR Fuel

   The failures were mitigated by designing debris-filtering fuel-inlet nozzles (R-3). However, cross-flow conditions aggravated by the debris filter bottom nozzles in some mixed-core configurations are believed to be responsible for fretting failures due to vibration at the lower grids.

4. Vibration-induced Grid-to-Rod Fretting Failures at Low-pressure Drop mid-grid Locations of PWR Fuel

   Defects associated with specific fuel designs have been attributed to flow conditions for fuel assemblies adjacent to the core baffle. This problem, resulting in fuel failures at several reactors, could have been avoided by more extensive vibration testing under the full range of flow conditions that may exist in the reactor (R-4).

Impact of high burnup and longer cycles on fuel and control rod performance

The improvement in fuel performance, as measured by the incidence of fuel failures, has been accomplished despite the industry trend to longer operating cycles and higher burnup fuel. Batch average discharge burnups in U.S. BWRs have increased from 25 GWd/MTU to 33 GWd/MTU during 1990-1995, and from 35 to 42 GWd/MTU in PWRs during the same period. However, recent operating experience has revealed that the design of fuel for higher burnup and longer operating cycles has contributed to new issues and new failure mechanisms. A noticeable increase in the severity of secondary hydriding failures with loss of fuel material from leaker fuel rods has been attributed to grain structure refinement for greater resistance to high-burnup corrosion. High-fluence exposure of control rods has revealed design deficiencies leading to such problems in BWRs and PWRs as, for example, cracks in the sheath or cladding, poison leachout, hydriding, swelling, and corrosion.

Fuel designs for extended cycles require higher enrichment and higher burnable poison loadings or soluble poison concentrations to overcome the beginning-of- cycle (BOC) reactivity. The higher soluble poison concentrations and associated water chemistry adjustments have led to corrosion problems with the fuel, and crud problems which have impaired the scram performance of the control rods. Extended core residence times have also been a factor in corrosion performance and have not been adequately evaluated. Examples of problems related to high burnup and cycle length follow.

1. Bowing of Thimble Tubes in PWR Control Rods

In early 1996, during normal full-power operation, a PWR plant experienced difficulty maintaining adequate essential cooling water flow because of freezing conditions caused by cold weather. Plant personnel decided to shutdown operations to avoid violating facility license conditions. Not all of the rod cluster control assemblies (RCCAs) functioned as expected during the manual reactor trip. Of the 53 RCCAs, 5 failed to fully insert. When these five RCCAs paused at various core elevations and then continued to insert, control room personnel initiated emergency boration of the reactor coolant. Following the reactor trip, the core was shut down prior to full insertion of the slow RCCAs.

The licensee examined all the affected RCCAs, and discovered that excessive fuel assembly growth had bowed the thimble tubes, impeding control rod insertion. These fuel assemblies were irradiated to burnups beyond 40,000 MWd/MTU. Similar problems with rod insertions were noted at two other PWR plants shortly before and after the described event. At two of the three affected plants, licensees had the same type of fuel design (R-5). The root-cause evaluation by the fuel vendor attributed the rod insertion malfunctions for the described event to excessive compressive loads on the fuel assembly guide tubes leading to thimble tube distortion. The excessive loads were caused by accelerated irradiation growth of the fuel assembly associated with oxide accumulation. Distortions are observed only in high-temperature plants and in those high-burnup assemblies that have certain types of power histories. The root cause of the rod-insertion difficulties at the other plant is still under evaluation (R-6).

2. Distinctive Crud Pattern on Fuel Rods

During its refueling outage, a PWR licensee discovered failed fuel rods and cladding wall thinning associated with a distinctive crud pattern on the rod surface. The pattern resembled marbling. The failed rods were once-burned, high enrichment, and located at high relative power regions. The licensee speculated that the distinctive crud pattern came from plant operations with a large amount of soluble boron and a small amount of associated lithium concentrations in the water chemistry for an extended cycle length (R-7). Although this type of distinctive crud pattern had not been observed previously, it occurred shortly afterward in a second PWR plant that had similar water chemistry.

3. Power Offset Anomaly

A negative core axial offset power distribution occurred in a few PWR plants which indicated an axial power distribution shift toward the bottom of the core. The axial offset anomaly was attributed to the accumulation of porous, boron-enriched crud deposits on the fuel in the upper half of the core. The increased boron concentration in the crud was believed to have come from reaction mechanisms within the otherwise typical crud deposits (R-8). In some cases, the anomaly was associated with high boron concentration and high- burnup fuel. The effect of this anomaly on safety analyses for the operating cycle is a concern.

Fuel and core performance action plan

In late 1994, the NRC began a new plan to monitor, review, and improve the operating performance of fuel and core components. The intent of the plan is to provide a more proactive regulatory review stance with greater emphasis on identifying and correcting weaknesses associated with generic performance problems before they cause operational events. The initial plan contained a description of observed core performance issues, including preliminary causal factors, historical activities, and proposed actions to evaluate the issues and to develop recommendations to improve core operating performance. The plan provided for performance-based inspections of domestic fuel vendors and for evaluation of selected licensee/plant organizations involved in reload core designs interfacing with the inspected fuel vendor and from high- visibility events and issues. Concurrently, operating events and vendor fabrication and quality problems are being surveyed and catalogued more carefully in an effort to identify and correct common weaknesses associated with generic issues. The inspection activities completed to date have involved all five of the domestic reload vendors and covered approximately 25 plant/licensee reload analyses that were evaluated for continuing licensee inspection efforts. In addition, the plan provides for the identification and documentation of core performance problems and root cause evaluations. Recently the plan was expanded to evaluate the results of the preceding inspection activities for use in various regulatory actions (e.g., generic communications and guidance for regional and resident inspectors); to evaluate lead test programs for early identification of performance problems; and to conduct a workshop to present regulatory concerns and obtain feedback from industry groups and international participants. The workshop was scheduled for October 24 and 25, 1996, in Bethesda, Maryland, following the 24th Water Reactor Safety Information Meeting.

Issues and weaknesses addresed by inspections

The completed inspections of vendors and licensees have noted some common weaknesses that are often associated with similar issues. Much attention has been given to mixed-core licensee/vendor interface issues involving clear definition of licensee and vendor core design and quality assurance responsibilities for mixed-fuel designs in transition cores. Other mixed-core design issues targeted for inspection attention are:

- thermal-hydraulic compatibility of mixed-fuel designs,
- application of critical heat flux correlations to "other vendor fuel" when proprietary test data on the other fuel are not available, and
- safety analyses and on-line monitoring of thermal margin for transition mixed cores with fuel from different vendors.

The vendor inspections have also emphasized identifying weaknesses in the design analysis and testing of new fuel and control rod products. Concurrently, operating events and vendor fabrication and quality problems are being surveyed and catalogued more carefully in an effort to identify and correct common weaknesses. Examples of problems identified after fuel was placed in service follow:

- inadequate sensitivity of nondestructive test equipment leading to non-detection of defective end cap welds and leaker fuel rods placed in service,
- misloading (incorrect enrichment) of BWR fuel bundles
- misoriented BWR fuel bundle loaded in accordance with current vendor generic guidance,
- vibration problems encountered with PWR fuel and identified in reactor service,
- mis-loading of asymmetric burnable poison rods in PWR fuel leading to core misloading in the reactor,
- inappropriate application of departure from nucleate boiling (DNB) correlation to PWR fuel because of incomplete testing preceding reactor service,
- PWR fuel assembly burnup-related bowing resulting in degradation of rod scram times due to interference of the bowed control rod thimble tubes, and
- faulty pellet-sintering process resulting in delivery of many fuel rods containing out-of-specification low-density fuel pellets.

Conclusions

Many of the operating performance failures of fuel and control rods are due to inadequate preservice testing of new design features or inadequate evaluation of the component's capability to withstand more severe service requirements. "Minor" design changes are often incorporated without paying attention to the sensitivity of the vibration behavior, thermal margin, and other characteristics of the fuel assembly to the modified design features. Lead test assemblies are often employed without complete and well defined test programs, and sometimes without full commitment of the user licensee to obtain desired inspection data because it could have adverse economic impact on operations. Better regulatory oversight through improved inspection programs is intended to correct these weaknesses.

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Nuclear Fuel in France: an ever changing world - most recent safety concerns of DSIN

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1. Introduction

The design and management of nuclear fuel have seen many changes in France the last few years. Such changes, brought about by the operating organisation, are essentially based on the increase of the fuel discharge burn up before unloading, the adoption of MOX fuel in reactors, and the improvement and diversification of nuclear fuel products (introduction of fuels of foreign origins, i.e. German, Spanish and Swedish).

The French safety authority is not, a priori, opposed to these changes fostered by international competition in the nuclear fuel market as well as by the willingness of the operating organisation to optimize its industrial tool to reduce costs and improve operational flexibility. But it has to ensure that, at each step, identified safety problems are being analysed and resolved appropriately.

Developments or changes obviously bring out new safety concerns. They can be in a latent state for a time and be revealed only a few years after the changes have been made. A few examples are the behaviour of high burn-up fuel under normal and abnormal conditions (loss of coolant accident -LOCA- and reactivity initiated accident -RIA-), the comprehensive effect of irradiation over the fuel assembly structure (bowing) and control rod drop, or the storage in the reactor pool of high enriched fuels.

Although the examples quoted are chiefly connected with reactor operation, new safety concerns arise in the whole fuel cycle : better care needs to be taken in the handling and transporting of high burn up fuels than fresh UO2 fuels, the use of MOX fuel demands the reinforcement of the radiation protection measures taken at all steps of the fuel cycle (manufacturing, handling during fabrication and in the reactor, transportation, reprocessing and disposal), and imposes a longer cooling period than UO2 fuel.

Such changes might also bring out new concerns as regard as their compatibility with current standards and regulations. As an example, provision must be made for the use of MOX fuel in a reactor in the ministerial order creating the nuclear installation housing the reactor. As this is not the case for Chinon Nuclear Power Plant, reforming the corresponding orders is thus necessary to be able to use

MOX fuel in the Chinon reactors (this administrative procedure which has started in 1996 is addressed in IV.3).

The present paper tries to roughly depict the most recent changes in management and design that affected the nuclear fuel loaded in French pressurized water reactors. It also intends to address some of the main current safety and regulatory issues associated with these changes.

Before embarking upon this subject, the following chapter attempts to give the reader a comprehensive view of the main regulatory actions of the French nuclear safety authority, the DSIN (Nuclear Installations Safety Directorate), supported by the Institute for Nuclear Safety and Protection (IPSN), in the field of nuclear fuel, emphasizing its responsibilities versus those of the operating organisation in particular where the evolution of fuel is concerned.

2. How the french nuclear safety authority faces fuel developments

Before getting into the subject, it is worth briefly reiterating the general organizational system implemented in France to control nuclear safety.

2.1. General organizational structure of the control of nuclear safety in France

Prior to refering to the way the French nuclear safety authority deals with changes in nuclear fuel, we should have a general understanding of the government authorities organizational structure for checking technical nuclear safety, which is one aspect of the regulatory action in this field (see figure 1).

The goverment bodies in charge of nuclear safety

The ministry in charge of industry and the ministry in charge of the environment are responsible, within the French government, for matters relating to the safety of civil nuclear installations. The main bodies involved in nuclear safety and working for these two ministries are the High Council for Nuclear Safety and Information (CSSIN), mainly in charge of supplying information relating to nuclear safety to the population and the media, the Interministerial Commission for Basic Nuclear Installations, which is consulted by the two ministries mainly for basic nuclear installation creation and modification licences, and the Nuclear Installations Safety Directorate (DSIN).

The DSIN is a specialized department of the ministry in charge of industry which also works for the ministry of the environment. Its responsibilities are described by the modified ministerial order n.° 63-1228 of December 11, 1963 and by ministerial order n.° 73-405 of March 27, 1973. Its principal tasks are to handle licensing procedures for basic nuclear installations, to organize and head the surveillance of such installations during all stages of their life (construction, operation, decommissioning), and to draw up and monitor the application of the general technical regulations.

The DSIN also pursues research and development work in the field of technical safety which is carried out by organizations attached to the ministry in charge of industry, particularly the CEA (Atomic Energy Commission) and the French electrical power supplier and nuclear power plants operator, Electricité de France (EDF).

The technical support bodies of the French nuclear safety authorities

In making its decisions and analyses, the DSIN, is mainly assisted by the IPSN (Institute for Nuclear Safety and Protection). The IPSN carries out technical safety analyses making it possible to assess the provisions made by the operating organisations of nuclear installations. The DSIN can also ask the advice and recommendations of groups of experts such as the Standing Groups and the Standing Nuclear Section of the Central Commission for Pressure Vessels.

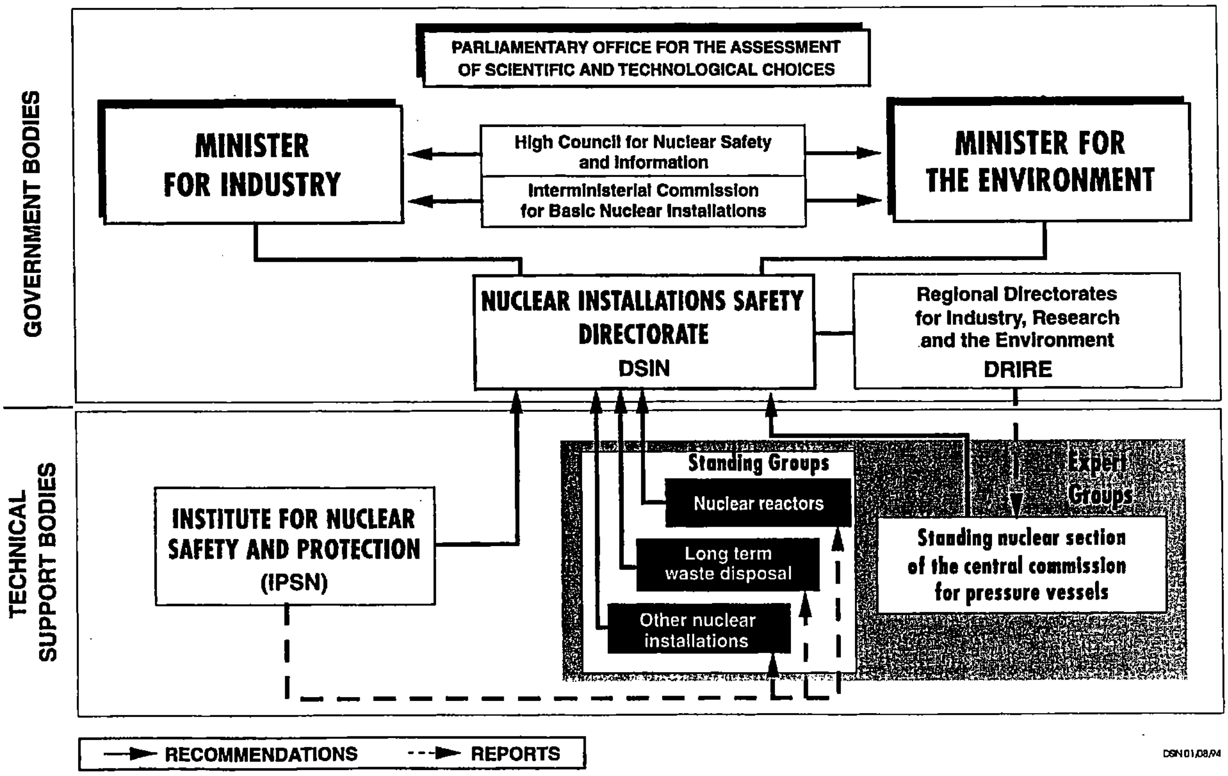


Figure 1

The scope of the French nuclear power program and its development during the eighties led the governmental safety authorities to create additional means of checking technical safety. As these means were strenghtened, the need was felt to devolve the monitoring of nuclear installations to the regional directorates for industry, research and the environment (DRIRE) so as to benefit from the efficiency resulting from geographical proximity to the installations.

2.2. Nuclear safety control principles in the field of nuclear fuel

The former concisely described the general organizational system implemented by the French governmental authorities in order to control nuclear safety. We will now focus on its applications in the more restricted field of nuclear fuel.

As a general rule, the DSIN intervenes at all the stages of the installation's life and establishes general safety objectives (see figure 2). The operating organisation proposes the technical means enabling them to be reacted and justifies such means. The DSIN checks, with the aid of its technical supports bodies (mainly the IPSN and the Standing Groups), the adequacy of these means in relation to the objectives, and if necessary makes recommendations before approving them. Then, the operating organisation implements the approved dispositions. Finally, the DSIN verifies, particularly during inspections, the proper implementation of these dispositions and draws conclusions. The control methodology implemented by the safety authority is based on sample examinations of the adequacy of implemented dispositions in relation to current standards and regulations. The figure 2 dipicts such a distribution of responsabilities.

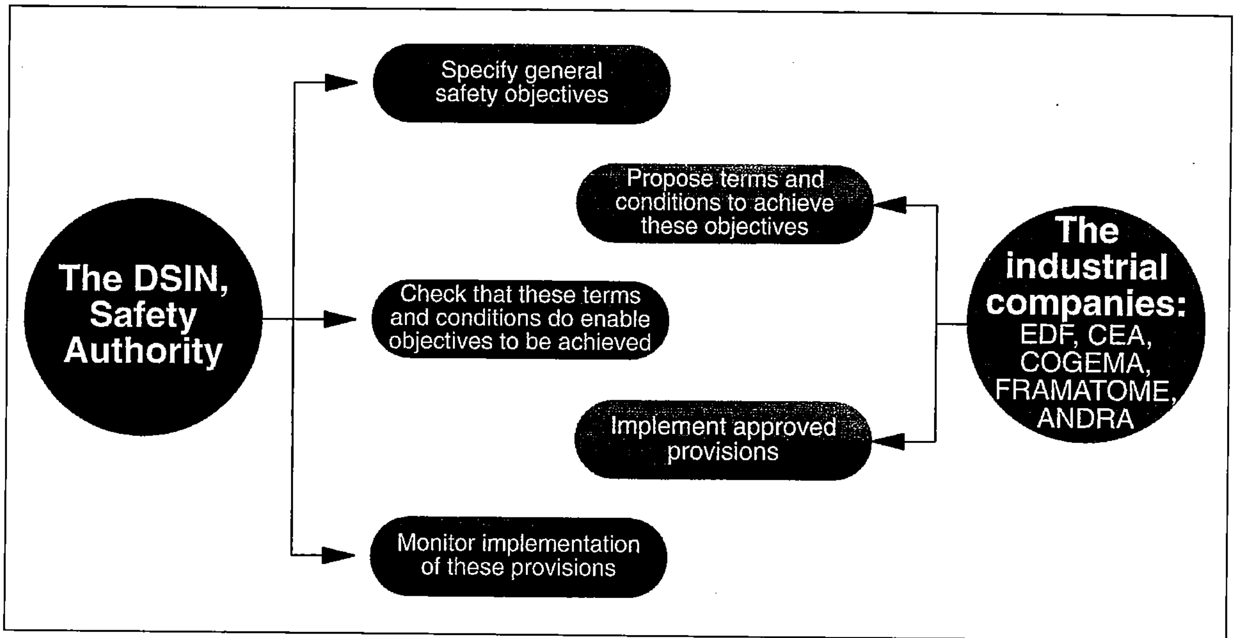


Figure 2

2.2.1. Safety objectives in the field of nuclear fuel

Major safety objectives applying to the fuel used in French nuclear power plants are specified in the ministerial order authorizing the creation of the nuclear installation (also called a construction license decree).

From these, we can quote that the operating organisation must :

- closely watch the core coolant water radioactivity in order to detect possible fuel leaks;
- maintain conservative margins for fuel integrity in all plausible situations;
- ensure that vibrations and other mechanical phenomena do not affect the integrity of the reactor internal equipment;
- ensure that fuel handling and storage is being carried out in such a way as to avoid criticality risks and to limit the risk of fuel bundle drop and overheating.

Apart from these general instructions, more specific licensing guidelines are specified in the Fundamental Safety Rules (French acronym RFS) relating to the design, manufacturing, and control of the fuel bundle assemblies (RFS n°V.2.e of December 28, 1982 and its revisions). This safety rule is one of the various Fundamental Safety Rules which have been issued by DSIN and that constitute a set of major regulations on various technical subjects for ensuring the safety of nuclear installations, especially the pressurized water reactors.

Other specific prescriptions are made by DSIN in the form of letters. Regarding reactor refuelling, DSIN has established the following rules:

- for each reactor, a safety assessment must be carried out at each refuelling outage and must be sent to the DSIN before the reactor goes critical. Furthermore, a document describing the startup tests has to be transmitted as well within the same time limit;

- concerning the approval of such documents by DSIN, two cases have to be considered:

  a) the new fuel cycle is "standard": the cycle, except for special requirements from the DSIN, is covered by a generic authorization,

  b) the new fuel cycle is not "standard": an authorization from DSIN is then necessary before the reactor goes critical. Such authorization covers the refuelling operations and power operation in a given mode (load follow operation, extended reduced power operation, etc.).

A reactor is considered as "standard" for a given fuel cycle if the following conditions are met:

- the reactor is loaded with fuel assemblies that the DSIN has already authorized by a generic authorization (FRAGEMA FUEL AFA, AFA XL, AFA-2G, AFA-2GL, AFA-2G-MOX, etc.);

- the fuel management has already been authorized (cycle length, operation mode, etc.);

- the fuel burn-up does not exceed the authorized limit of 47000 MWd/t (maximum average on a fuel assembly).

Consequently, a fuel development, either a modification of the product or of the fuel management, is subject to the authorization mentioned above (Point b).

When new fuel is concerned, the main jusifications requested by the DSIN are the following:

1) in the case of a new fuel supplier without change of the fuel specifications:

- the demonstration of the supplier's ability to deliver the fuel bundles with the desired quality,

- the transmission of the fuel qualification program in test loops,

- the evaluation of the consequences on the fuel behaviour of the possible fuel manufacturing differences (differences concerning the quality of the materials, the fuel bundle components, the manufacturing processes of the different fuel components, ...).

2) in the case of a fuel presenting new specifications :

- all the previously mentioned justifications (in the case of a new fuel presented by a new supplier),

- the design safety criteria and studies of the fuel and its components,

- the document describing the quality control program concerning the fuel,

- proof of the compatibility of the new fuel being introduced with the authorized one already in reactor.

Regarding changes in fuel management, the impact on the fuel assemblie's behaviour has to be examined together with the impact on the whole installation, for such changes generally affect a large number of system equipment in normal and abnormal conditions, as well as their maintenance program and periodic tests, the associated technical specifications, and, in a more extensive manner, the accident related studies.

Then, due to complexity aspects, the DSIN deals with changes in fuel management on a case-by-case basis in close collaboration with its technical support bodies. The DSIN recently authorized the generic implementation in the 1300 MWe pressurized water reactors of the new 'GEMMES' fuel management, caracterized by lengthened fuel campaigns.

2.2.2. Acceptability of the operating organisation's technical proposals

In order to check the adequacy of the provisions that the operating organisation proposes for meeting the safety objectives, the DSIN benefits from the support of the IPSN and the standing groups.

This is particularly the case for a fuel product or management development.

Regarding changes in fuel products for example, the safety analysis conducted by the IPSN aims principally to verify that the fuel has been designed, manufactured, handled and checked with the desired quality and that the proof of safety presented by the operating organisation takes into account all the possible conditions of operation of the fuel in reactor, including abnormal situations, and that in such situations all the design-basis safety criteria and associated safety margins are respected.

Following its analysis, the DSIN sends its instructions to the operating organisation in the form of a letter. Such a letter can concern a single reactor or a set of reactors of the same type.

2.2.3. Implementation of the approved provisions by the operating organisation and monitoring of this implementation by the safety authority

The operator implements the provisions approved by the DSIN.

The DSIN periodically checks the application of these provisions, mainly by performing inspections. Such inspections are run either by DSIN personnel or DRIRE personnel (located near the installations) accompanied by an IPSN specialist choosen in accordance with the subject of the inspection.

Regarding the fuel field, inspections are periodically performed at the different stages of the fuel cycle: during manufacturing processes, during refueling of the reactor core, during recycling at the fuel reprocessing plant. Inspections are also conducted during the design phases of a new fuel management, a new fuel bundle or a new fuel component (grids, nozzles, fuel pellets, etc.), and can take place either in France, if the designer, the manufacturer and the components suppliers are located in France, or abroad if they are located in foreign countries (this is the case for SIEMENS, ENUSA and ABB fuels for example).

3. Most recent fuel developments

The following chapter intends to address the most recent changes affecting French nuclear fuel. Regarding the development of fuel management methods, the new "GEMMES" management method will be depicted as well as the MOX fuel management that is being increasingly applied to the 900 Mwe reactors. Concerning the development of fuel products, the following will focus on the ongoing penetration of foreign fuels and the improvements made to French fuel.

The following chapter does not extensively develop safety and regulatory issues relating to the addressed fuel developments, such aspects are the subject of chapter IV *(Safety and regulatory aspects associated with fuel developments).*

3.1. Changes in fuel management

3.1.1. "GEMMES" fuel management

The French operating organisation EDF has developed a new fuel management method called "GEMMES" to be applied to all the 1300 MW reactors. Its main caracteristics are cycle lengthening, passing from 12 to 18 months on average, 3 batch loads, and the use of a burnable poison made of gadolinium. The modes of operation are grid following including extended intermediate power operation and extended cycle operation.

As far as safety is concerned, this new fuel management method has been developed in particular to improve the preparation and the carrying out of outages and to reduce overall exposure of personnel radiation (but economic considerations also motivated the implementation of the new fuel management system).

The new GEMMES fuel management system made it necessary to develop two new UO2 fuel assemblies caracterized with 4 % enrichment in 235U and the use of gadolinium oxyde (Gd2O3) as burnable poison. These fuel assemblies basically derive from the FRAMATOME AFA-2GL presently used in 1300 MW reactors and caracterized by a 3,1 % enrichment in the 235U isotope.

EDF aims at reaching higher burn up with the implementation of GEMMES fuel management. Indeed, the third extended cycle would lead to surpassing the current burn limit of 47GWd/t (assembly average) and approaching 51GWd/t.

To meet a DSIN demand, this new fuel management method relies on new accident study rules using, in particular, the ones of the 1450 MW reactors. Such rules take into account new safety issues as pellet-cladding interaction (PCI) or the long term consequences of the accidents.

The cycle extension affects the technical specifications applicable for operation, the incident and accident procedures, as well as the preventive maintenance program plan, generally based on an annual periodicity of outages. It then made it compulsory to review such specifications, procedures and plans.

Considering the importance in terms of safety of the modifications to be made to implement such a new fuel management, the DSIN made its application subject to its authorization. The DSIN wanted to take the advice of the standing group in charge of power reactors concerning this new fuel management system before taking the decision to authorize its implementation. The standing group in charge of power reactors held a first meeting to examine how well-founded this fuel nanagement system is and the details of its implementation. It concluded favorably on a first implementation.

The DSIN authorized this first implementation on the 4th reactor of Cattenom in May 1996 and its generic implementation, on certain conditions, on the 1300 MW reactors (20 units) in July 1996. As an example,one condition concerned the carrying out of a complementary test program at Cattenom 4 in order to verify the core conformity in operation, another one related to the augmentation of the boron concentration of the spent fuel pool in certain shutdown conditions.

Apart from Cattenom 4, six other 1300 MW reactors should adopt this new fuel management system in 1996. The DSIN has granted the generic authorization in particular on the condition that the fuel irradiation does not exceed the authorized limit of 47GWd/t. The possibility of going beyond this limit will be examined later by the DSIN and its technical support bodies (see also section III.1, Safety concerns associated with discharge burn up increase).

Figure 3 depicts the current situation of the fuel management system implemented on the French reactors.

3.1.2. Penetration of mox fuel

Recycling plutonium in pressurized water reactors is one of the main objectives of the French operator.

Today, penetration of MOX fuel depends in particular on the manufacturing capacity of the French MELOX plant (able to produce around 120 t/year of MOX fuel).

Each reload is composed of 16 MOX fuel assemblies and 36 UO2 fuel assemblies. The oxyde mix used is a blend of depleted UO2 (0.225% 235U per unit volume) and a mean value of 5,3 % in total Pu.

Plutonium recycling started in France in 1987 at Saint-Laurent B1. It continued with Saint-Laurent B2 in 1988, Gravelines 3 and 4 in 1989, Dampierre 1 in 1990, Dampierre 2 in 1993, Blayais 2 in 1994, and Tricastin 2 in 1996.

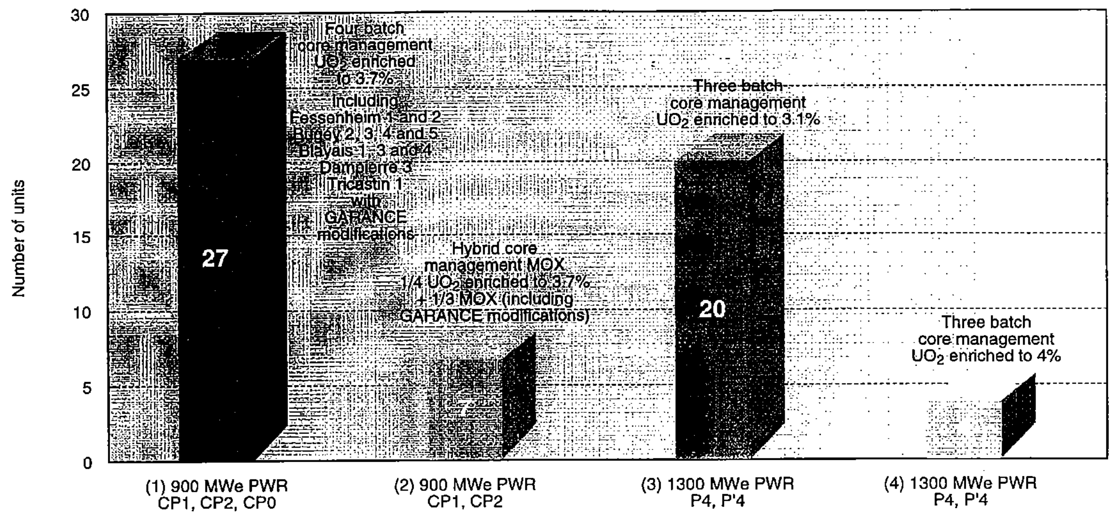
Today, eight 900 MW reactors are using MOX fuel (the plutonium recycling rate is 30 %). Eight other reactors could do so, for their construction license decrees provide for it. The MOX fuel used is energetically equivalent to the 3,25 % UO2 fuel. The current fuel management system allows 400 Kg of Plutonium to be recycled per fuel reload, each reload consisting of 16 fuel assemblies representing 8 t of MOX fuel.

The fuel management system currently used is hybrid consisting of 4 batches of UO2 fuel caracterized by a 3,7 % enrichment rate in 235U, and 3 batches of MOX fuel caracterized by a 5,3 % enrichment rate in Pu. EDF relies on the possibility of increasing MOX fuel discharge burn up so as to make the hybrid fuel management system uniform (4 batches of MOX fuel and 4 batches of UO2 fuel).

EDF wishes to extend the use of MOX fuel to 20 other 900 MW reactors before the years 2005, so that 28 reactors of 900 MW (out of a total of 34) may recycle plutonium.

The next reactors expected to use MOX fuel are the 4 reactors of Chinon B. Modifications for these reactors are planned by EDF in 1997.

Fuel management methods for the EDF population of PWRs
Situation in November 1996



(1)Tricastin 1, 3 and 4; Gravelines 1, 2, 5 and 6; Dampierre 3 and 4; Chinon B1, B2, B3 and B4; Blayais 1, 3 and 4; Fessenheim 1 and 2; Bugey 2, 3, 4 and 5; Cruas 1, 2, 3 and 4
(2)Dampierre 1 and 2; Blayais 2; Gravelines 3 and 4; St Laurent B1 and B2; Tricastin 2
(3)St Alban 1 and 2; Penly 2; Paluel 1, 2, 3 and 4; Nogent 1 and 2; Golfech 1 and 2; Flamanville 1 and 2; Cattenom 1 and 2; Belleville 2
(4)Cattenon 4; Penly 1; Cattenom 3; Belleville 1

Figure 3

3.2. Fuel product development

3.2.1. Penetration of foreign fuels

Since the beginning of the 80s and concurrently to new fuel management development, the French operator wishes to open up its fuel market to new fuel suppliers in order to obtain more competitive fuels in terms of cost, and more efficient fuels in terms of safety.

The first foreign suppliers of EDF were SIEMENS/KWU and ANF in the early 80s (both German, ANF has been bought by SIEMENS in the meantime).

Since the 90s, ABB Atom from Sweden and ENUSA from Spain strive to enter the French fuel market. ABB and ENUSA delivered their first fuel assemblies to EDF in 1994. In 1994 the DSIN authorized the introduction of 4 lead test assemblies supplied by ABB in Flamanville 2 and, in 1995, the introduction of 4 ENUSA lead test assemblies in Belleville 1.

The first complete reloads of ABB and ENUSA fuel assemblies are under construction and should be delivered to EDF in 1997.

These fuel manufacturers are striving to improve their products and are able to propose enhanced fuel assemblies, their efforts concentrated in particular on anti-debris grids.

3.2.2. Evolution of FRAMATOME fuels

The French fuels are also subject to change. Fuel manufacturers are continuously striving to improve their products. For many years the French fuel manufacturer FRAMATOME has been particularly striving to develop fuels with a better ability to resist corrosion (in particular by using low tin zircaloy material for building fuel cladding) and less sensible to debris migration in the primary coolant system (by adding an anti-debris grid at the bottom end of the fuel assembly).

Fuel assemblies presenting such evolution are used in 900 MW reactors, they are the UO2 AFA-2G fuel assemblies. Concerning the 1300 MW reactors, the DSIN authorized in 1994 the use of the new UO2 AFA-2GL, evolution of the previous AFA-XL presenting the same new capabilities as the AFA-2G.

As for the MOX reactors, FRAMATOME developped the AFA-2G-MOX, to be used on the 900 MWe reactors, which includes the same major evolution. The DSIN authorized its utilization in 1995 allowing load following operation. This new fuel will replace the AFA-MOX fuel.

In order to develop enhanced fuel assemblies for the next decade in 1987 FRAMATOME and EDF initiated a large experiment program called X1. Since this time various X1 fuel assemblies have been introduced in reactors. The last 4 X1 fuel assemblies were loaded in reactor 1 of Paluel in September 1996 and presented the following main caracteristics : use of new M5 (Zirconium-Nobium) alloy for 2 fuel assemblie's cladding and 4R-1 (Zirconium-Tin-Vanadium) alloy for the 2 other fuel assemblie's cladding and an increase in cladding thickness.

4. Safety and regulatory aspects associateds with fuel evolutions

The previous chapters briefly described the last changes in nuclear fuel management and design in France. These changes have to be considered from a safety point of view. Furthermore, their compatibility with existing regulations must be ensured. The following attempts to deal with these two subjects.

4.1. Fuel experience feedback in reactors - most recent safety concerns

Recently, the ocurrence of some incidents due to fuel structure deformation brought new safety concerns in Europe and the United States. These incidents relate to control rod drop. They were noted in Sweden, the United States, France and Belgium.

In August 1994 at Ringhals 4 (Sweden), one control rod didn't drop completely and remained a few centimeters above its normal position after a manual scram ; 11 control rods had a higher measured drop time than expected. Following the incident, burn up limits were imposed on the fuel assemblies located below the control rods.

In December 1995 at South Texas 1 (USA), following a reactor trip, the operators noted that 3 control rods did not fully insert. During subsequent testing of all control rods operators noted the same problem for the same 3 control rods as well as for a new one. After checking, all four control rods were located in fuel assemblies that were in third cycle with burn-up greater than 42GWd/t.

In January 1996 at Wolf Creek, following a manual scram, five control rod assemblies failed to insert. The five control rod assemblies were located in fuel assemblies with burn-up greater than 47GWd/t.

The most recent similar incidents occured in France and in Belgium in February, April, June and August 1996.

Following a control rod drop test in Nogent 1 reactor in February 1996, the operators noticed that the drop times of 3 control rods were higher than expected. Further examination revealed that the corresponding assemblies were abnormally deformed, particularly in the dash-pot zone. The assemblies deformed in the form of "W", "banana" and "S" and the maximum amplitudes of deformation noticed were 10 mm for the "W" deformation and 20 mm for the "S" deformation. The assemblies had passed 3 cycles in reactors reaching burn-ups around 30000 MWd/t.

Deformation problems were also noticed on several fuel assemblies at Belleville 2 in April 1996. The assemblies deformed in the form of "banana" and "S" and the maximum amplitudes of deformation noticed were 7 mm for the "banana" deformation and 11 mm for the "S" deformation. The assemblies had passed 3 cycles in reactors reaching burn-ups around 30000 MWd/t.

Confronted with this increasing problem, in June 1996, the DSIN initiated a meeting with the French operating organisation in order to, firstly, take note of the provisions he has taken to deal with the problem. The DSIN is especially determined that the program of investigation implemented by the operating organisation in order to analyse the phenomena be conducted on all types of reactors (900 MWe and 1300 MWe) and be aimed at examining the different fuel types extensively used in reactors (AFA-2G, AFA-2GL, ...) or that the operating organisation plans to use on a larger scale (ABB and ENUSA fuels).

Alike fuel deformations were noticed in DOEL 4 and TIHANGE 3 in Belgium in June and August 1996. Such deformations provoked the incomplete insertion of control rods. The fuel assemblies had passed 3 cycles in reactors reaching burn-ups between 29000 MWd/t and 32000 Mwd/t.

Control rod necanisms failures may also origin drop time problems. This subject is evoked in the IPSN presentation "Control rod cluster drop time anomalies - Guandong and EDF power stations".

4.2. Safety concerns associated with discharge burn up increase

The operating organisation's willingness to take better advantage of the nuclear fuel performance levels drove them to contemplate fuel burn up increase. The new GEMMES fuel management system is indeed based on the use of fuel reaching high discharge burn up (around 51 GWd/t).

This aim is being supported by the collection of result coming from examinations and tests performed upon highly irradiated fuels, by the development of new calculation methods and a more extensive knowledge of the various phenomena affecting the fuel behaviour at high burn up.

In 1995, the French operator, EDF, renewed its generic authorization demand to operate 900 and 1300 MW reactors up to a burn up of 52 GWd/t for UO2 assemblies. The DSIN considers that proof of safety is not complete and that the current authorized limit of 47 GWd/t can't be exceeded. Moreover, the DSIN deems it necessary to fully justify, in a first stage, the 47 GMWd/t limit itself. If the operating experience and the examinations already carried out on higtly irradiated fuels tend to indicate that high burn up can be achieved in normal operating conditions, the results of research programs (started in France and in other countries) indicate that, in some incident situations, highly irradiated fuel might suffer considerable damage. Therefore, the DSIN considers that the operator will have to continue with experiments (in particular the CABRI tests) in order prove that the burn up increase will not endanger safety, and to extend such experiments to MOX fuel. However, the DSIN plans to consult the "standing group" in charge of power reactors on the actual burn up authorization limit relevance and on its possible extension, but also on the special dispensations, granted every year to EDF, allowing this limit to be exceeded on a limited number of fuel assemblies.

In addition, it seems clear that the impact of the burn up has to be taken into account properly in the studies of normal and accident situations, as regards the fuel structure deformation.

4.3. Safety concerns associated with new fuel assemblies

The DSIN has adopted a pragmatic approach concerning the introduction of new fuels in reactors, favorizing efficiency and simplicity as elements of safety : the progressive introduction of new fuel assemblies, and the extensive follow-up of their behaviour in operationnal conditions during three to five cycles.

The DSIN is particularly vigilant regarding the qualtity assurance of the design and manufacture of the new fuel assemblies and usually performs inspections in particular during the design phase and the fabrication phase of precursor fuel assemblies. Such vigilance recently led the DSIN to investigate the impact of vibration phenomena noticed on a few fuel assemblies of some US reactors, on four demonstration assemblies presenting similar features which had to be loaded in French reactors. The fuel designer proposed a small change to the fuel so that such phenomena do not occur and the DSIN authorized the loading of the four demonstration assemblies.

Each cycle is specially monitored and results of examinations are to be sent to the safety authority prior to reloading the new assemblies for the next cycle.

A generic authorization allowing the loading of a complete reload of new fuels cannot be granted before the complete analysis of the experience gained by irradiating precursor fuels during a definite number of fuel cycles (three to five).

The example of ENUSA fuel loading illustrates this methodology. Four precursor assemblies were loaded in Belleville 1 in 1994 and four demonstration assemblies were loaded in the same reactor in 1995.

4.4. Safety concerns and regulation aspects associated with the penetration of mox fuel

4.4.1. Some safety concerns associated with the penetration of MOX fuel

More than 400 MOX fuel assemblies were authorized to be delivered to reactors by the DSIN at the end of 1995. By this time, almost 200 of them had completed 3 irradiation cycles with a lead MOX fuel burn up of 39 GWd/t.

In 1995, the DSIN made the split batch fuel management (or hybrid management) generic, and, in 1996, it authorized the reactors loaded with MOX fuel to be operated under load following conditions.

This authorization was granted after the operating organisation brought proved to the French safety authorities that load following cycles didn't affect the safety of the installations. This proof was principally based on close monitoring of the Saint-Laurent B reactor's operation (the first reactors to operate in load following conditions) and on close monitoring of Dampierre 1 and 2 reactor's operation.

The proof given relied on the physical criteria of the fuel in reactor being met in normal and abnormal situations, the results of irradiation experience both in reactors and test loops (R&D programs), and the MOX fuel assemblies surveillance programs, both on site and in hot cell.

The DSIN accurately monitors the manufacturing quality of the MOX fuel assemblies fabrication by realizing inspections at the MOX manufacturer's plants (especially at the new Melox plant). The DSIN is also particularly vigilant regarding mid and long term changes concerning MOX fuels such as the contemplation of discharge burn up increase, and the Pu enrichment ratio increase for example.

4.4.2. Regulatory aspects associated with the penetration of MOX fuel

In France, the use of Mox fuel is subject to special licensing procedures. Mox fuel can be used only in reactors with a construction license decree which provides for its use. The construction licence decree is the ministerial order authorizing an operating organisation to create a nuclear reactor within a nuclear installation, it prescribes among others the type of fissile material (U or Pu).

In France, 16 reactors (900 MW pressurized water reactors) have a construction licence decree allowing the use of MOX fuel: Blayais 1 and 2; Dampierre 1, 2, 3 and 4; Gravelines 1, 2, 3, 4; Saint-Laurent B1 and B2, Tricastin 1, 2, 3, and 4.

EDF wishes to extend the use of Mox fuel to 20 other 900 MW reactors before the year 2005, including reactors with a construction lecense decree which does not provide for it. 12 reactors fall into this category, but are technically designed to do so, they are the following: Chinon B1, B2, B3 and B4; Cruas 1, 2, 3 and 4, Blayais 3 and 4; Gravelines 5 and 6.

For each of these 12 reactors, DSIN will have to carry out the following administrative procedure (see figure 4) :

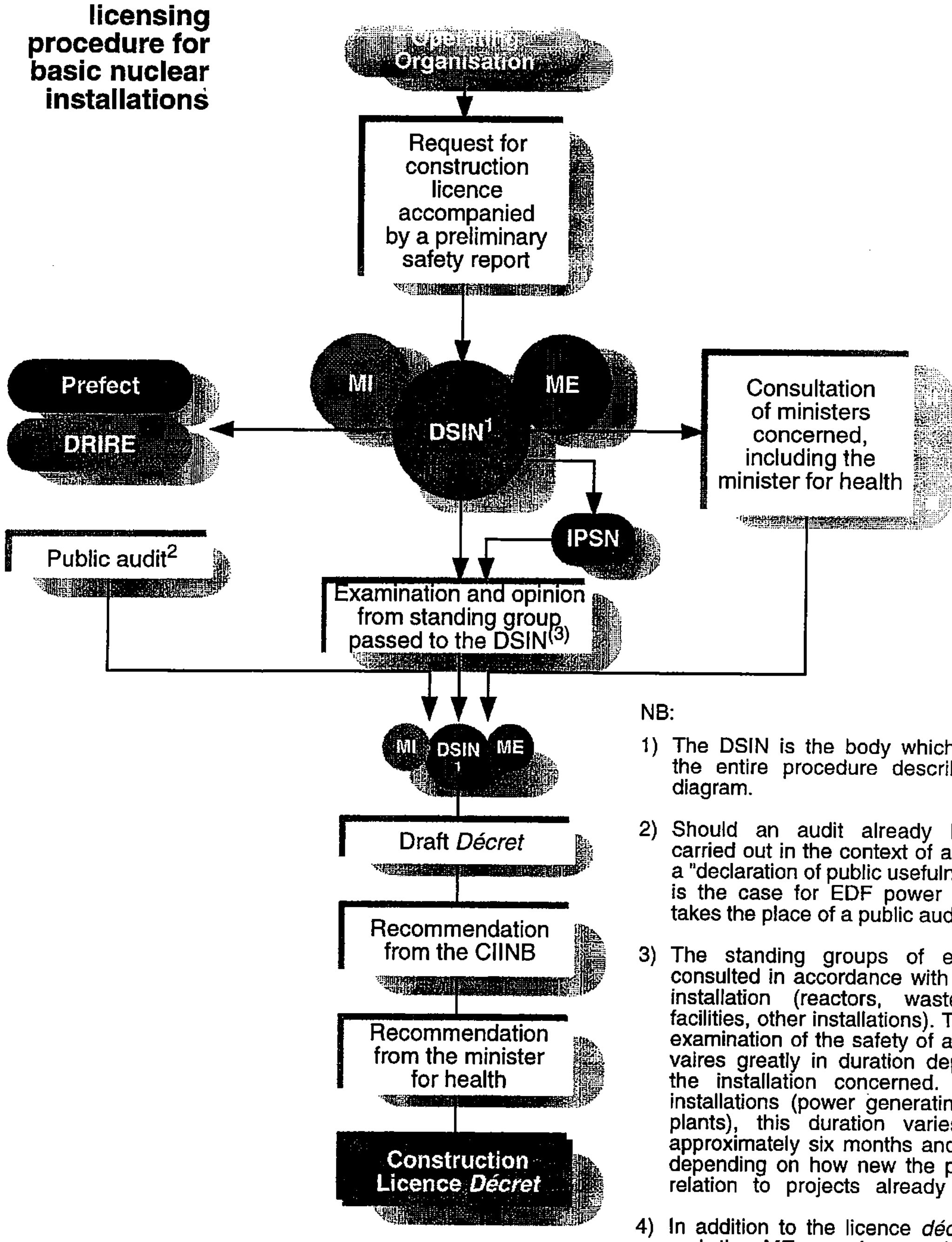
- summarise the initial decree creating the installation,
- carry out a public audit,
- obtain ministerial approval.

Such a procedure was initiated in 1996 to allow the use of MOX fuel in the 4 reactors of Chinon. It could be finished in mid 1997.

Furthermore, the DSIN has to authorize the delivery and the storage of each reload of Mox fuel assemblies, its introduction in reactor as well as the operation of the refueled reactor.

Today, there is no technical obstacle limiting the use of MOX fuel in 900 Mwe reactors, as far as safety is concerned. Nevertheless, precautions have to be taken to ensure proper personnel protection vis-à-vis radiation protection issues mostly during fuel handling.

Construction licensing procedure for basic nuclear installations



NB:

1) The DSIN is the body which processes the entire procedure described in this diagram.

2) Should an audit already have been carried out in the context of a request for a "declaration of public usefulness" (which is the case for EDF power stations), it takes the place of a public audit.

3) The standing groups of experts are consulted in accordance with the type of installation (reactors, waste disposal facilities, other installations). The planned examination of the safety of an installtion vaires greatly in duration depending on the installation concerned. For major installations (power generating reactors, plants), this duration varies between approximately six months and two years depending on how new the project is in relation to projects already examined.

4) In addition to the licence *décrêt*, the MI and the ME can impose instructions.

MI	Minister for industry
ME	Minister for the environment
DSIN	Nuclear Installations Safety Directorate
DRIRE	Regional Directorates for Industry, Research and the Environment
CIINB	Interministerial Commission for Basic Nuclear Installations
IPSN	Institute for Nuclear Safety and Protection

Figure 4

5. Conclusions

The development of nuclear fuel (new fuel suppliers, new materials, new specifications, new codes,etc.), of fuel management, or utilization, and the better understanding of the various phenomena affecting the fuel behaviour, require the safety authorities to be ever vigilant.

Indeed, although the safety authorities are not a priori opposed to such changes, fostered by an industrial logic, they have to be watchful and strive to ensure that, at each stage of developments, the safety problems identified are being analysed appropriately.

Moreover, the safety authorities have to take advantage of foreign experiences and not only look in their own backyard, for what happens today in foreign reactors can tomorrow affect ours. Finally, nowdays such vigilance cannot be limited to safety in operation (including refuelling). The impact on the entire fuel cycle and particularly on the reprocessing part has to be taken into consideration in authorizing a new fuel design or a new fuel management system.

Indeed, it is important that the operator, acting as fuel user and high activity waste producer ensures the coherence, in terms of safety, of its own objectives with the constraints and strategies inherent to its fuel cycle partners, and evaluates the global impact of the choices that it can make.

Dialogue and cooperation between the operating organisation and its fuel cycle partners is then absolutely necessary, in particular for the long term strategies.

Moreover, we cannot deny that the choices made especially when modifying an installation to adopt a new fuel management system might, in some way, affect the population and its environment around the nuclear installations (in particular when MOX fuel is concerned). It is then right that these populations get a large range of information about such choices, and, it is the duty of the operating organisation and of the government bodies to organize and provide such information.

Experience in Spain with Fuel and control Rods

J. R. León (CNA), A. Pérez-Navas (ANA), J. L. Buedo (Iberdrola)

1. Fuel assemblies design evolution

1.1. PWR reactors

Most of the Spanish PWR's (five 930 MW reactors), uses the 12 ft. 17x17 fuel assembly as developed by Westinghouse and commercialised and fabricated by ENUSA. One 1000 MW reactor uses the 16x16 SIEMENS-KWU fuel assembly, and finally other 160 MW reactor uses the 14x14 small fuel assembly developed by Westinghouse.

These general products had been modified in the past in several ways to satisfy the Spanish PWR's demands. The incorporated changes from the original products had been in general to improve neutronic, high burnup resistance and reliability, as explained below.

1.1.1. Neutronic

The mean change had been the introduction of the Zircaloy-4 spacer grid material to improve neutronic economy. The original spacer grid material was Inconel-718, that exhibited high neutronic absorption in comparison with Zircaloy-4. This neutronic economy may be applied to a fuel enrichment saving maintaining the same energy requirements.

At the present some of the PWR's reactors are involved in an evaluation process of axial blankets that consist in the introduction of pellets of reduced enrichment at the top and the bottom portion of the fuel rod. This improvement would reduce the axial neutron leakage and would result in a fuel cycle cost saving.

1.1.2. High Burnup Resistance

In order to accommodate the increase in the fuel discharged burnup in the PWR's reactors, had been necessary to introduce some changes in the fuel assembly geometry. One of these changes is the high burnup resistance geometry, that basically consists in a top and bottom fuel assembly nozzles height reduction to allow higher fuel rod irradiation growth, and a volume reduction in the inner fuel rod spring to accommodate higher rod intemal fission products pressure at the end of life conditions.

Other change had been an improved of the Zircaloy-4 fuel rod material. The new material improves the corrosion behaviour decreasing the alloy tin content. The corrosion behaviour of the fuel rod material is an important concern at high discharge bumups.

Currently, most of the PWR's are involved in demonstration programs for new fuel rod alloys that would be necessary for possible future fuel bumups increase demands. The alloys tested in this demonstration programs typically introduce further tin reduction contents in order to increase corrosion resistance, and incorporate slight niobium contents with the same objective.

1.1.3 Reliability

Other changes in the PWR's fuel assembly design had been introduced in the past, in order to achieve higher reliability levels. One of this changes is the anti-snag spacer grid corner, in order to facilitate the loading / unloading operations of the core during fuel assemblies reload manoeuvres.

The mean fuel rod failure mechanisms in the Spanish PWR's are induced by debris - fretting. To reduce the probability of fuel failure for this cause, an anti - debris bottom nozzle had been incorporated in the fuel assembly. With this improvement most of the metallic particles would be retained at this bottom filter.

Finally, a removable top nozzle had been incorporated at the fuel assembly in other to facilitate reparation of possible failed fuel rods.

1.2 BWR reactors

In Spain there are two BWR Plants (single unit) in operation, S.M. de Garoña a GE BWR/3 with 460 MWe (commercial operation in 1971) and Cofrentes a GE BWR/6 with 990 MWe (commercial operation in 1984) . Both units have been using the fuel designed by GE and fabricated by ENUSA. And also in 1994 one plant loaded 4 Lead Test Assemblies (LTA) designed and fabricated by ABB.

From the beginning, cycle by cycle, we are buying the latest design available in the market and supplied by GE/ENUSA. And because of that, we could say that we are following the evolution of the standards GE fuel assemblies.

This evolution includes the improvements in the number of fuel rods in a bundle, in the cladding material, in the number and disposition of water rods, in the spacers designs, and in the channel design.

We start using the 7x7 (in the 70's) and the 8x8 (GE6 & GE7) Non-Barrier fuel assembly without heat treatment, and with enrichment bellow 3%.

We have been using also the 8x8 (GE10) Barrier fuel with heat treatment.

Now we use the 9x9 (GE11) Barrier fuel assembly with heat treatment, and with enrichment between 3% and 4%. And also we have loaded 4 LTA 10x10 GE12 and 4 LTA 10x10 SVEA96 (ABB).

For the next future we are thinking to use 10x10 fuel assemblies with any type of device like a debris filter or a debris catcher even if we didn't experiment any debris problem but in the rest of the world experience this is one of the actual major fuel problem.

2. Fuel management strategies

2.1. PWR reactors

2.1.1. Cycle Length

Before 1986 PWR's in Spain operated 12 months cycles, with refuelling fractions close to 30 % and feeding enrichments of 3.15 - 3.25 w/o U-235. In 1986 some reactors initiated a transition to 18 months cycles, and the current situation is that the most of the reactors operate with this cycle length in equilibrium conditions, or foresight this operation in a near future. The refuelling fractions for the 18 months cycles are close to 40% with feeding enrichments of 4.15 - 4.25 w/o U-235.

The principal reasons to operate 18 month instead 12 month cycles are to increase nuclear availability reducing other sources with higher variable cost (e.g. coil), and to reduce the number of the refuelling shutdowns with an important saving in maintenance cost. The benefices mentioned above compensate the increase in the fuel cycle cost.

2.1.2. Fuel Discharge Burnup and Core Configuration

In the past (before 1985), the fuel for the PWR's reactors was designed to achieve an average region burnup of 33,000 MWd/tU, like similar reactors in other countries. At that time the core configuration was the so-called OUT-IN-IN loading pattern, with the fresh fuel loaded at the periphery of the core, and with the irradiated fuel loaded in the centre of the core. This type of operation was not optimal from an economic point of view although exhibited high margins to the surveillance core parameters like MTC, F_Q, $F_{\Delta H}$ fuel rod corrosion, etc.

Currently, most of the PWR's in Spain operate with average region burnups near 45,000 MWd/tU, and core configuration of Low Leakage Loading Pattern (L3P). This configuration loads the fesh fuel at the centre of the core, and the most irradiated fuel at the periphery. The currentstrategy (45,000 MWd/tU + L3P) represents an important fuel cycle cost benefit, when compared with the old strategy (33,000 MWd/tU + OUT-IN-IN).

2.1.3. Strechout Operation

Most of the PWR's in Spain stretch the cycle beyond nominal End of Life (EOL) conditions during a period that varies between a few days and 1.5 months in some cases. This type of operation is made reducing thermal power that allows an additional core bumup due to the reactivity gained by the power reduction.

From an economic point of view a fuel cycle cost saving is obtained due to the additional burnup, although a loss of nuclear availability is produced during the strechout period due to the power reduction. This two different effects results in an optimal strechout period (maximum savings) that corresponds to approximately 20 days for the Spanish situation.

Additionally this type of operation allows a higher flexibility (reduced cost impact), to correct any deviation in the scheduled refuelling outages due to deviations in the availability factors.

2.1.4. Surveillance Core Parameters

In order to accommodate the current PWR's management strategies (18 month cycles, L3P, etc.), had been necessary in the past to review the surveillance core parameters like Moderator Temperature Coefficient (MTC), F_Q, $F_{\Delta H}$, Boron concentrations, etc. It had been done by reanalyses of the base accidents contemplated in the FSAR.

For instance the original $F_{\Delta H}$ limit was typically 1.55 for most of the PWR's reactors, while currently after an accident review process is 1.62. This increase had been necessary to accommodate L3P core configuration. Similarly the F_Q had increased from a typical value of 2.3 to a current value of 2.4.

The original MTC limit was zero. It means that operation with a positive MTC was not allowed. Currently the typical MTC limit for PWR's reactors allow operation with a positive value as indicated in the following figure. This new limit had been necessary to accommodate 18 months cycle operation that results in higher boron concentrations at Beginning of Life (BOL) conditions.

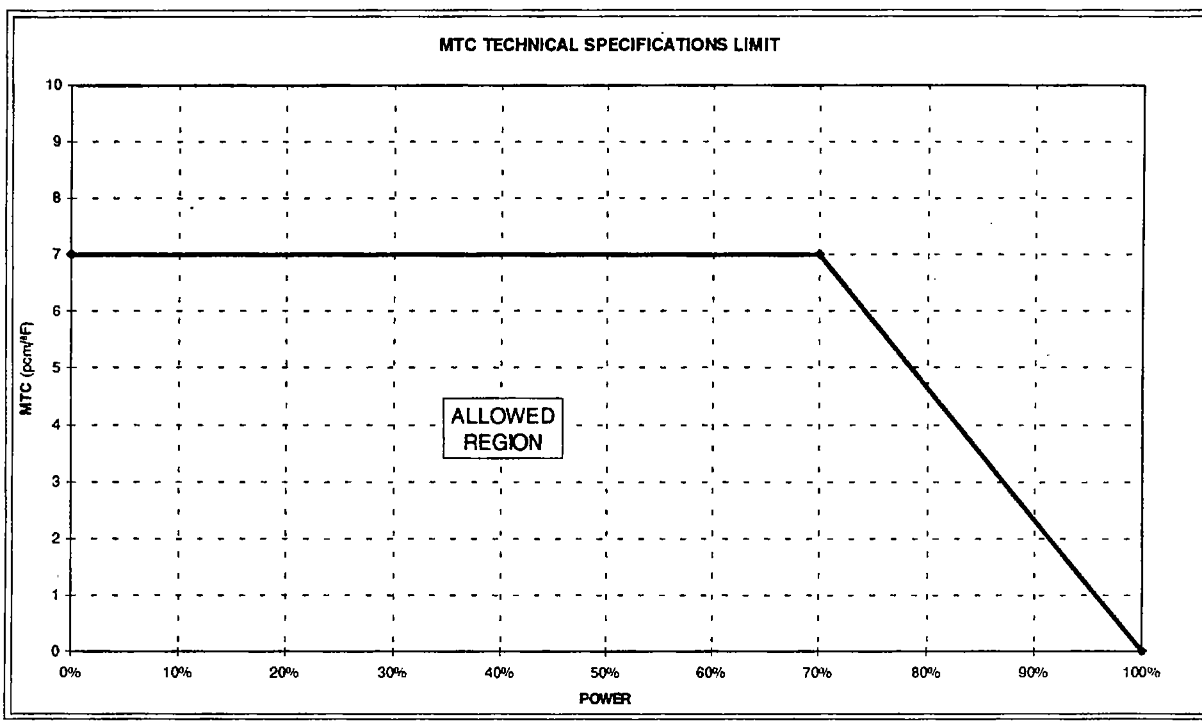


Figura 1.

From the fuel rod corrosion point of view the current discharged burnups exhibited low margins to the design limits as contemplated in the different corrosion models of the fuel vendors. Any future increase in the discharge burnup will probably require improved fuel rod materials. Some of the Spanish PWR's are involved in demonstration process of new alloys for the fuel cladding.

2.2. BWR Reactors

2.2.1. Cycle Data

In Spain the initial cycles for the BWR's were 12 months cycles, with a refuelling fraction of 1/3 and with a conventional reload strategy.

Like other countries we are improving our capacity factors with longer cycles.

The actual cycles length are 18 and 24 months, with the same refuelling fraction of 1/3.

About the reload strategy, the biggest core is getting advantage of the control cell core meanwhile the smallest core is still running with the conventional reload strategy.

2.2.2. Operating Data

The operating data for the initial cycles was that were required the PCIOMR operating restrictions, with the PCI threshold power level, the power ramp rate controls, the conditioning and deconditioning rules.

Also was required the sequence exchange every month a half due to the conventional reload strategy.

For the actual cycles we apply the recommendations for "soft operation", that are very similar the PCIOMR recommendations (Power reductions of 300 MWe/hr, power increases at 150 MWe/hr until get the envelop and later increases at 20 MWe/hr).

The actual cycles keeps the sequence exchange every month and a half for the conventional reload strategy and required the control rod shuffling every two months for the Control Cell Core strategy in order to avoid the new PCI.

In all the cycles in Spain the BWR's use to extend the cycle length from the nominal EOC

conditions, using the FFWTR (Final Feedwater Temperature Reduction) and the Coastdown. Each of those operations strategies give us approximately one extra month of operation.

3. Fuel and RCCA'S behaviour

3.1. PWR reactors

The Spanish Nuclear Power Plants (NPP) operating PWR's reactors have an experience than can be measured in terms of the overall accumulated production: more than 500,000 GWh (electric) since José Cabrera NPP was connected to the grid back in 1969.

During this time we have been aware that the operation have not always been free from flaws in the fuel assemblies (FA). We have found a wide variety of causes that ended damaging FA's although we have not been able to determine the exact cause of a quite important number of defects. In the attached table we summarise the reported failures of our reactors.

But, before going on, we would like to clarify the terminology used in this part of the paper.

Checked occurrence: In some instances we know there is a failed rod in the core but for reasons that are not the object of this paper we do not finally determine which FA it belongs to; but in other cases we are strongly interested in spotting the failed FA and we do it: it is only in this case that we say we have a "checked occurrence".

Unknown cause: Once we have determine that a FA has a failed rod and we actually know which specific rod is it, we could be interested in replacing that specific rod with another one. If this is the case we can closely inspect that rod and, more likely, find the reason of the failure. But if we do not remove it or if we cannot say which mechanism caused the rod to fail we will say the failure cause is "Unknown".

In some cases we saw, live, how the FA was damaged: this is the case of a failing FA against the core shroud ("Fall") when loading a reactor. This has happened twice and in one case the FA was reconstructed in the spent fuel pool of the NPP.

Another interesting case happens when the adjacent grids of two FA's are torn after been hooked during the handling for loading or unloading the reactor ("Grids"). The slight deformation detected in irradiated FA (torsion and bowing) turns the manipulation of FA in a cumbersome task.

Although it only happened once, three FA's were "Smashed": When proceeding to place the reactor vessel upper head in place at the end of a refuelling outage, three stuck Control Rod Drive Mechanisms (CRDM) made the three corresponding FA's to collapse under the weight of the whole set.

Theses three types of "failures", could be rather defined as manipulating incidents and they resulted in no activity leakage to the coolant.

In other instance, when trying to lift a FA from its spent fuel pool rack position we ended upholding only its upper nozzle. Several FA's were found to have seriously affected the link between the upper nozzle sleeves and the rest of the skeleton ("Sleeves"). These cases can not been classified as manipulating incidents since the root cause has been defined as "chemical". No activity leakage was detected but the FA's were not longer used.

On the other hand, our fuel has experienced the type of rod failures that usually take place inside the core during operation; these can only be detected by means of radio-chemistry.

"Baffle J." means baffle jetting. All the cases were reported in the same NPP; four of them occurred at the beginning of the operation and the last one took place a few years ago. All the other NPP baffles were or have been converted to the type that prevents the jets to be formed ("up-flow").

"Debris". The fretting of a metallic debris with the rod, usually at the level of the lower grid, produces an orifice that allows fission gases in the rod to leak to the primary cooling system. As if can be noticed from attached figure about 75 % of the failures that result in activity leakage to the cooling water are known to be due to existing debris in the primary system. After some campaigns to clean the bottom of the vessel, to avoid the introduction of debris during certain activities while in reload outages or to implement slight modifications of FA's to prevent the introduction of existing debris inside the FA's, the impact of this type of failures has decreased.

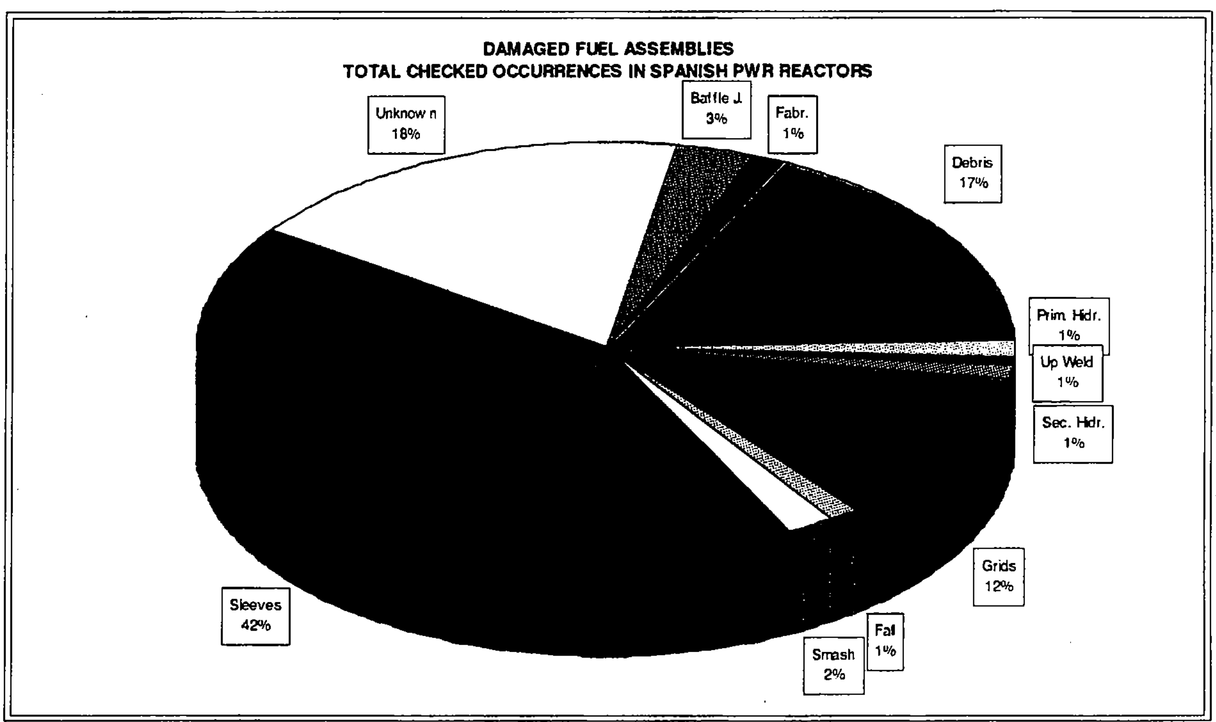
"**Up weld**". We refer to a flaw in the upper plug-tubing welding. We have checked only two cases and we do not have absolute certainty of the root cause since the rods have not been thoroughly inspected in a "hot-cell".

"**Prim. Hidr.**". When the contents of Hydrogen inside the rod is left too high during fabrication process, the Zircaloy is degraded and a "blister" can appear in the rod resulting in leaking fission gases to the cooling water. Only one case has been reported and not very recently.

"**Sec. Hidr.**". One the rod has lost its integrity by any reason, the water inside it can proceed like in the previous case resulting again in further blisters.

The table and figures annexed summarise all the checked failures in Spanish PWR plants.

Occurrences	Quantity
Baffle J.	5
Fabr.	2
Debris	28
Up Weld	2
Prim. Hidr.	1
Sec. Hidr.	2
Grids	19
Fall	2
Smash	3
Sleeves	68
Unknown	29
TOTAL	**161**



On August 1995 a 930 MW PWR reactor had a rod cluster control assembly (RCCA) stuking incident. During a reload shutdown a RCCA was stuk at 24 steps from the FA bottom, but into the dashpot region.

Several tests were made during that reload period. Preliminary conclusion on September 1995 indicated that a single guide tube of the FA exhibited some type of obstruction.

On May 1996 additional test were made with the objective of continue investigation of 1995 incident and compare with other know related incident at other PWR reactor in Europe and USA.

Boroscope inspection was made on the affected FA so as on other FA s with higher rod drop time measurement. No visible particles or condition that can explain previous suspected obstruction was found.

Dimensionally inspection was also made on the affected FA so as on others FA s of the same region and two previous region. Results indicate that most of the FA s from the affected region exhibited "S" shape bow, while FA s of previous region exhibited "C" or "banana" shape with less bow that the affected region.

3.2. BWR reactors

The first BWR connected to the grid in Spain was S. M. Garoña in 1971, and later was connect Cofrentes in 1984. In this presentation we didn't take into account the problems found in the fuel in the 70's because the later design take care of them and are no longer such problem.

During the initial cycles (in the 80's) in the Spanish BWR's we are found four major detected problems in the fuel:

- UNKNOWN
- UNDETECTED MANUFACTURING DEFECTS: Cladding defects, weld defects and end plug porosity.
- PCI (Pellet-Clad Interaction): Cladding fracture by combined effects of pellet expansion imposed cladding stresses and aggressive fission product environment.
- CILC (Crud Induced Localised Corrosion): Our plants are susceptible to CILC dueto the metal composition of the main condenser for using cooper (Admiralty).

With those major fuel problems, we had a failure rate of 1 to 2 leakers per cycle. With one exception cycle in 1986 with 21 leakers only by CILC. Also in the 70's we had more fuel problems due to the design available in the fuel at that time.

Some mitigation actions was taken to avoid those fuel major problems:

- To Avoid Undetected Manufacturing Defects, the vendors improve the manufacturing process and the inspection, sharing among them the best practices. And also we improve the inspection of new fuel at the plant.

- To avoid PCI, PCIOMR (Preconditioning Interim Operating Management Recommendations) operating restrictions applied to Non-Barrier fuel. The Barrier fuel was developed to avoid the PCI without using the PCIOMR's operating restrictions. Barrier Fuel is a fuel with a thin layer of sponge Zirconium inside the Zircaloy-2 cladding inner surface.

- To avoid CILC, the design was modified with fabrication-induced microstructure changes through heat treatment. And also to improved the Water Chemistry at the Plant.

During the actual cycles (in 90's) in the Spanish BWR's we are found another four major detected problems in the fuel:

- UNKNOWN

- UNDETECTED MANUFACTURING DEFECTS: Cladding defects, weld defects and end plug porosity.

- FAST DEGRADATION AFTER INITIAL FAILURE.

- NEW PCI (Pellet-Clad Interaction): Cladding fracture PCI scenario appear in the Barrier Fuel if the fuel have been operating more than 2000 MWd/sT at lower power, has a pellet missing surface and also has a big increase in nodal power produced by a control rod withdrawal.

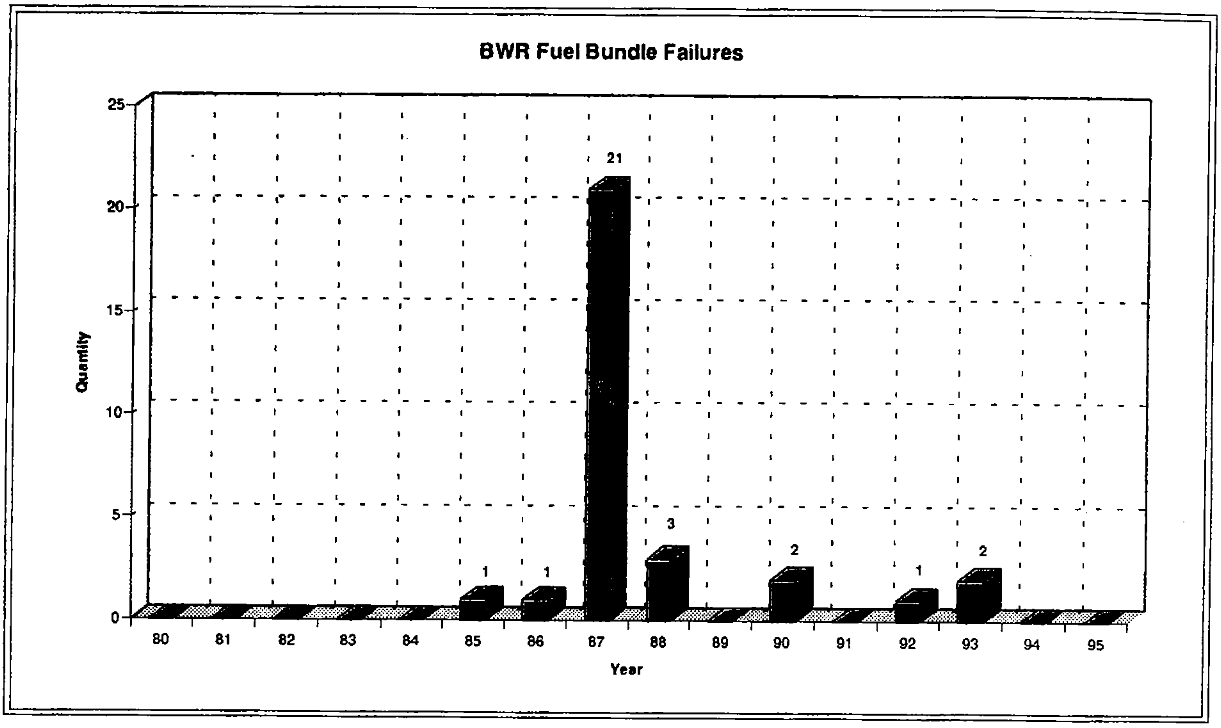
With those major fuel problems, we had a failure rate of 0 to 1 leakers per cycle. In the last cycle and in the currents ones until now, we didn't experiment any fuel failure.

Some mitigation actions was taken to avoid those fuel major problems:

- To Avoid Undetected Manufacturing Defects, the vendors keep improving the manufacturing process and the inspection, sharing among them the best practices. And also we keep improving the inspection of new fuel at the plant.

- To avoid Fast Degradation, the design was change and the fuel has heat treatment only externally and similar internal microstructure to the Non-Barrier fuel. Also the iron content was modified to be similar the Non-barrier fuel.

- To avoid New PCI, we have Control rod shuffling every 2000 MWd/sT, and apply the "Soft Operation" (pseudo PCIOMR Recommendations) to all the fuel.


BWR Fuel Bundle Failures
Quantity
0
5
10
15
20
25
80
81
82
83
84
85
86
87
88
89
90
91
92
93
94
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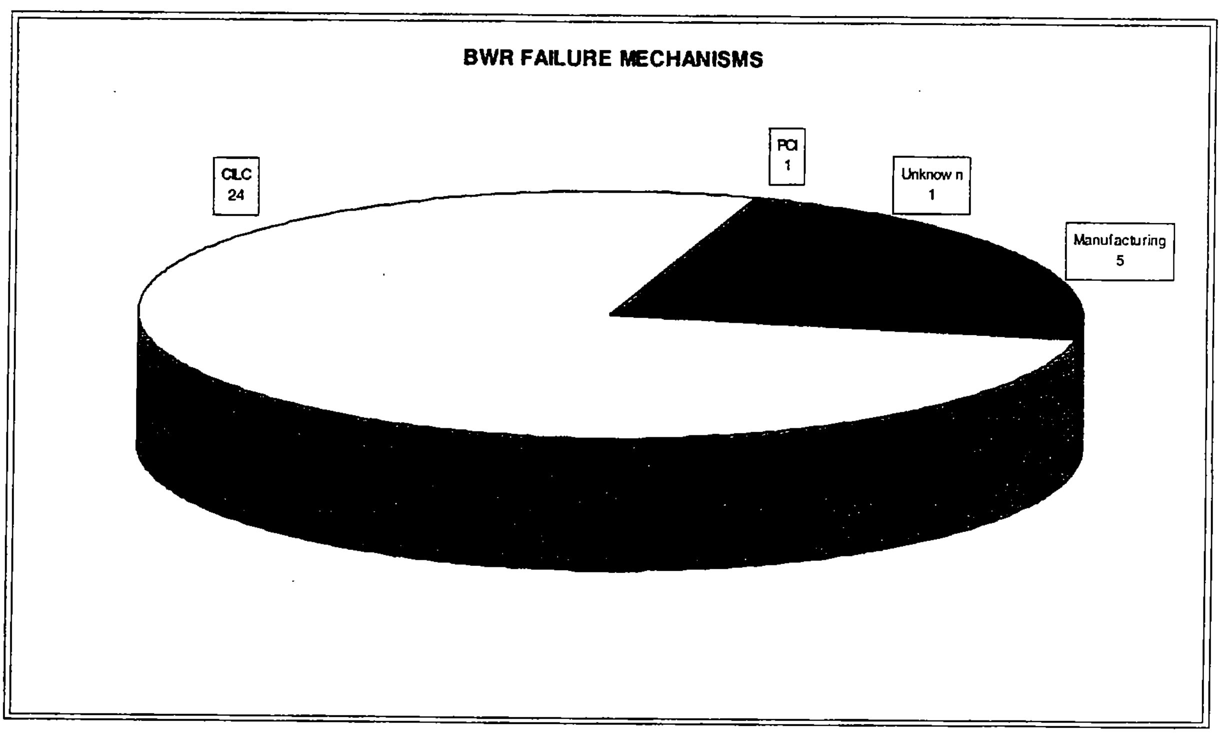
Year
1
1
21
3
2
1
2


BWR FAILURE MECHANISMS
CILC
24
PCI
1
Unknow n
1
Manufacturing
5

TVO'S Experiences with Fuel and Control Rods

R. Lunabba and E. Muttilainen
TVO

1. Introduction

The Finnish regulatory guides require that the nuclear power plants must maintain a supervision programme for the fuel and the control rods. The aim of the supervision is to evaluate whether the performance is in accordance with the design bases and the expectation. In case of failure the failure cause has to be found.

At each unit at least 8 fuel assemblies and fuel channels are annually inspected visually. Many dimensional measurements and oxide thickness measurements have also been performed. During the 80's TVO reused fuel channels. The dimensions of the channels were measured before reuse, which means that TVO has measured the dimension of more than 1000 fuel channels. The rate of fission gas release from the fuel pellets to the free volume of the rods has been determined, based on the Kr-85 activity of the plenum gas, on 22 fuel assemblies. Eddy current inspection has been applied to find leakers.

During shut downs the high exposure control rods of control cells are rather exchanged than inspected and reinserted, the inspection of these control rods is made later. Some medium exposure control rods are selected for inspection during the shut down. Normally only visual inspection is performed. However, some rods exhibiting cracks in the visual inspection have been neutron radiographed.

A summary of the inspection results is given in this paper. The future development foreseen for the design and operation strategy is also presented.

2. Fuel assemblies and control rods used by TVO

2.1. Fuel Assemblies

The initial fuel design used was ASEA-ATOM's 8x8-1. Since the mid 80's the Olkiluoto 1 reactor has been fuelled with Siemens 9x9-1 and the Olkiluoto 2 with ABB Atom's SVEA-64 fuel. Currently both reactors are in a transition stage to 10x10 fuel designs; Siemens ATRIUM 10 for Olkiluoto 1 and ABB Atom SVEA-100 for Olkiluoto 2. In 1997 Lead Use Assemblies of the type GE12 will be loaded into Olkiluoto 2. The GE12 lead assemblies will be the first TVO assemblies with liner cladding.

The fuel inventories of the cores during the different cycles are presented in Figure 1.

2.2. Control Rods

In both the TVO reactors there are 121 control rods. The control rods are made of low carbon stainless steel. Most of the control rods have been delivered by ABB Atom, some by Siemens/KWU. In the ABB Atom control rods the upper part is made of plates welded together to a cruciform structure, where the absorber, i.e. the boron carbide and hafnium in the upper tip, is filled into drilled horizontal channels. In the Siemens/KWU control rods the absorber is filled into vertical tubes, that are kept together by plates.

Originally all the control rods were of the type ABB Atom CR70, which is an all-B4C type. From 1986 through 1992 TVO has been replacing the CR70 control rods with Hf-tip rods from ABB Atom (CR82) and Siemens/KWU. Since 1993 the replacement control rods have been of the type ABB Atom CR82M, where the blade material has been changed from low carbon AISI 304 to low carbon AISI 316 and the wall between the absorber channels and the outer surface is thicker than in the older ABB Atom control rods.

The control rod inventories of the cores in 1994 - 1995 are given in Table 1.

3. Results of fuel and control rod inspections

3.1. Fuel Assemblies and Fuel Channels

3.1.1. Visual inspection

The visual appearance of the fuel assemblies has usually been as expected. Typical inspection results are:

- differential rod growth usually less than 7 mm, in one assembly 8 - 10 mm
- some rod bow in the past, but no rod to rod contact
- oxide on cladding tubes usually thin, some spalling near Inconel spacers due to "-shine"
- some spalling of the channel oxide near steel and Inconel components
- rods covered by red crud
- some broken Inconel X-750 screws in the past
- only limited degradation of most of the leaking rods, considerable degradation of 7 leaking high power rods

Two main causes of fuel rod primary failure can be identified; debris fretting and PCI. The PCI failures of the 9x9 rods can be attributed to pellet chips in the pellet-cladding gap. These rods have not degraded. In the early 90's the vendor improved the geometry and the mechanical properties of the pellets in order to reduce the risk of PCI failures. No fuel rods with pellets of the new design have failed in TVO. Three 8x8 rods have failed by PCI. The operational restrictions have been modified since then and further PCI failures have been avoided. Among the leaking rods five high power rods have broken and two developed axial splits due to secondary degradation. The primary failure cause of the rods with axial splits was PCI. A summary of the fuel rod failures can be seen in Table 2.

The failure of the Inconel X-750 screws was caused by a design error that has been corrected.

3.1.2. Dimensional measurements

The growth of the fuel rods and the fuel assemblies has been as expected. The guidance of the fuel rod end plugs in the tie plates as well as the compatibility of the fuel assemblies with the grapple of the reactor service bridge has been verified.

The results of the channel bow measurements are shown in Figure 2. In a D-lattice like in the TVO reactors the majority of channels bow away from the control rods. The channel bow has not caused any strong interaction with other core internals at TVO.

3.1.3. Zircaloy water side corrosion

The results of our measurements on fuel rod cladding are shown in Figure 3. The growth rate of the oxide is generally relatively slow. The maximum thickness values stay, with some exceptions, below 70 m. Usually the oxide has been measured without disassembling the fuel. The areas covered by the spacers have not been measured. Based on the visual appearance the Inconel spacers seem to accelerate the cladding corrosion locally, but this is missed in Figure 3.

The growth rate of the channel oxide is stronger than that of the fuel rod cladding oxide, as seen when comparing the results in Figure 3 and Figure 4. Some flaking of the channel oxide has occurred, especially in areas in the vicinity of steel or Inconel components.

3.1.4. Fission gas release

According to a Finnish authority requirement the gas pressure inside fuel rods may exceed the system pressure only in a very limited number of rods. In order to verify the fuel rod performance the fission gas release has been determined on rods from 22 different fuel assemblies. The fission gas release rate has been determined non-destructively by taking Kr-85 readings from the gas volume of the rods. Our results are shown in Figure 5.

Considerable fission gas release has occurred in some rods that have experienced relatively high power late in life. The linear heat generation rate of 10x10 rods is low and therefore the fission gas release rate has been low in these rods.

3.2. Control Rods

In 1985 the first cracks near the upmost B4C-channels of CR70 control rods were found. In later investigations the cracking was identified as IASCC (irradiation assisted stress corrosion cracking). The cracking tendency strongly depends on the B-10 burn-up and cracks were found only in areas where the burn-up exceeded 40 %.

The experience of the Hf-tip control rods in shut-down positions has been good but in deep control positions also these rods have experienced cracking. For the Hf-tip control rods the burn-up in the tip is not critical but the average burn-up of the rod. The first cracks in deep control rods have appeared at an average B-10 burn-up of approximately 18 - 22 %.

Neutron radiography investigations have shown that most of the B4C-material stays inside the absorber channels in spite of quite severe cracking of the plate material.

4. Development

4.1. Fuel assemblies

Some aspects emphasised based on the experience of TVO fuel performance can be mentioned:

- The stresses in components made of Inconel X-750 must be maintained sufficiently low.

- Attention must be paid to the corrosion properties of the cladding tubes and channel sheets, especially to the effects of the -shine. One vendor has already modified the heat treatment parameters and the chemistry within the ASTM specification to improve the properties. For the channels from ABB Atom we switched from Zircaloy-4 to Zircaloy-2 a couple of years ago.

- The introduction of debris into the core must be avoided as far as possible. The risk of debris failures can also be reduced by equipping the assemblies with debris filters. Lead assemblies with debris filters have been loaded into Olkiluoto 1. Assemblies with debris filters may be our standard design in the future.

- The operating restrictions for control cells of Olkiluoto 2 have lately been modified to avoid PCI failures. The new recommendation says that conditioning of fuel shall be avoided above a certain burn-up dependent power. The modification affects primarily SVEA-64. The power in SVEA-100 rods will remain below the threshold. In order to fulfil the new criteria for SVEA-64, one or two sequence changes per cycle are performed at Olkiluoto 2, which will reduce the control rod history effects in control cells. In the forthcoming cycle 1997 - 1998 there will be two year old SVEA-100 assemblies available for control cells. SVEA-100 fuel will have a lower heat generation rate and will make it possible to return to mono-sequence operation.

TVO is considering to switch to liner fuel. Liner fuel would allow less restrictive operating rules.

- Considerable fission gas release will occur in some 8x8 rods even at the present power level 2160 MWth. Also in 9x9 some fission gas release will occur. The limit for rod pressure is lower in Finland than in many other countries. TVO decided to switch from SVEA-64 and 9x9-1 fuel to 10x10 designs already before the plans to uprate the power of the reactors. With 10x10 fuel we can license a somewhat higher burn-up.

4.2. Control Rods

TVO is looking for remedies to achieve longer life times of control rods in deep control positions. Both the steel types AISI 304 and AISI 316 are susceptible to stress corrosion cracking in a BWR environment. A change to an other material that we do not have experience of is not possible within a short time. In the Olkiluoto reactors it is not easy to modify the water chemistry. Therefore the most likely remedy for control rods is one that leads to only low stresses in the structural materials in spite of swelling of the absorber material.

4.3. Operating Strategies - Power Uprating

TVO is planning to uprate the reactor power from the current level of 2160 to 2500 MWth. This will increase the average power density from 49.6 to 57.5 MW/m3. The uprating will be performed by flattening the power distribution of the cores. The local maximum ratings will remain unchanged. The preliminary power-flow regime for the uprated power level is given in Figure 6. The regime for the current power level is also given. The power uprating is a part of the plant modernisation project. The criteria for the power uprating are:

- safety must be at least at the present level
- licensability must not be endangered
- susceptibility for disturbances must not increase
- life time of the plant must not be shortened
- electricity must be produced to a competitive price.

The power uprating will be accompanied with many modifications of the Olkiluoto plant. Some of the major modifications are:

- change from 8x8 and 9x9 fuel to 10x10 fuel (better performance)
- steam separators are replaced with new having a better efficiency and a lower pressure loss (stability is improved)
- increased over-pressure protection and pressure relief capacity (pressure transient will be less limiting)
- new power supply for the main circulation pumps (will eliminate the pump-trip as limiting for dry-out)
- modification of turbine (increased efficiency)
- modification of generator
- transformer will be exchanged.

The reactor power will be increased stepwise. A test run at 105 % is performed during the running cycle of Olkiluoto 1. Both units are planned to reach the final level in 1998.

After the uprating the power density of the Olkiluoto reactors will be among the highest in BWRs. The operating principles will remain essentially unchanged:

- operation at base load, no load-follow
- annual cycles, typically 8000 EFPH before stretch-out
- short stretch-out, typically 400 EFPH
- short refuelling outages.

There is little or no experience of operating such cycles in a mono-sequence mode. Therefore we consider to perform sequence changes at both units in the future.

5 Conclusions

The general experience with TVO's fuel and control rods has been fairly good. The fuel failures, some followed by degradation, and the stress corrosion cracks in deep control rods must be addressed in the design and operating recommendations of future fuel and control rods.

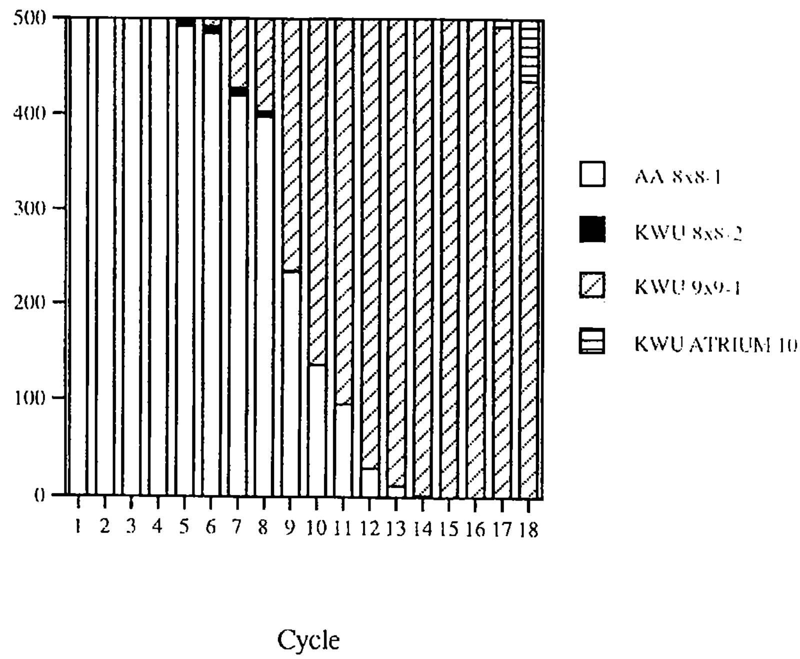
Reactor	Type and number of control rods			
	ABB Atom CR70	ABB Atom CR82	ABB Atom CR82M	Siemens/KWU
Olkiluoto 1	63	35	8	15
Olkiluoto 2	33	82	6	

Table 1. Control rod inventory of the Olkiluoto 1 and Olkiluoto 2 cores during cycle 1994-1995

Reactor	Failure type	Number of failures	Burnup MWd/kgU EOL	Degradation
Olkiluoto 1	Debris	1	15,4	No
	PCI or probable PCI	13	23.8 - 34.8	No
Olkiluoto 2	Unknown	1	24,3	Transversal fracture
	Debris or probable debris	5	10.7 - 36.2	Two rods with transversal fractures
	PCI	3	27.0 - 27,3	Two rods with axial splits
	Not inspected	2	11.0 - 19,8	Transversal fractures

Table 2. TVO fuel failures

OLKILUOTO 1



OLKILUOTO 2

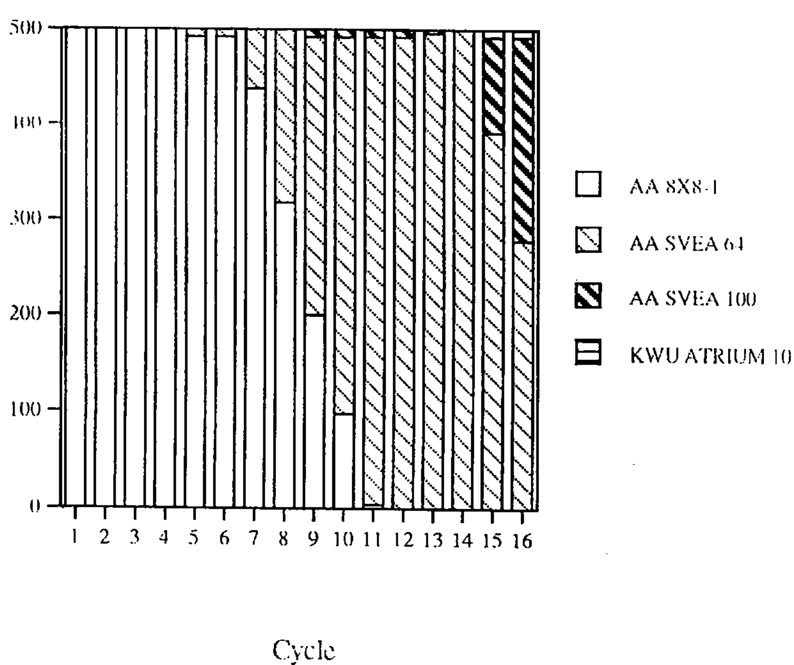


Figure 1. Fuel inventory of the Olkiluoto 1 and Olkiluoto 2 cores du cycles

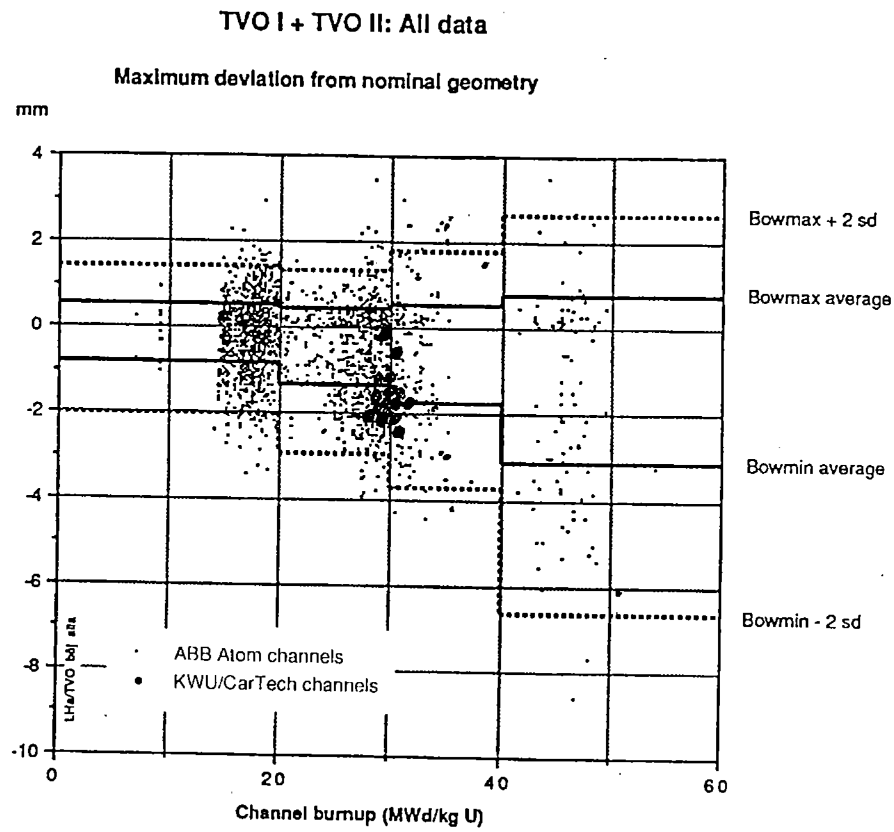


Figure 2. TVO channel bow data

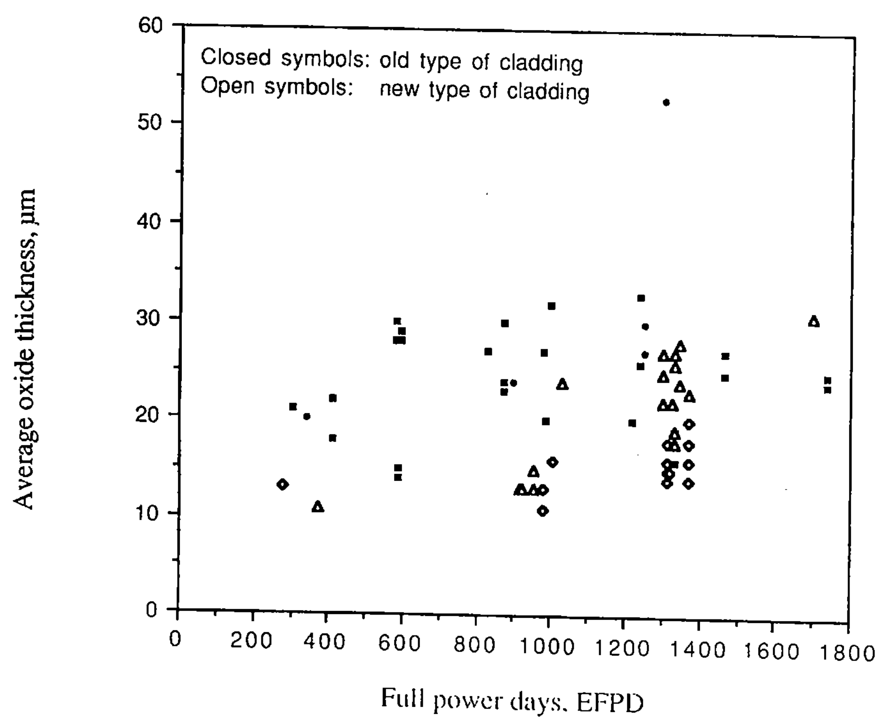


Figure 2. TVO fuel cladding corrosion

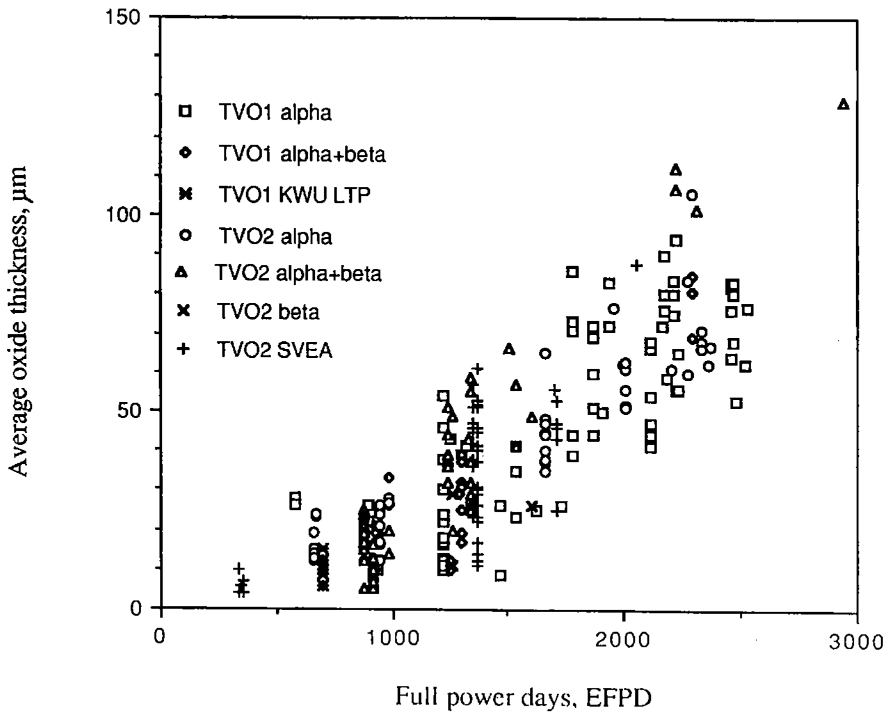


Figure 4. TVO fuel channel corrosion

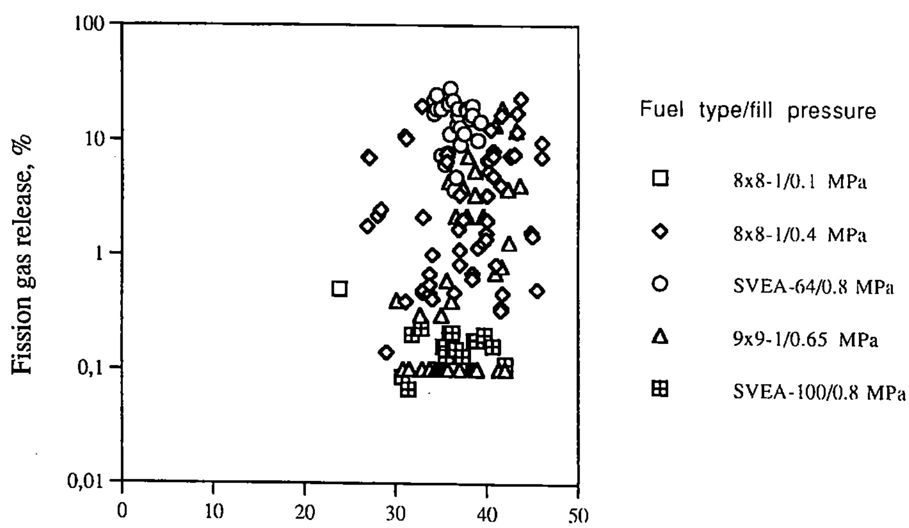


Figure 5. TVO fission gas release data

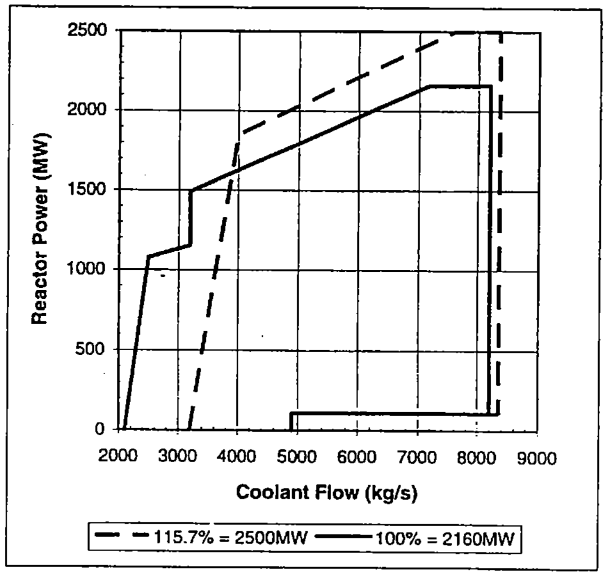


Figure 6. Present and preliminary uprated power-flow regime of Olkiluoto 1 and Olkiluoto 2

Review of safety related aspects in the operational experience of fuel assemblies and shut-down systems in germany

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1. Introduction

High quality of fuel rods is an important safety requirement. Damaged or distorted fuel can potentially inhibit cooling and reactivity control. Furthermore, severe cladding failure represents a basic loss of the defence in depth. Less severe damage may reduce the ability of the fuel to withstand accident conditions /INS 88/. For these reasons, special quality assurance measures are taken during design and manufacturing of fuel assemblies. During operation fuel integrity is verified by monitoring the level of radioactivity in the coolant. In plant design limited leakage of the fuel rods due to cladding defects is taken into account during normal operation.

In the following the operational experience with fuel assemblies and shut-down systems in German light water reactors (LWRs) is reviewed. It shows that there are no severe problems with regard to safety. Mechanisms of fuel rod degradation in pressurised water reactors (PWRs) are presented in more detail for two recent events. Furthermore, events related to the shut-down systems are described and assessed.

2. Historical Overview on Operating Experience with Fuel Assemblies

The development of the total number of fuel assemblies with defects since 1975 in German boiling water reactors (BWRs) and PWRs, respectively, is shown in figures 1 and 2. To give a clearer picture of the distribution of these defects among the whole population of plants, the number of plants of both types with defective fuel assemblies and the total number of plants in operation are also shown in these figures. Test reactors, the first commercial BWR (KRB-A) which has been shut down in 1977 after 10 years of operation and one PWR which is in stand-by operation since 1988 are not included here.

The figures summarise the information given by the plant operators in /ATO XY/, complemented by the reports of events to the regulatory bodies and personal communication with the operators. Since there is some uncertainty about the definition of the term „defect" these figures might differ slightly from numbers given in other sources. Besides the exact number of defective fuel assemblies resulting from the first inspections might not be validated later.

In this paper the term „defective" fuel assembly shall denote all those with cladding failure of one or more fuel rods and cases of degradation which are seen as possible precursors to cladding failure or distortion of the assembly. The development of defect rates is different for both reactor types:

For BWRs relatively high numbers of fuel assembly defects were reported in the late seventies and the eighties. Mostly one or two plants dominated the total number of a calender year having problems with pellet-cladding interaction (PCI). In the nineties the number of defects remained low and the affected fuel assemblies were distributed among most of the BWRs in operation, apparantly not systematic and not related to changes in the design of the assemblies or the operation of the plants. In 1995 all failures were due to mechanical damage during handling of the fuel assemblies and fretting by loose particles. These causes do not indicate any need for general measures to improve the safety of plant operation or the design of the fuel assemblies.

For PWRs the historical trend is different: A larger number of failures occurred in the late seventies, dominated by two plants and mostly due to spacer fretting. From 1980 to 1993 the total number of defective fuel assemblies remained in the range of 5 to 10 per year for all German PWRs, with one exception in 87 which is dominated by the occurrance of PCI at one plant. The defects reported during the period 1991 to 93 are distributed rather randomly among the plants in operation. This situation has changed in the last two years, where the total numbers are in the range of 40 with most defective fuel assemblies in two plants in '94 and two different plants in '95. Most of the defects in these plants seem to be related to changes in the design and/or material of the fuel assemblies. Two root causes are described in more detail in the next chapter.

In most cases of defective fuel assemblies only 1 to 3 fuel rods per assembly were affected, so the total number of defective fuel rods remained very low, i.e. in the range of 10 to 100 per year for all German PWRs together. This represents a fraction of 2 to 20 . 10-5 in relation to the total number of fuel rods in operation in these plants, where 2 . 10-5 per year seems to represent something like a baseline.

3. Recent Cases of Defects at Fuel Assemblies in PWRs

3.1. Dominant Degradation Mechanisms in the Nineties

For the defects at fuel assemblies reported in the last years, the following degradation mechanisms have been found:

- spacer fretting (fretting of adjacent spacer corners with each other, no fuel rod defects occurred),
- fuel rod fretting due to vibrations in certain positions in the core,
- deformed and/or broken spacer springs in fuel assemblies of two manufacturers, and
- accelerated corrosion of a newly developed fuel rod cladding material.

As the analysis of the operational experience shows new failure causes may result from new developments in design and/or material of the fuel assemblies. In the following the potential impact of such developments on operational behaviour of fuel assemblies is discussed for two recent events.

3.2. Accelerated Corrosion of Newly Developed Fuel Rod Claddings

The trend to minimize the fuel costs, e.g. by increasing discharge burn up, and to optimize the loading strategy was the incentive for the development of new fuel rod claddings in the mid-eighties. To optimize the corrosion behaviour, the ranges of allowed contents for several alloying elements of the Zircaloy-4 material were restricted with respect to the ASTM specification, especially the maximum Sn content was reduced.

In the framework of a fuel inspection programme, performed by the manufacturer to check the behaviour of the new product, accelerated cladding corrosion has been found in fuel assemblies made from the new material in German "Konvoi" plants in 1990/1991 /BRE 93/ and in 1993/1994 also in "Pre-Konvoi" plants. The accelerated corrosion was found in the upper half of the rod. It started at the end of the second cycle and increased during the third cycle. It has to be pointed out that no cladding failures were found, however, this accelerated corrosion has to be regarded as a precursor for potential failures during the last cycles.

The results of the root cause analysis are described in /SEI 95/. The final conclusion was that the enhanced corrosion was due to a high pick up rate of corrosion hydrogen in the newly developed fuel rod claddings with Fe content at the lower side of the ASTM specification range. This lead to higher corrosion rates compared to the conventional Zircaloy-4 material at operational periods with high heat flux.

All metallic cross sections taken during post irradion examinations in a hot cell from positions with accelerated corrosion have shown the presence of a fully developed outside surface layer of hydride in such positions. The full hydride layer developed after relatively short times in operation obviously due to a combination of a high hydrogen ingress rate and a high heat flux. This high heat flux is typical for the fuel assemblies in the center of the core in a full low leakage loading scheme and causes a high temperature gradient across the cladding wall. The hydrogen drifts to the outer surface of the wall by thermal diffusion forming the hydride layer there. Solid hydride corrodes faster than the Zircaloy-4 metal thereby further accelerating the corrosion process.

According to the manufacturer accelerated corrosion can be avoided by increasing the Fe and Cr content outside the range of the ASTM specification and by modification of the fabrication routine.

3.3. Cladding Failures of FOCUS Fuel Assemblies in a German PWR

In the late eighties Fuel Assemblies with Optimized Cladding and Upgraded Structure (FOCUS) have been developed to improve the corrosion behaviour of the fuel rod cladding as well as the neutron economy, the thermohydraulic margins, the reliability and the repairability of the fuel assemblies /KRE 93/. Relevant changes were made in the design to achieve these objectives. E.g. new spacers have been developed with mixing vanes at the upper side of the spacers to improve the DNB- (Departure from Nucleate Boiling) behaviour by optimizing the flow conditions.

Furthermore, in FOCUS fuel assemblies all spacer grids are made of Zircaloy. Within the spacer grids, the fuel rods are held in place by spacer springs which are made of the nickel-based alloy Inconel 718. Before, the spacer grids at the uppermost and lowest level had been made of Inconel 718 with integrated Inconel spacer springs.

FOCUS fuel assemblies are used in German PWRs since 1991. In the course of sipping tests failed fuel claddings were detected at 22 FOCUS fuel assemblies during the 1995 outage of one German PWR. Visual inspections revealed fretting marks in the area of the lowest spacer. Fragments of spacer springs were found on the fuel assembly lower end pieces. The cladding defects observed were limited to fuel assemblies in operation for two cycles. There were no defects at FOCUS fuel assemblies which had been in operation for one cycle only /IRS 96/.

An examination of the documentation of all FOCUS fuel assemblies delivered to the PWR concerned showed that the spacers were manufactured according to the specifications. In addition, the flow conditions in the core were analyzed. No indications have been found for any plant-specific abnormalities in comparison to the flow conditions in the tests performed in a test rig by the fuel manufacturer to verify sufficient fuel rod fixing. These tests did not reveal any hint on insufficient support conditions or induced vibration.

Five spring fragments were examined by the manufacturer, using metallographic and fractographic methods. These fragments were almost identical with regard to their dimensions (length approx. 4 mm), fracture location and fracture surface. The damage is characterized by an intergranular fracture surface with little deformation. No indications have been found to preoperational damage of the springs. The fracture structures did not show any sign of forced or vibration-induced rupture. A tension test of an unirradiated spring carried out for comparison resulted in a ductile, entirely transgranular fracture surface.

Intergranular stress corrosion cracking (IGSCC) of the springs of the lowest spacer is considered as the cause of the damage. The failure of spacer springs then led to loss of fuel rod support with subsequent fretting of the fuel claddings. On the basis of the investigations performed so far it is assumed that all conditions necessary for stress corrosion cracking were met: water chemistry, stress and sensitive material.

The high-temperature coolant conditions are given. The relevant stress and material sensivity can be influenced by design and manufacturing of the spacer grids. In fact the tensile stresses of the spacer springs lie locally within the range of the yield strength.

On the basis of transmission electron microscopy (TEM) of stock material, the ortho- rhombic structure of the so-called d-phase (Ni3Nb) was found at the grain boundaries. With reference to the relevant technical literature, e.g. /MIG 93, PRY 86, SHE 93/, the existence of the d-phase is seen as an indication for the sensitivity of the spring material to IGSCC in high-temperature water. It is intended to avoid the sensitization of the spring material to IGSCC by optimizing the heat treatment of the spacer springs in order to change the microstructure.

4. Operating Experience with Shut-Down Systems

Shut-down systems in German PWRs and BWRs have two different driving systems: The control rod drive mechanism in PWRs is a magnetic jack assembly which ensures the scram function and a step by step motion during operation. The drive mechanism in BWRs consists of a hydraulic scram system and a motor-driven operational system. The scram function of both reactor types is designed according to the "fail-safe" principle.

For this presentation the operating experience of the fail-safe system was evaluated taking into account the reported events from the beginning until the end of 1995 including events during commissioning and events in prototype power reactors with comparable design of the scram system. From all reported events those were selected where the scram function for one or more control rods was not available or potentially not available (precursors). These events include defects of

- the mechanical equipment of the drive mechanism,
- the control rods,
- the related electrical equipment excluding failures of the reactor protection system.

The overall experience of the "fail-safe" part of the shut-down systems in German PWRs and BWRs is very good. Significant differences in the deficiencies detected can be revealed between PWRs and BWRs. These differences are based on the different actuation principles.

For German PWRs only ten events have been reported until the end of 1995. The various failure modes do not show any significant main deficiency, see table 1. Only one event was reported that affected more than one control rod at a time. This event was related to flow induced friction that resulted in prolonged rod drop times and occurred more than 20 years ago. After design modifications of the guide thimbles of the control rods similar events did not recur.

During one event one control rod was not inserted at all. This was caused by a breaker failure in this specific control rod drive. Incomplete insertion of control rods was reported three times: Twice the control rods reached the dash pot region and once a foreign particle stopped the insertion of the control rod in an upper position. Each time only one control rod was affected.

German BWRs experienced 26 events related to the scram system, see table 2. Only twice control rods were not inserted at all. Once this occurred at the only plant where each control rod is connected to a single accumulator tank. Here the affected accumulator tank was inadvertently left isolated after maintenance works. The respective control rod did not insert, but the motor-driven system was still available. There was no effect on the other rods. The other event is related to failures of the scram system headers and the related valves. During this event scram valves were blocked by foreign particles in the scram headers. Here three control rods were not inserted, however, the motor-driven system was still available as in the case mentioned above. This event occurred about 20 years ago during commissioning of the plant.

In five cases the contol rods had prolonged insertion times. In all of these events only one or few control rods were affected and the operational motor-driven system was still available.

For both reactor types during none of all the reported events the scram function of the plant was significantly degraded. No events have been reported that resulted in prolonged rod drop time or incomplete insertion due to bent fuel assemblies.

5. Summary and Conclusions

The operational experience of fuel assemblies and shut-down systems in German LWRs was reviewed. With regard to safety related aspects the overall experience is very good.

The operational behaviour of fuel assemblies depends on the operational conditions (e.g. discharge burn up, loading strategy, water chemistry) and the design of the fuel assemblies. New developments seem to be stimulated by the trend to minimize operating costs and to increase the reliability of the fuel assemblies. However, as the analysis of the operational experience shows new degradation mechanisms may result from new developments of the design and material at the initial stage. These led to the increase of fuel assembly defect rates in some German PWRs in the last years.

Furthermore, it has to be pointed out that changes of the coolant chemistry are in discussion also in Germany to optimize the corrosion behaviour of steel components. For example injection of hydrogen in BWRs can be used to avoid intergranular stress corrosion cracking in stainless steel or increase of the LiOH content in the primary circuits of PWRs may result in reducing the build up of activated corrosion products. In such cases the potential impact on fuel cladding performance has to be taken into account /RUD 95/.

The analysis of the operational experience of the „fail-safe" part of the shut-down systems of German LWRs shows different degradation mechanisms in BWRs and PWRs due to the different actuation principles. However, non-insertion of control rods was extremely rare for both reactor types and at none of the events the scram function of the plant was significantly degraded.

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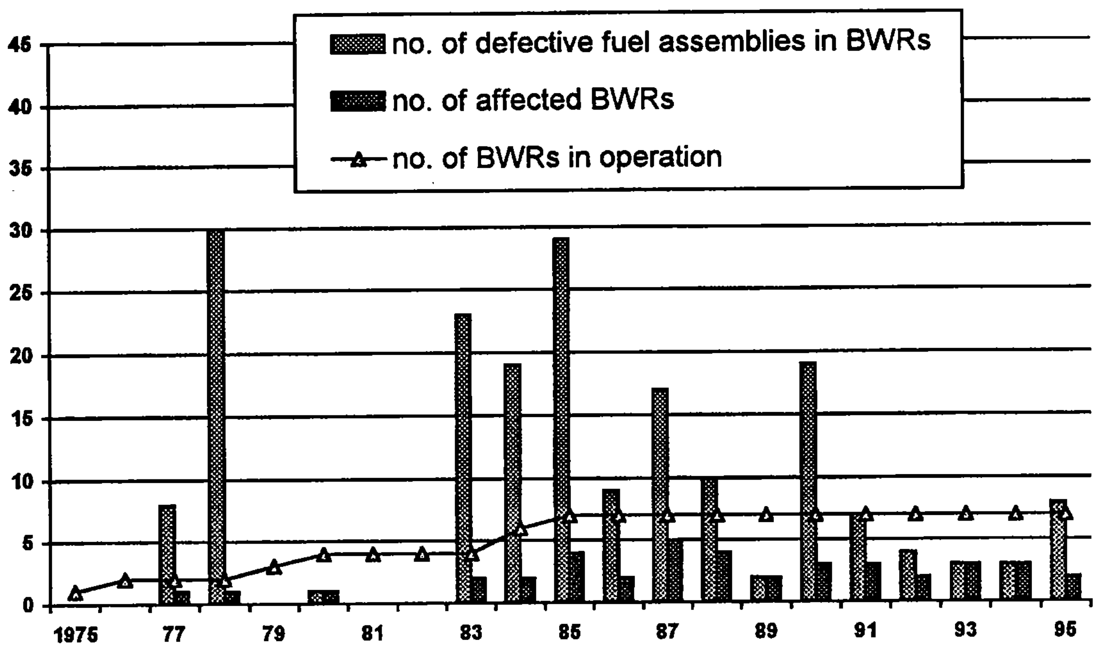

Fig. 1: Total number of defective fuel assemblies per calender year in all German BWRs, number of BWRs with defective fuel assemblies and total number of BWRs in operation in Germany.

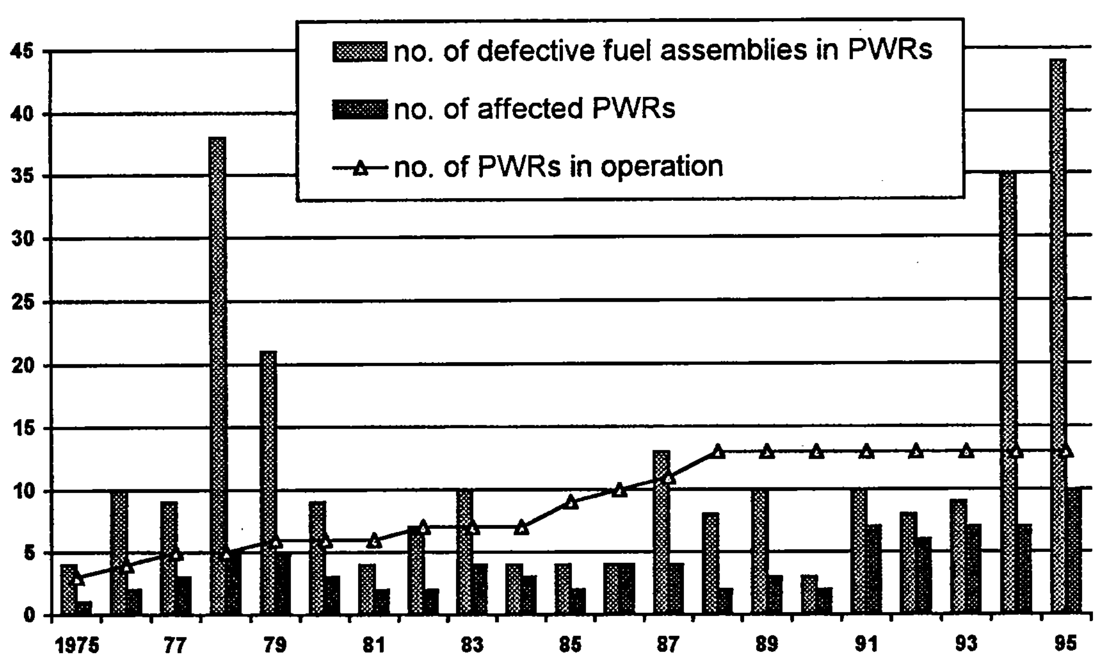

Fig. 2: Total number of defective fuel assemblies per calender year in all German PWRs, number of PWRs with defective fuel assemblies and total number of PWRs in operation in Germany.

Failure	Effect	Number of events	Percentage
rupture of pins	none	4	40
foreign particle in coil	prolonged drop time	1	10
flow induced friction	prolonged drop time	1	10
foreign particle in guide	incomplete insertion	1	10
foreign particle in dash pot	incomplete insertion	1	10
loose screw in dash pot	incomplete insertion	1	10
breaker failure	rod not inserted	1	10
swelling of pin	none	1	10

Table 1: Events relevant for the scram function in German PWRs (until end of 1995)

Failure	Effect	Number of events	Percentage
Scram accumulator tank and related valves	none	10	38
	control rod not inserted	1	4
Scram system headers and related valves	none	2	8
	prolonged insertion time	3	12
	control rod not inserted	1	4
Control rods, guides and drives	none	7	27
	prolonged insertion time	2	8

Table 2: Events relevant for the scram function in German BWRs (until end of 1995)

Performance of KOFA Fuel for PWR's in Korea

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Abstract

Korea fuel assembly(KOFA) is the first locally-supplied fuel for the PWR's in Korea. It was developed and manufactured by KAERI(Korea Atomic Energy Research Institute) and KNFC(Korea Nuclear Fuel Company), in cooperation with Siemens/KWU. Three types of KOFA fuels were developed for eight Westinghouse-designed PWR plants in Korea such that 17 x 17 KOFA for six plants, 16 x 16 and 14 x 14 KOFA's for one plant each. Starting from 1990, in total 1896 fuel assemblies for 35 core reloads were delivered to the plants by 1995. 17 x 17 KOFA was licensed up to the peak rod average burnup of 53 MWD/KGU and irradiated in the plant so far to the peak rod average burnup of 48.71 MWD/KGU and peak fuel assembly burnup of 46.64 MWD/KGU. Except 16 x 16 KOFA which had been stopped producing after fuel failure by rod-to-spacer grid fretting was identified, the performance of 17 x 17 and 14 x 14 KOFA's was quite satisfactory with the fuel rod reliability less than 10-5 per rod. Post-irradiation examination of KOFA showed good mechanical integrity and in-pile performance of KOFA fuel was better than the expected by the design model prediction. Based on the successful implementation of KOFA, fuels for Korea standard PWR has been supplied and currently under irradiation in the plants, and further improvement and development of the fuels for all the PWR's in Korea will be followed.

1. Introduction : KOFA Development

For the supply of fuels to the reload cores of the PWR's in Korea, Korean Fuel Assembly (KOFA) development program has started in 1986 in cooperation with Siemens/KWU[1]. KOFA is the first locally-supplied fuel for the PWR's in Korea while fuels for CANDU-PHWR in Korea were previously developed and supplied by KAERI. Three types of KOFA fuels were developed for eight Westinghouse-designed PWR plants in Korea such that 17 x 17 fuel as shown in Figure 1 for six plants, 16 x 16 and 14 x 14 fuels for one plant each. Fuel design and plant safety analysis was performed by KAERI and fuel manufacturing was done by KNFC, both in cooperation with Siemens/KWU. KOFA was first loaded in Kori-2 in February, 1990 and since that, in total, 1896 fuel assemblies for 35 core reloads were delivered to the plants by 1995, some of which are still under irradiation. To manufacture the KOFA, cladding tubes and straps for the spacer grids were ordered abroad and other parts of the fuel assembly such as UO_2 pellet, top and bottom end pieces were produced in Korea along with assemblying of the fuel rod and assembly. Features of KOFA fuel are summarized in Table 1. Thicker cladding tubes were used for better mechanical

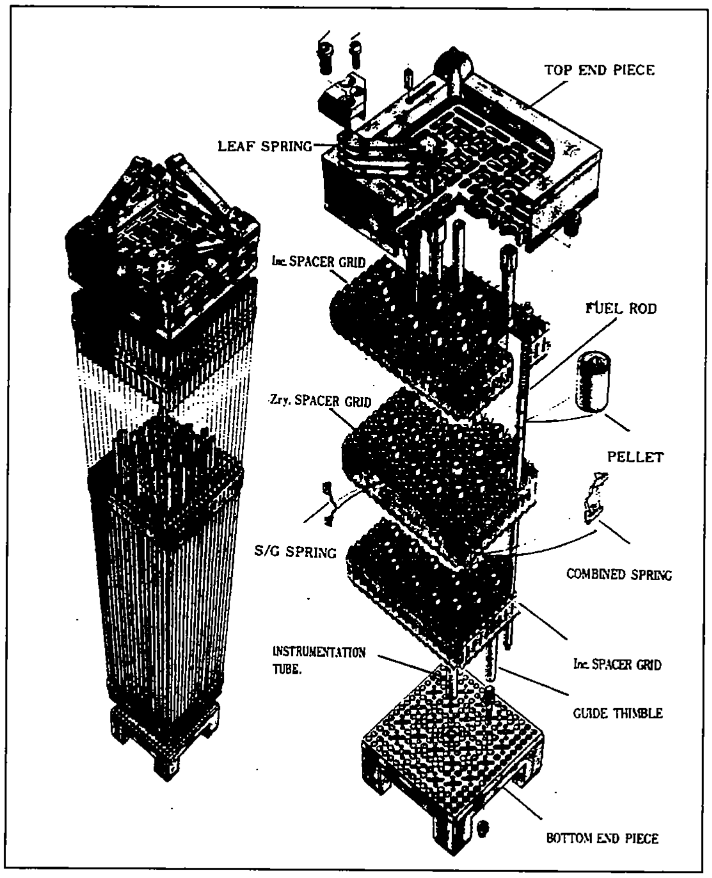

Figure 1. 17 x 17 Korea Fuel Assembly

integrity and the reconstitutable top and bottom end pieces were employed. Zircaloy spacer grids were used for 14 x 14 and 17 x 17 fuel assemblies. As a burnable poison, gadolinia rod was used in which 6 to 9 w/o gadolinia was mixed with 1.8 w/o enriched uranium oxide. Four to eight gadolinia rods were arrayed in a fuel assembly. Licensed design burnups of 14 x 14, 16 x 16 and 17 x 17 KOFA's are 51.2, 48 and 53 MWD/KGU in rod average, respectively.

	14 x 14 fuel	16 x 16 fuel	17 x 17 fuel
plants	Kori-1(587 Mwe)	Kori-2(650 Mwe)	Kori-3&4, Yonggwang-1&2, Ulchin-1&2(950 Mwe)
licensed rod average burnup (MWD/KGU)	51.2	48	53
rod diameter (mm)	10.75	9.5	9.5
cladding thickness(mm)	0.725	0.64	0.64
pellet diameter (mm)	9.11	8.05	8.05
spacer grids	5 Zircaloy-4 2 Inconel	8 Inconel	6 Zircaloy-4 2 Inconel

Table 1. KOFA fuel Characteristics

2. KOFA Performance

2.1. General Irradiation Performance

Three types of KOFA were developed such that 14 x 14 fuel for Kori-1, 16 x 16 fuel for Kori-2 and 17 x 17 fuel for Kori-3 & 4, Ulchin 1 & 2 and Yonggwang 1 & 2 plants. KOFA fuel supply and irradiation in the plants from 1990 to 1995 is summarized in Table 2. In total, 1896 fuel assemblies or 476,980 rods were delivered for 35 reloads. U-235 enrichment varied between 3.5 and 4.0 w/o, and core cycle lengths ranges from 12 to 18 month depending on the plant. Maximum irradiated burnup so far is 46.64 MWD/KGU in fuel assembly average and 48.71 MWD/KGU in fuel rod average. 16 x 16 KOFA was stopped producing after three reloads since fuel failure by rod-to-spacer grid fretting was found in the second irradiation cycle. Therefore, 14 x 14 and 17 x 17 KOFA's were supplied until 1995. Among 440,320 fuel rods of 14 x 14 and 17 x 17 KOFA irradiated, so far only one fuel rod was found leaking by loss of the lower end cap. Therefore, performance of 14 x 14 and 17 x 17 KOFA's was quite satisfactory with the fuel rod reliability less than 10-5 per rod. In particular, even though KOFA does not have the advanced debris filtering feature, fuel failure by debris has not occurred yet showing that all the plants were operated under debris-free condition.

	no. of reloads	U-235 enrichment (w/o)	cycle length (month)	no. of fuel assemblies/rods	maximum irradiated burnup* of fuel assembly/rod (MWD/KGU)
14 x 14 fuel	5	3.5 - 3.8	15	224/40,096	45.11/47.05
16 x 16 fuel	3	3.5	15	156/36,660	34.97/36.64
17 x 17 fuel	27	3.5 - 4.0	12 - 18	1,516/400,224	46.64/48.71
total	35	-		1,896/476,980	

(*) as of Nov. 1996.

Table 2. Summary of KOFA Delivery and Irradiation

2.2. In-Pile Performance

2.2.1. Surveillance of KOFA Fuel Assemblies

For fuel surveillance, visual examination of the irradiated KOFA fuel assemblies were performed after reactor shutdown showing that the irradiated KOFA fuels were in good condition without any anomaly. During and after fuel manufacturing, some of KOFA fuels were selected and pre-characterized such that as-built properties of the pellet and the cladding were characterized and physical dimensions of fuel assembly and its components were measured. Six fuel assemblies were selected and irradiated respectively for 14 x 14 fuel in Kori-1, 16 x 16 fuel in Kori-2, 17 x 17 fuel in Kori-4 and 17 x 17 fuel in Ulchin-1. These pre-characterized fuel assemblies are currently in the spent fuel storage pool at site ready for further PIE, if necessary after completion of the scheduled irradiation.

2.2.2. Cladding

Cladding of KOFA fuel was a standard Zircaloy-4 of high cold-worked and partially recrystallized. It was ordered from Westinghouse/SMP during first three years and after that from

Siemens/NRG. Characterization test results of Westinghouse-supplied cladding tubes are shown in Table 3[2]. Its in-pile irradiation growth and corrosion performance are as follows.

		specification or target value	measured value
chemical composition	Sn(%)	1.20 - 1.70	1.42 - 1.60
	Fe(%)	0.18 - 0.24	0.19 - 0.22
	Cr(%)	0.07 - 0.13	0.10 - 0.12
	O(%)	0.10 - 0.16	0.11 - 0.12
	C(ppm)	270 max.	135 - 147
	N(ppm)	80 max.	27 - 34
	Si(ppm)	120 max.	81 - 109
texture (Kearns number)	Fr	0.49	0.521
	Ft	0.41	0.413
	Fz	0.10	0.060
recrystallization(%)		10 - 40	13 - 24
average second phase particle size(μm)		> 0.15	> 0.2
accumulated annealing parameter(hr)		7.1×10^{-18} - 40×10^{-18}	17.6×10^{-18}

Table 3. Characteristics of KOFA Cladding

Irradiation Growth

Irradiation growth of the cladding was measured for the surveillance fuel assemblies in Kori-2 after first and second cycle irradiation. Measurement was done by VFES(Versatile Fuel Examination System) in site pool after first cycle irradiation[3] and by direct measurement in the hot cell at KAERI after second cycle irradiation[4], both with measurement error less than 1 mm. Fast neutron fluences of the measured rods were obtained by core follow calculation with the uncertainty less than 4 %. Figure 2 shows the measured irradiation growth versus fast neutron fluence. When compared with the design model prediction, measured irradiation growth of the KOFA cladding was about 20 % lower than the design model prediction. And irradiation growth with fast neutron fluence (E > 0.821 Mev) showed the trend of two step behavior of the saturation and the growth which is known as a typical behavior of the annealed Zircaloy-4[5]. Therefore, based upon the measured data, a new two step irradiation growth model was derived for the KOFA cladding as follows[6].

$\Delta L/L(\%) = 0.145\ [1 - EXP(-0.6\phi / 10^{21})]$, for fast neutron fluence $(\phi t) < 3.2 \times 10^{21}$ n/cm^2

$\Delta L/L(\%) = 7.74 \times 10^{-2}\ (\phi t / 10^{21}) - 0{,}128$, for fast neutron fluence $(\phi t) \geq 3.2 \times 10^{21}$ n/cm^2

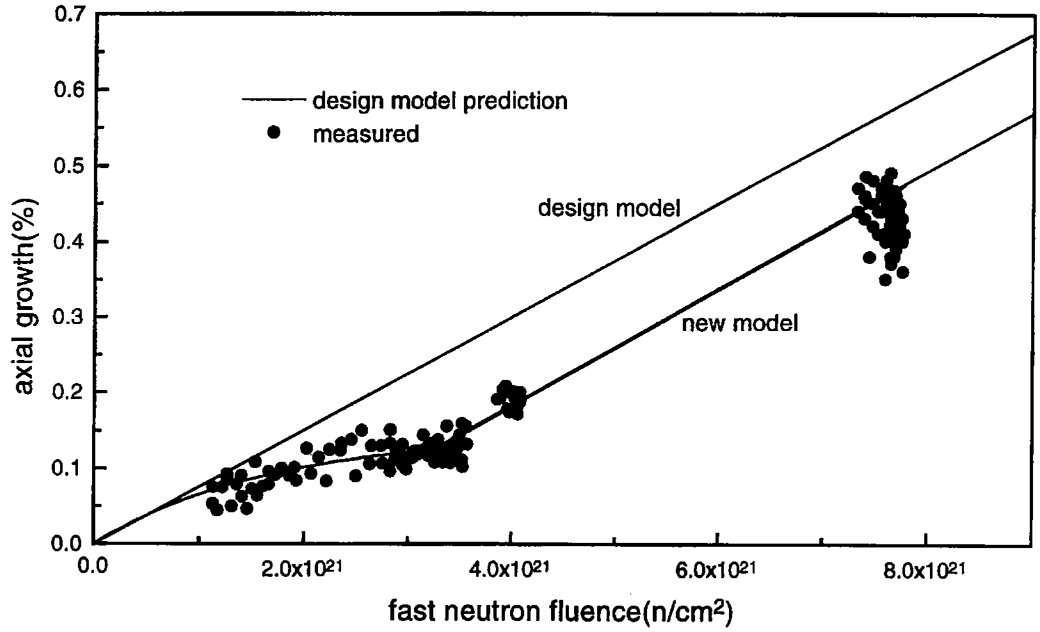

Figure 2. Measured and predicted irradiation growth of KOFA dladding as a function of fast neutron fluence

Corrosion

Oxide layer thickness of the cladding was measured for the fuel rods irradiated for two cycles in Kori-2 plant. Ten fuel rods in three fuel assemblies was measured in the site pool and the hot cell. Burnup of the measured rods ranges from 25 MWD/KGU to 35 MWD/KGU. Figure 3 compares the measured oxide layer thickness which is the axial peak thickness occurring near 80 % from the fuel rod bottom with the design model prediction. It shows that corrosion of KOFA fuel cladding was about 30 % lower than the expected for the standard Zircaloy-4 cladding. During the operation of Kori-2 with KOFA fuel loaded, there was no anomaly significantly different from other PWR's. Coolant temperature at the core outlet of Kori-2 is 325 C. Primary water chemistry of Kori-2 followed the constant pH (6.9 at 300 C) scheme during first cycle and changed to the coordinated boronlithium scheme with maximum lithium level below 2.2 ppm in second cycle. Therefore, it is speculated that lower corrosion of the KOFA fuel cladding may resulted from improvement in the cladding manufacturing processes such as heat treatment and cold working and better control of chemical impurity elements such as silicon and nitrogen as shown in Table 3[7].

2.2.3. UO_2 Pellet

UO2 pellets of KOFA fuel were manufactured by KNFC with the imported enriched UF6. Reconversion of UF6 into UOz was done by AUC process. Post-irradiation examination(PIE) at the hot cell was done for the fuel rods of burnup 35 MWD/KGU irradiated for two cycles in Kori-2. Properties of the as-manufactured UO2 pellet lot of the rod are summarized in Table 4. Density of UO2 pellet was 95.1 % TD and average grain size was 7.9, μm with sintering temperature of 1,700 ° C. PIE results can be analyzed as follows[7]. Released fission gas in the fuel gap was collected and measured for the one rod showing that fractional release of the fission gas was 0.34 % while bestestimated value of the rod by design model is 1.49 %. Densities of the UO2 pellets were measured by toluen immer-

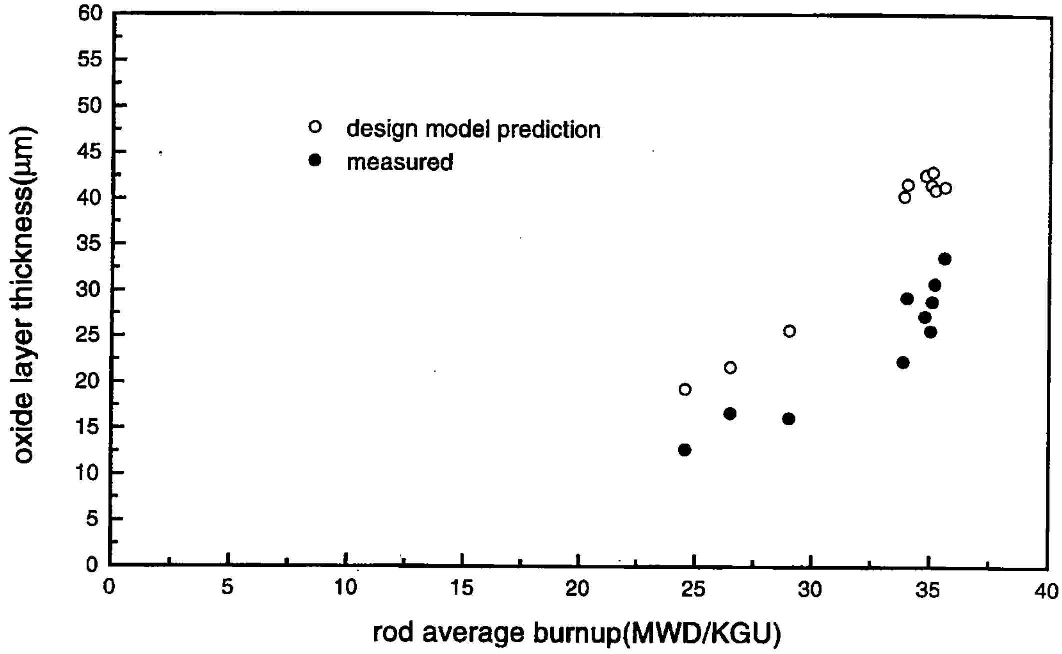

Figure 3. Measured oxide layer thickness of KOFA clanding compared with the design model predicition

sion method showing that those were within the design model predictions as shown in Figure 4. Optical microscopic examination of the irradiated pellet proved the stability of the pellet microstructure with no grain growth and axial gamma scanning of the fuel rods showed that pellets were stacked stable and there was no significant axial gap formed.

	specification	measured value
density(g/cc)	10.4 ± 0.15	10.42 ± 0.03*
open porosity(%)	-	0.19
diameter(mm)	8.05 ± 0.01	8.05 ± 0.0026*
surface roughness(μm)	< 2	0.95 ± 0.097*
average grain size(μm)	4 - 25	7.9
resintering density(g/cc)	< 0.15	0.057 ± 0.015

(*) standard deviation

Table 4. Characreristics of UO_2 Pellet Lot

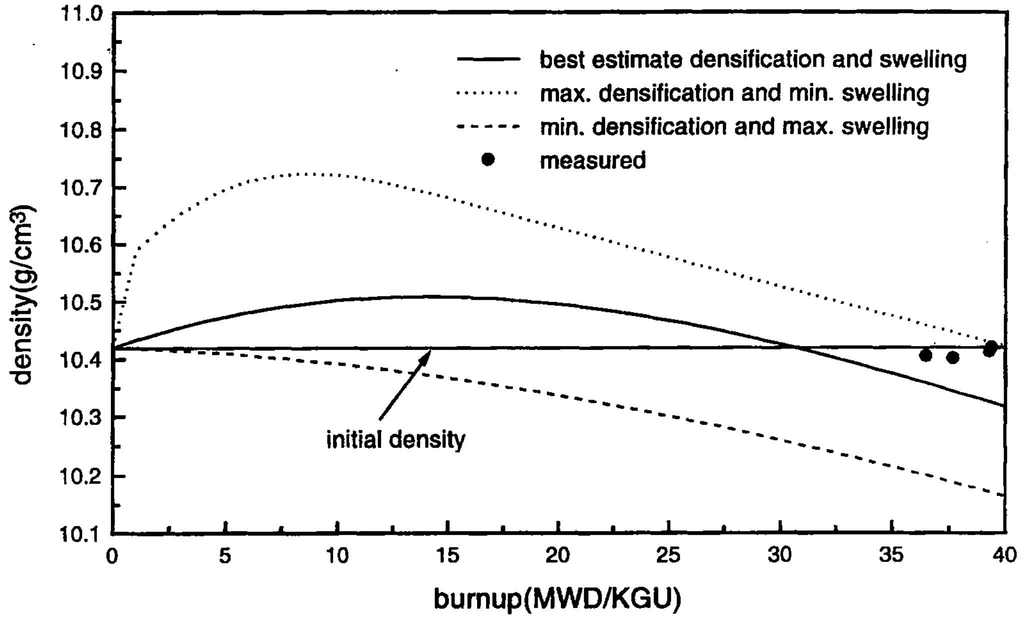

Figure 4. Measured density of UO_2 pellets compared with the design model prediction

3. Conclusion

KOFA is the first locally-supplied fuel for the PWR's in Korea. It was developed and manufactured by KAERI and KNFC, in cooperation with Siemens/KWU. From 1990 to 1995, in total, 1896 fuel assemblies of KOFA were delivered for 35 core reloads. Post irradiation examination showed that in-pile performance of KOFA was better than the expected by design model prediction. Based upon the successful implementation of KOFA in Korea, fuel for the Korea Standard PWR originated from ABB/CE System 80 fuel was successfully supplied in 1995, and further improvement and development of the fuels for both the Westinghouse-designed PWR's and the Korea Standard PWR's in Korea will be followed.

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TECHNICAL SESSION II

Exceptional crud build-up in Loviisa-2 fuel bundles

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Summary

Anomalous primary coolant outlet temperatures at Loviisa 2 unit were first discovered in October, 1994, one month after the start of the 15th cycle. The reason for increased outlet temperatures was soon found out to be decreased coolant flow through part of the fuel assemblies. This phenomenon was most pronounced in six first cycle fuel assemblies with spacer grids made of Zr1%Nb (later: ZR assemblies).

Due to continuously increasing outlet temperature the reactor was shut down at the end of January, 1995. The six ZR assemblies were discharged from the reactor. Towards the end of cycle no. 15 the rate of outlet temperature increase slowed down and essentially stopped in the remaining assemblies, which had spacer grids made of stainless steel (later: SS assemblies). One of the ZR assemblies was visually inspected using the pool-side inspection equipment at Loviisa 2 unit. This inspection showed that the reason for the decreased coolant flow was deposition of crud in the spacer grids, especially in the lower parts of the assembly.

Based on data of coolant outlet temperatures, flow resistance measurements were carried out for eighty SS assemblies during the refuelling outage between cycles no. 15 and no. 16. As a result thirty assemblies, which had the most clogged spacer grids, were discharged from the reactor before their planned end of life.

The cycle no. 16 started with an indication of a small leakage in September, 1995. Primary coolant activity kept increasing steadily, indicating more fuel failures, up to values never reached before at Loviisa NPP. The estimated number of leaking rods varied from approximately 10 rods up to ca. 70 rods. Finally, Loviisa 2 unit was decided to be shut down in late October, 1995. Sipping of the core indicated that there were seven leaking fuel assemblies in the reactor. All leaking assemblies had earlier been identified as being slightly clogged due to the deposition of crud in the spacer grids.

Altogether thirty-two slightly clogged assemblies, including the seven leakers, were discharged from the reactor. The rest of cycle no. 16 was operated without further indications of fuel leakage or any indications of further clogging of the assemblies.

Up to now, pool side inspections of two leaking assemblies have been carried out. It has been concluded that the fuel failures were caused by mechanical wear of the cladding in contact with the spacer grid. It is assumed that the increased flow resistance of the lowest spacer grids has caused turbulence in the coolant flow and thereby vibrations which have led to wear of the cladding tube.

Equipment

1. In-core instrumentation

Standard in-core instrumentation of a VVER-440 reactor consists of 210 fuel assembly outlet temperature gauges distributed among the total of 349 fuel assembly locations. Since the 36 outermost fuel assemblies at Loviisa NPP reactors have been replaced with dummy steel assemblies to protect the pressure vessel from excessive fast neutron flux, the outlet temperatures are measured in 190 out of 313 assembly locations. In addition, there are 36 locations in the core where instrument lance is pushed into the fuel assembly instrument tube. Each instrument lance consists of one inlet temperature gauge, one integral and four axially located local neutron detectors. Figure 1 presents the locations of the instruments in the core.

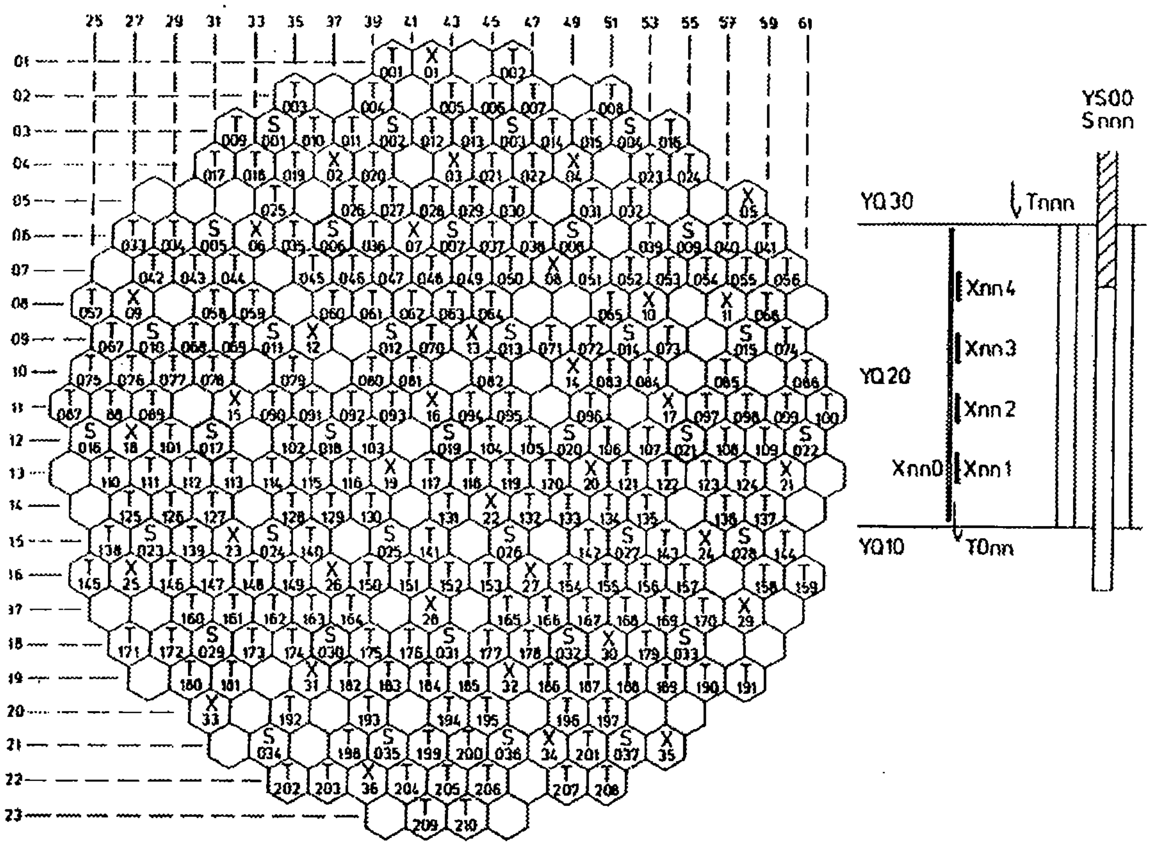


Figure 1. In-core instrumentation system of the Loviisa reactors
T - Outlet temperature
X - Inlet temperature, local and integral neutron flux
S - Control rod position

Extensive instrumentation together with the fact that the fuel assemblies are surrounded by channels preventing cross flow ensures good means of following up what is happening in the fuel assemblies during operation. The symmetry of the loading patterns is usually 30 degrees, which gives another means to compare the behaviour of the fuel assemblies in symmetrical core locations.

2. Flow resistance measurements of fuel assemblies

To assess the degree of clogging of any individual fuel assembly, a pool-side flow measurement loop was designed and installed to Loviisa 2 unit. The loop consists of a test section, a pump, valves and instrumentation to measure flow rate, temperature and pressure of the system, see figure 2. The coolant flow was adjusted to a pre-set value of 10 kg/s (compared to 25 kg/s in operating conditions) for all measurements. The flow resistance of an assembly was calculated using the results of the flow, pressure and temperature measurements. Fresh fuel assemblies were used as a reference to establish a flow resistance value for a clean assembly. The results of the measurements showed that the equipment performed reliably. The standard deviation for a repeated set of measurements was generally 1 % or lower.

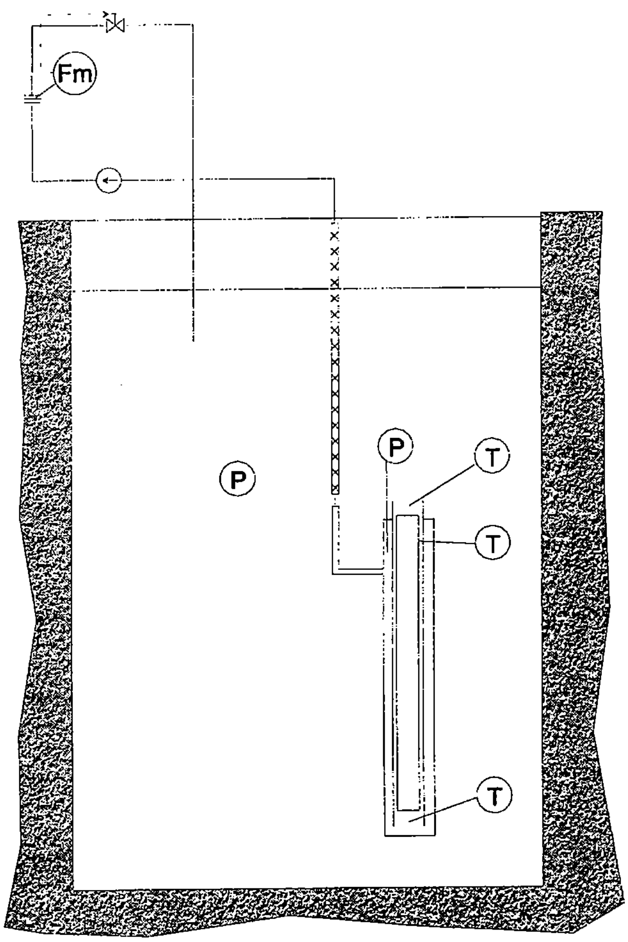


Figure 2. Flow resistance measurement loop

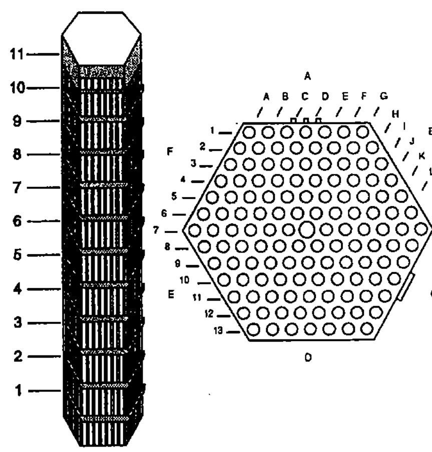


Figure 3. Denotation of the spacer grids, fuel rods and the six sides of the assembly

3. Identification of leaking rods with ultrasonic inspection

The leaking rods of two SS assemblies were identified using ultrasonic inspection device by ABB Reactor GmbH. The signal for a sound rod is a set of two distinct peaks. If a rod is defected the sizes of the signals decrease, which is an indication of a large amount of moisture inside the rod and/or hydrides in the cladding tube.

4. Visual inspection

The visual inspections of the fuel assemblies are carried out using a pool-side inspection equipment in the spent fuel storage of Loviisa 2 unit. The equipment consists of a periscope and a camera. A video recorder can be installed to the periscope. There is also tooling to dismantle the fuel assembly channel, which would otherwise prevent the visual examination of the rod bundle of VVER-440 fuel. To examine individual fuel rods more closely the lower end plugs of the fuel rods in the outer row of the assembly may be cut above the locking wire enabling lifting and rotation of the rods.

5. Structure of the fuel assembly

The 126 fuel rods of VVER-440 assembly are attached with a locking wire to a lower support plate and supported by ten spacer grids and an upper grid. The rod bundle is surrounded by a hexagonal channel fixed to the upper and lower nozzles. Figure 3 shows the structure of the assembly schematically as well as the denotation used for the spacer grids, fuel rods and the six sides of the assembly.

Observations

1. Outlet temperatures of fuel assemblies

After approximately one month's operation of cycle no. 15 of Loviisa 2 unit the coolant outlet temperatures of several fuel assemblies started to rise in an unexpected manner. The phenomenon was more pronounced in the six ZR assemblies, which had been loaded into the reactor for their first cycle. Within 10 weeks the extra temperature rise of the most severely affected assembly contributed to ca. 10 % of the total temperature rise, which indicates 10 % reduction in the coolant flow through that assembly. Figure 4 shows the changes in the coolant outlet temperatures for ZR assemblies, for the first cycle SS assemblies and for few of the most affected SS assemblies during the first 14 weeks of cycle no. 15. It is worth to notice that the largest temperature increase was experienced by all four ZR assemblies, for which the outlet temperatures were measured. Even the largest temperature increase in the SS assemblies was approximately 50 % of those in the ZR assemblies. There was no indication of anomalous outlet temperatures in the first cycle SS assemblies.

Another interesting feature related to the anomalous outlet temperatures was that they were observed only in the other half of the reactor core. Figure 5 illustrates those core locations, in which increased temperatures were observed. Within the area of increased coolant temperatures there were still fuel assemblies, which showed no anomalous changes in their outlet temperatures. These assemblies were mainly SS assemblies operating on their first cycle in the reactor.

Towards the end of cycle 15 the rate of outlet temperature increase in affected assemblies slowed down and essentially stopped. During the 16th cycle very low temperature increase was still noticed in a number of assemblies, which were located in the "crudded" part of the core during the 15th cycle.

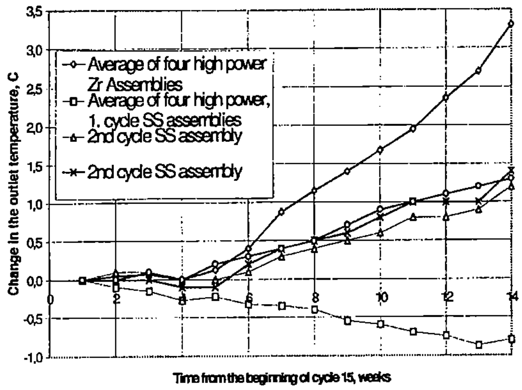


Figure 4. Outlet temperature changes in the beginning of 15th cycle

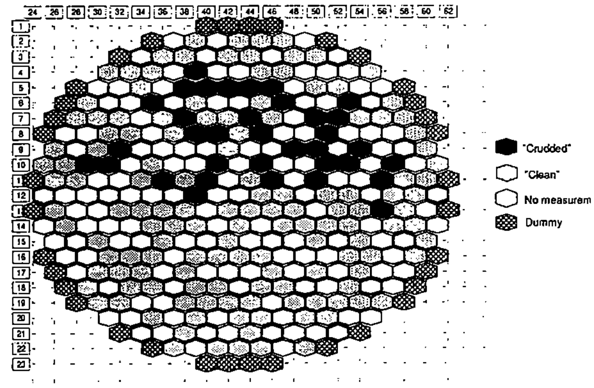


Figure 5. Location of observed increased outlet temperatures (after discharge of ZR assemblies)

2. Flow resistance measurements of fuel assemblies

Based on coolant outlet temperature measurements during the 15th cycle 80 fuel assemblies were selected for flow resistance measurements, which were carried out during the refuelling shutdown between the 15th and the 16th cycle in August, 1995. The selection was based on:

- location in the reactor:
  - assemblies in region of anomalous outlet temperatures,
- irradiation time:
  - assemblies planned to be discharged finally from the reactor were not chosen,
  - assemblies irradiated only for one cycle were not chosen (with some exceptions) because such assemblies did not show anomalous increase of outlet coolant temperature.

In addition, four unirradiated, clean fuel assemblies (three fixed fuel assemblies and one fuel follower) and five clogged ZR assemblies were measured as reference cases. Flow resistance K of a single fuel assembly was compared with the flow resistance of the reference (clean) fuel assemblies. Fuel assemblies with $K/K_{clean} > 1,35$ were rejected from further operation. The results of the flow resistance measurements are summarised in Table 1.

The results of the flow resistance measurements coincide well with the observations of the outlet temperature behaviour:

- crud build-up was clearly more intensive in the ZR assemblies than in the SS assemblies. Although the ZR assemblies had been operated only for 3,5 months compared to eleven months' operation of the first cycle SS assemblies, the flow resistance of the ZR assemblies was approximately 40 % higher than the flow resistance of the SS assemblies.
- crud build-up in the SS assemblies seemed to be clearly more intensive in those assemblies, which had been operated one or more cycles before the 15th cycle compared to assemblies which operated on their first cycle.
- only one half of the Loviisa 2 reactor core seemed to be affected by the crud build-up.

Fuel assembly group	Range of K/K_{clean}	No. of measured assemblies /no. of rejected assemblies
ZR assemblies (after 3,5 months operation)	1,45...1,72	5/5
Fixed SS assemblies after 1 cycle operation in the "crudded" part of the core	1,12...1,12	2/0
Follower SS assemblies after 1 cycle operation in the "crudded" part of the core	0,97...1,06	2/0
Fixed SS assemblies after 1 cycle in a clean core and one cycle in the "crudded" part of the core	1,08...1,50	45/21
Follower SS assemblies after 1 cycle in a clean core and one cycle in the "crudded" part of the core	1,25...1,59	8/5
Fixed SS assemblies after 1 cycle in a clean core and one cycle in the area between "crudded" and "clean" part of the core	1,00...1,25	17/0
Follower SS assemblies after 1 cycle in a clean core and one cycle in the area between "crudded" and "clean" part of the core	1,01	1/0

Table 1. Summary of flow resistance measurements after cycle 15 of Lovisa-2

Failed fuel assemblies in the beginning of 16th cycle were all slightly "crudded" assemblies with K/K_{clean}-value varying mainly between 1,20...1,30 (1,11 for one assembly). The flow resistance measurements were repeated after the 16th cycle in September 1996 to verify the K/K_{clean}-values for those assemblies which were planned to be operated for another cycle. K/K_{clean}-value (1,15 was set as a criterion for further operation. Measurements showed that the crud build-up had continued also during the 16th cycle, but with a much lower rate than in the 15th cycle. Altogether 12 fixed assemblies and 4 fuel followers exceeded the acceptance criterion and were replaced by clean assemblies, which had been discharged as planned from Loviisa-1 and Loviisa-2 earlier. The maximum K/K_{clean}-value in the discharged assemblies was 1,22.

3. Ultrasonic inspections of two leaking SS assemblies

Failed rods in two of the leaking SS assemblies, later called SS-1 and SS-2, were identified using ultrasonic inspection device in order to clarify their failure mechanism by visual inspection later on.

Only one rod in assembly SS-1gave indication of a defect in the ultrasonic inspection. This rod was D7 located inside the assembly in the fourth rod row from outside. Rod D7 was not visually inspected since not more than two outer rows of the rods are usually detached from the fuel assemblies for visual inspection.

In assembly SS-2 altogether 12 rods gave indication of a defect. The leaking rods were mainly corner rods or rods next to them. Two of the defected rods were located inside the assembly in the second and third rows. All defected rods (A1, E1, F1, G1, K5, L6, L8, M13, H13, D8, A7 and A2) are depicted in Figure 6.

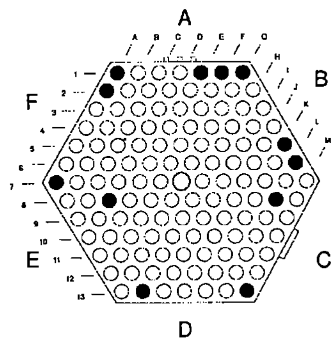


Figure 6. Leaking fuel rods in assembly SS-2

4. Visual inspections

Up to now visual inspections of three fuel assemblies with increased outlet temperature and flow resistance values have been carried out. First inspected assembly was a ZR assembly (ZR-1) discharged from the reactor after 3,5 months' operation. The two other assemblies were SS-1 and SS-2, which had started their third cycle (16th cycle of Loviisa 2) but were found out to be leaking and discharged from the reactor during shut down after one month's operation of cycle 16.

4.1. Outer appearance

The outer appearance of all assemblies was more or less similar. The lower nozzle was covered with a very thin, reddish brown layer of corrosion products. There was a 50 - 100 mm high zone around the shroud tube, just above the lower by-pass holes, where the corrosion product layer was thicker, or at least had different colour compared with the areas around it. Rest of the shroud tube was covered with a uniform, grey coloured layer of corrosion products. In the upper part of the shroud tube, approximately at the same height as the upper end of fuel pellet column, the uniform grey coloured surface ended changing to a narrow belt of reddish brown colour. In the upper part of the shroud tube (200...300 mm) there were regions where the grey (crud) layer was spalling off.

4.2. Shroud removal

The shroud tubes were removed from all assemblies without any difficulty. The force needed in the initial phase of pulling out the shroud was some 10 % higher than the normal value, ca. 55 kp. During the pulling the force was normal, ca. 50 kp.

4.3. Inspection of the fuel rod bundles

In the lower parts of the assemblies, a thick deposit of crud covered the lower edges of the spacer grids. In the ZR-1 assembly the crud was deposited on grids no. 1 to 7, with most of the crud covering the 2nd... 4th grids. 9th and 10th spacer grids and the upper grid were practically clean (apart from a thin, dust like layer of crud). Figure 7 shows crud build-up in the 1st...4th and 10th spacer grids of ZR-1 assembly. In the SS assemblies the crud was deposited clearly on the two (SS-2) or three (SS-1) lowermost grids, the others being practically clean, except for the upper grid,

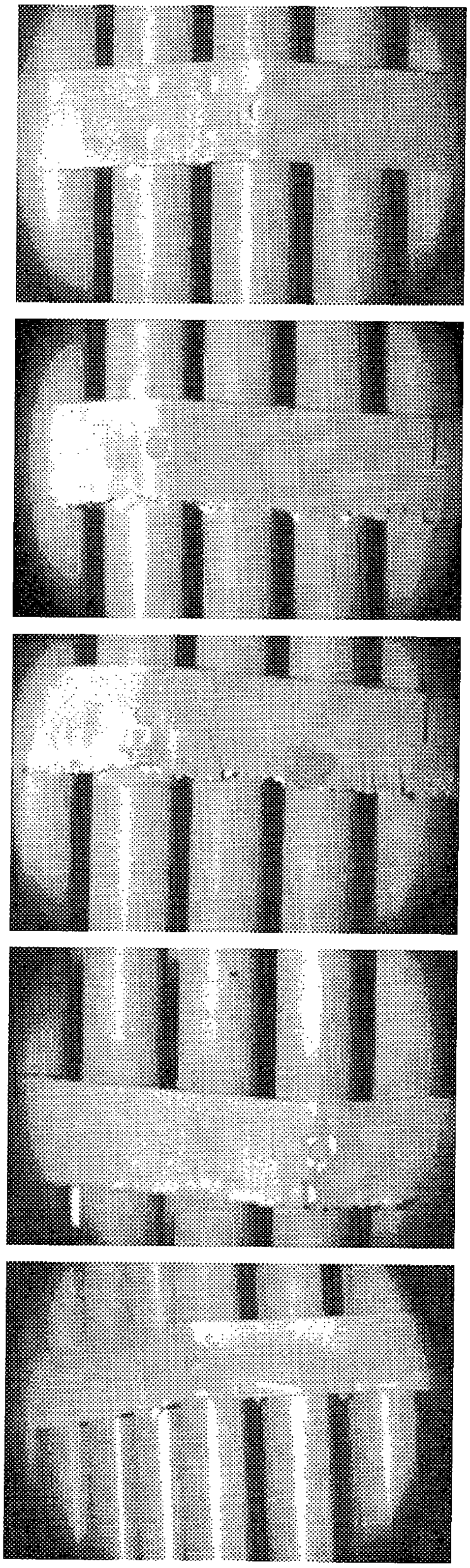


Figure 7
Crud build-up on the 1. 4. and 10 spacer grid of ZR assembly

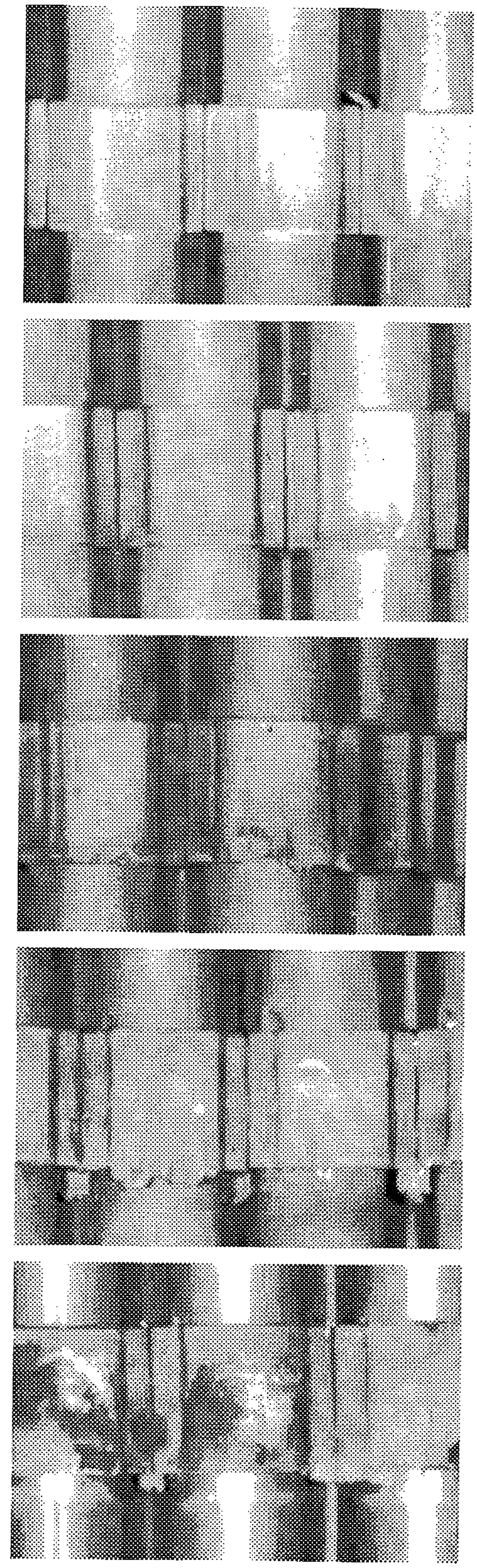


Figure 8
Crud build-up on the 1. 4. and 10 spacer grid of SS-2 - assembly

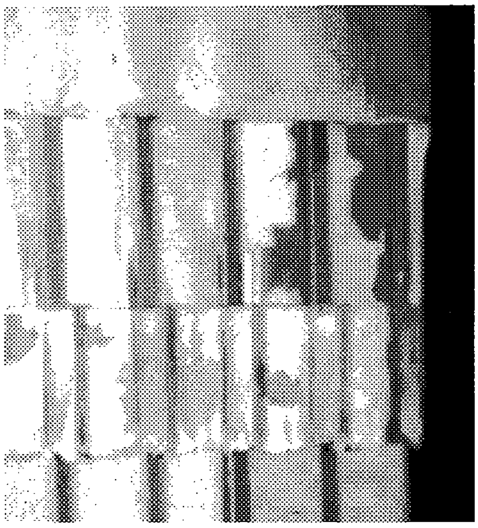


Figure 9
Upper ends of fuel rods in
SS-2 assembly with layers of crud
peeling off

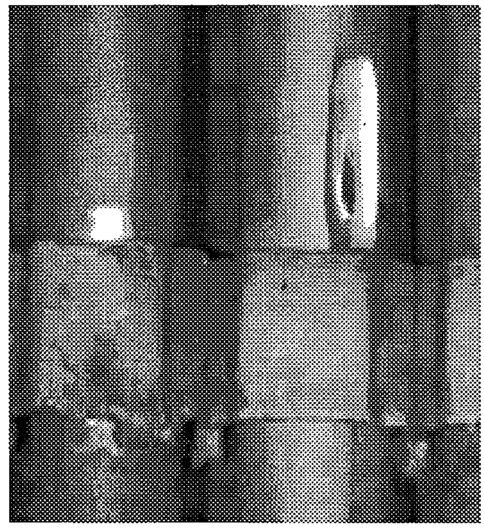


Figure 10
Through wall hole on rod F1 and
fretting mark extending above and
below of gridin rod G1 of SS-2
assembly

which had some crud deposited on the lower edge. Figure 8 shows the crud build-up in the 1st...4th and 10th spacer grids (corner A-F) respectively in SS-2 assembly. Figure 9 shows the upper ends of the fuel rods in SS-2 assembly, which were also covered with crud, which was peeling off in some parts of the cladding tubes.

Most of the crud was deposited on the spacer grids. However, there was also some crud on the cladding tubes on the spot corresponding the lower edge of the spacer grid. The width (height) of this crud layer ranged from half a mm up to some millimetres, the average being approximately two millimetres.

Since it is very difficult to visually inspect the "inside" of the bundle, it is impossible to determine the radial distribution of the crud build-up. The general impression was that the deposit was radially evenly distributed except the very outer rim, which was covered with greater deposition than the inner parts.

The fuel rod cladding tubes were covered with very thin dusty layer of crud. However, the metal gloss could be seen through the layer. In axial direction there was no essential difference in outer appearance of the cladding tubes. There were patch-like layers of crud on one corner rod (G1) of the ZR assembly. The patches were seen between several spacer grids. One explanation might be that these patches have built up as a consequence of an exceptionally small gap between the fuel rod and the shroud tube. The small gap, again, might be a consequence of a slight bow of the fuel rod bundle, or an abnormal shape of the shroud tube corner.

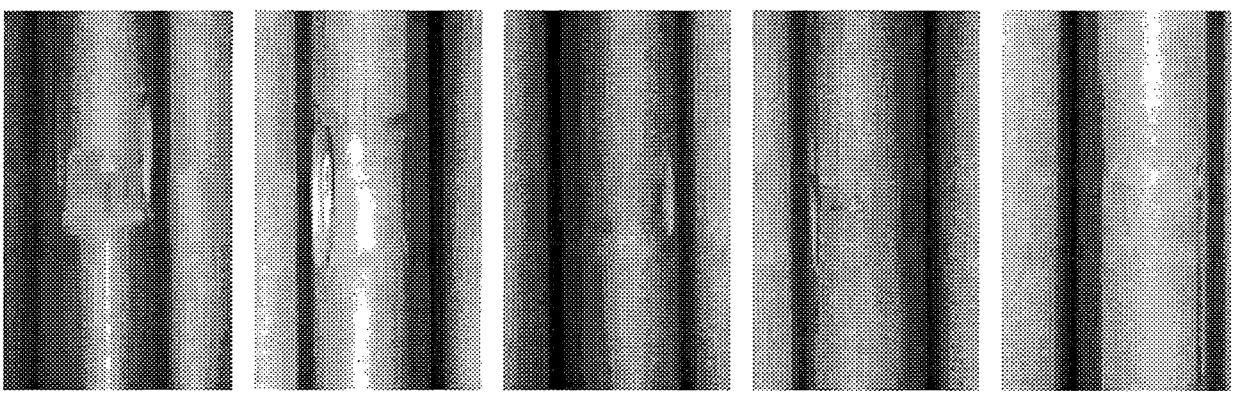
The distances between the fuel rod rows were normal. No rod bowing, nor other kind of blocking of the flow passages between the rods were observed.

4.4. *Visual inspection of defected rods of assembly SS-2*

Part of the failed fuel rods in the outer row of assembly SS-2 were detached from the lower support plate to enable lifting and rotation of the rods. The rods were then lifted up 50...100 mm so that the three contact points between the cladding tubes and the spacer grids became visible.

The largest fretting marks and all of the through-wall holes in the cladding tubes were found under the second spacer grid. The fretting marks were clearly longer (12 mm) than the height of the spacer grid (10 mm) and extended above and below the as-fabricated position of the spacer grids. The through-wall holes were located axially in the centre of the fretting marks beneath the spacer grid bulge against the cladding. An example of a through wall hole on the bottom of a fretting mark on rod F1 is seen in figure 10. The same figure shows a fretting mark on rod G1 extending above and below the 2nd spacer grid.

Rotation of rod G1 revealed two through-wall cracks under the second spacer grid. There were marks of wear also under the first, third, fourth and fifth spacer grids of several rods. Figure 11 shows these fretting marks for rod G1 from the first up to the fifth spacer grid, respectively. These fretting marks had a tendency to become less distinct when moving upwards the assembly. No marks of wear were found under the sixth up to tenth spacer grids.



1 2 3 4 5

Figure 11. Fretting marks on rod G1 of SS-2 assembly at grid contact points of 1. - 5. spacer grids

Above the seventh spacer grid of corner rod A7 there was oxidised/hydrided cladding material or partly loose spots of crud on the cladding tube, which is shown in Figure 12. These spots may be indications of a developing secondary defect caused by hydriding of the cladding (hydride blisters).

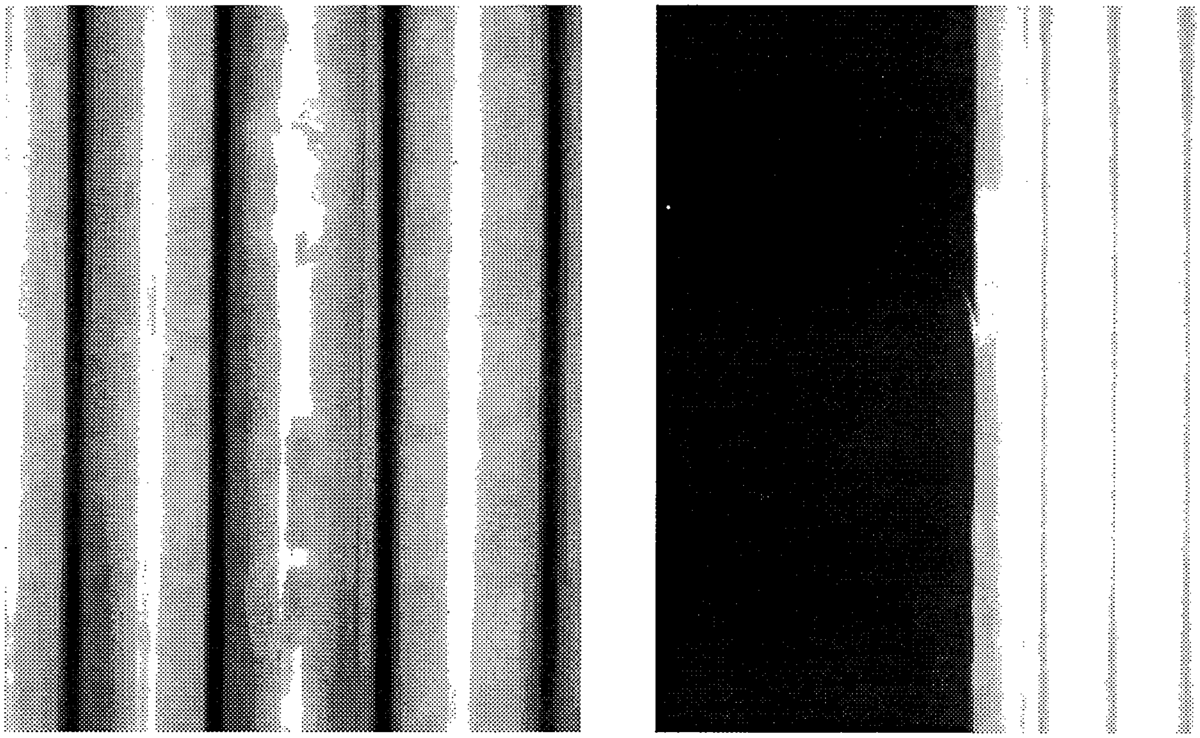


Figure 12. Indication of a developing secondary defect (hydirde blisters) above the 7th spacer grid of rod A7 of SS-2 assembly, photos taken in front of the defect (left side) and from the side (right side).

Discussion

1. Crud build-up

There are several unanswered questions concerning the basic reasons for this crud build-up phenomenon as well as concerning the detailed mechanisms of the crud build-up.

It is evident that during the build-up process the source of crud has been uniformly distributed in the primary coolant due to rather efficient mixing of coolant between the six loops. Yet, the crud was deposited only on assemblies located in a certain part of the Loviisa-2 reactor. Therefore, it is assumed that there must have been a separate initiating event which has "contaminated" fuel assemblies in a restricted area of the core. After that, crud has started to build up in these "contaminated" assemblies.

Another unanswered question is, why the "contamination" is more effective in the ZR assemblies than in the SS assemblies. It is also unclear why the first cycle SS assemblies (at the time of initiating event) were practically not at all "contaminated" contrary to second and third cycle assemblies. Latest flow resistance measurement results from September 1996 show, however, that crud build-up was systematic also in SS assemblies, which on their first cycle in the "contaminated" area of the core. The crud build-up rate in these assemblies has been significantly slower than in the second and third cycle assemblies.

Following factors may have influenced the mechanisms of clogging:

- decontamination of primary circuit of Loviisa 2 unit prior to the 15th cycle.
- growth of new oxide layers on the primary circuit surfaces after the decontamination during and after the start-up of Loviisa 2 unit, which has led to higher metal concentrations in the primary coolant
- influence of other actions taken during the outage.
- somewhat different structure of ZR and SS grids, ZR grid having an "extra" hoop around it.
- different surface properties (?) of ZR and SS spacer grids on one hand and of fresh and irradiated grids on the other hand.

During the refuelling outage in 1994 the whole primary circuit of Loviisa 2 unit was decontaminated [1] During the decontamination there was extensive repair and inspection work going on in the vicinity of the primary circuit. Relationship between the decontamination during the refuelling outage and the crud build-up is not clear. The outlet temperatures of the fuel assemblies started to rise only one month after the start up of the reactor following the refuelling outage (and decontamination). At this point some kind of a crud burst was also observed in the primary coolant.

In the case of ZR 1 assembly the spacer grids up to the sixth one were covered with crud the rest of the grids being practically free of crud. The crud build-up of SS assemblies was different. Three (or two) lowest spacer grids of SS assemblies were covered with considerable amount of crud. The rest of the grids, with the exception of the upper grid, were practically free of crud. The difference may be due to the fact that the SS assemblies stayed in the reactor for the whole cycle. The coolant flow may have flushed out some of the crud during the latter part of the cycle after the crud build-up process was nearly stopped.

To clarify the basic reasons for the crud build-up chemical and other analyses of the crud samples were taken from the spacer grids and other parts of the assemblies. The results of these investigations have been reported elsewhere [2].

2. Fuel failure mechanism

The visual inspection of assembly SS-2 showed that the reason for the fuel failures was mechanical wear (fretting) of the cladding at the contact points with the second spacer grid. The wear marks in the peripheral rods extended above and below the grid height suggesting that the spacer grid has been vibrating during operation. It is assumed that the vibration was caused by an increased turbulence in the coolant flow, due to massive build-up of crud in the spacer. Wear of cladding has loosened the grip of the spacer grid from the rod, thus, enabling the vibration of the fuel rods themselves. Rod vibration is demonstrated by fretting marks at spacer grid contact points in the range from 1st to 5th grid.

3. Acknowledgement

The pool-side visual inspections were carried out by Mr. Lauri Lehtiniemi and Mr. Arto Schildt, IVO Research and Development.

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Debris Mitigation Features and Their Impact on Fuel Performance

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1. Introduction

Achieving and maintaining zero-defect fuel reliability is one of the key objectives of the commercial nuclear industry, as well as a significant factor in maintaining low plant operating costs. The identification of fuel rod leakage mechanisms and the subsequent development of corrective actions to eliminate such mechanisms has been the method used to achieve zero-defect fuel reliability in operating PWRs. Debris-induced fuel rod fretting has been one such significant mechanism observed throughout the PWR fuel industry. Debris-induced fretting is caused by metallic material (small turnings from repair operations, loose wires, etc.) which passes through a fuel assembly's bottom nozzle (end fitting), becomes lodged between the fuel rods, and eventually frets through a fuel rod's cladding. Westinghouse has developed design changes which have enhanced the resistance of fuel assemblies to any debris which enters the fuel assembly. These features are available in Europe through the European Fuel Group (EFG), an alliance of BNFL, ENUSA and Westinghouse.

2. Background

Increases in coolant activity to significant levels were observed in several nuclear power plants in the early 1980s. Site examinations were conducted to determine the root cause of leaking fuel rods so that corrective actions could be performed. Debris-induced fretting was identified as the leakage mechanism at several operating plants. Relatively small pieces of material in the reactor coolant system entered the fuel assemblies through the flow holes in the bottom nozzle and became trapped, either in the bottom structural grid or between the bottom nozzle and the bottom structural grid. The debris, which was in contact with the fuel rods, vibrated due to the coolant flow and eventually fretted through a fuel rod's cladding. The vast majority of all debris defects in Westinghouse-supplied fuel have been observed at or below the bottom grid. The bottom grid acted as an excellent trap for debris and kept the majority of it from moving higher into the fuel assembly. The debris being observed was generally metallic turnings and chips from repair operations performed on components of the reactor coolant system.

At the time debris fretting was first observed, a large number of fuel rods were often observed to be leaking due to this mechanism (as many as 83 rods in one reactor). A summary of the number of debris-induced fretting defects observed in site examinations over the years is illustrated by Figure 1. Each point on the graph represents the number of debris-induced fretting defects observed during a specific site examination. In the early 1980s, between 10 and 30 rods leaking due to debris-induced fretting were sometimes observed after one cycle of operation. The initial thrusts of the corrective

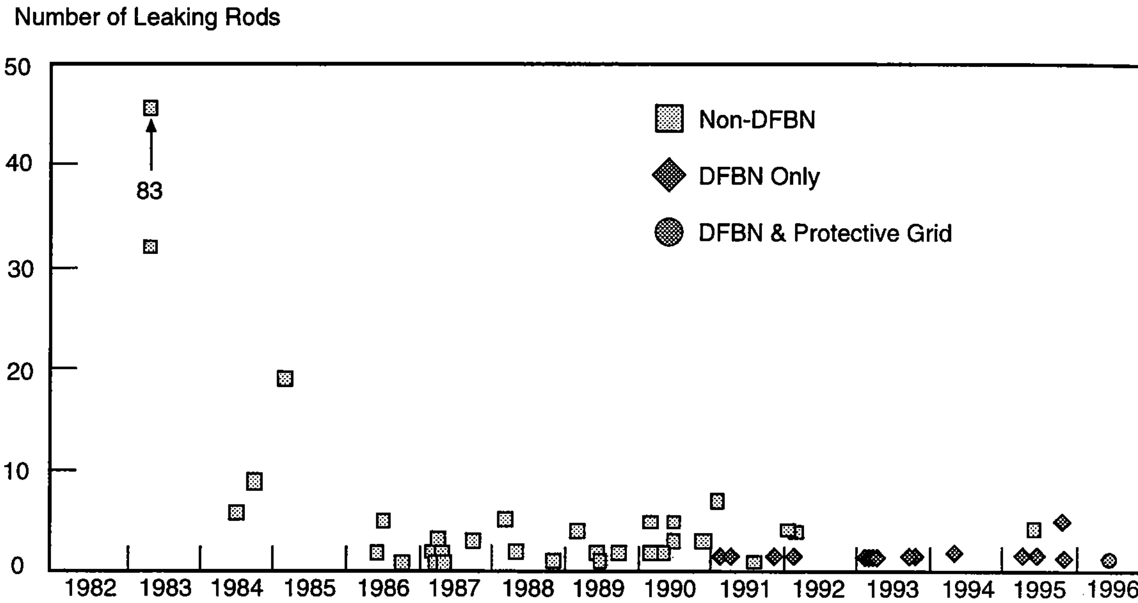


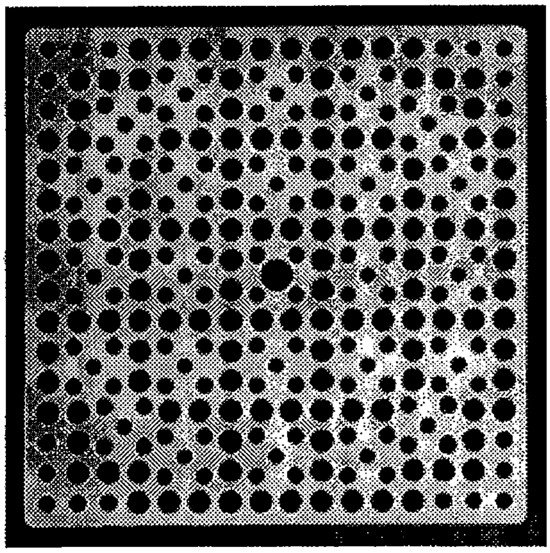
Figure 1. History of Debris-Induced Fretting Defects

actions were to remove debris already present in the reactor coolant system and to prevent the introduction of further debris through development and implementation of detailed Foreign Material Exclusion (FME) practices by utility operators. As seen in Figure 1, these efforts were successful in reducing the number of leaking rods. But debris-induced defects were still being experienced at a significant number of plants, although the number of rods had been reduced to the range of 1 to 10. It was apparent that the fuel assembly itself also needed to be made more resistant to debris-induced fretting in order to eliminate this mechanism.

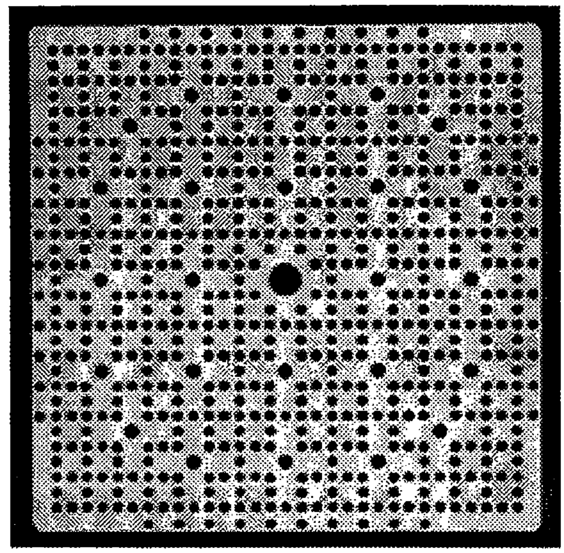
Based on customer input concerning the types of design changes desired, Westinghouse developed the Debris Filter Bottom Nozzle (DFBN). The flow holes in the DFBN were made smaller than those used previously so they would act as a filter and help prevent the debris from entering a fuel assembly. This not only reduced the amount of debris which came into contact with the fuel rods, but also limited its size. Smaller debris is less aggressive and, therefore, less likely to cause a fuel rod fretting defect.

The DFBN was accepted by the industry, and by the early-to-mid 1990s almost all of the Westinghouse fuel products being delivered used DFBNs. The impact of the DFBN can be seen in Figure 1. As the DFBN was implemented, the number of plants with debris-induced leaking rods decreased and the number of leaking rods at these sites also decreased (typically 1 to 2 rods per site). This represented another significant decrease in the occurrence of debris-related defects.

However, the DFBN did not completely eliminate occurrences. In the late 1980s and early 1990s, truly zero-defect operation became a primary industry objective. With this increased emphasis, a second design change was needed. Therefore, Westinghouse developed an additional debris mitigation feature to be used in combination with the DFBN. This consisted of an additional fuel assembly (protective) grid, located directly above the bottom nozzle. This grid also acts as a filtering device to further prevent debris from entering the fuel assembly. The protective grid traps the debris in an area of the fuel rod where it would contact a solid end plug (as opposed to contacting the active fuel portion of a rod). This prevents the debris from fretting through the cladding and penetrating the fuel rod. In addi-



Previous Bottom Nozzle



DFBN

Figure 2. Flow Hole Pattern of DFBN Compared to Previous Bottom Nozzle

tion to benefits of debris mitigation, the protective grid also provides additional capability to resist other (flow-induced) grid-to-rod fretting mechanisms.

The development and testing of the DFBN and the protective grid are discussed below, along with the operating experience of each.

3. Debris Filter Bottom Nozzle (DFBN)

Prior to the development of the Westinghouse DFBN, meetings were held with customers to determine requirements they would have for design changes to make the fuel assembly more resistant to debris-induced fretting. Several concepts were presented and input solicited. From these comments some key design requirements were generated. The first requirement was that the change be as transparent as possible. Examples of this were a minimum delta-pressure difference in the fuel assembly (preferably none), minimal licensing impact, and no new parts, if possible. A second requirement was that a new design not become a debris-generator in and of itself at any point in time. This led to the elimination of concepts that involved screens and wires as filtering devices. The design concept selected consisted of a modification to the bottom nozzle. This resulted in the introduction of no additional parts on the assembly and had no small parts which could become debris. By requiring the pressure drop in the bottom nozzle to match the previous design, operational and licensing impacts were essentially eliminated.

The DFBN concept selected was to reduce the diameter of the flow holes through the bottom nozzle while increasing their number in order to maintain the required fluid flow through the assembly and to match the pressure drop of the previous design. Maintaining the same pressure drop insured that there would be minimal crossflows generated between adjacent assemblies in the core (which had different bottom nozzles) during transition fuel cycles. The area of the flow holes was reduced by a factor of four to six. The original and DFBN designs are shown in Figure 2.

The benefit of the DFBN in trapping debris was verified by performing flow tests. Debris typical of that previously found in fuel assemblies and in the reactor vessel during refueling outages was placed in a flow loop to determine if it would pass through the DFBN. Only debris which had been previously shown to pass through the original bottom nozzle was used in the test. The results showed that approximately 90% of the debris was trapped by the DFBN. The more meaningful data, however, came from in-reactor performance. A leakage rate due to debris-induced fretting for fuel of DFBN design built over the past 3 to 4 years is less than 1 per 100,000 rods fabricated. Although this was consistent with the expectation of the initial design, the increased industry desire for zero-defect fuel reliability led to the development of the protective grid debris mitigation feature.

4. Protective Grid/Long End Plugs

The second feature developed was a protective bottom grid, to be used in combination with the DFBN. As was the case with the DFBN, there were significant interactions with our customers during the conceptual stage. The feature was intended to trap any debris which could pass through the DFBN before it could come into contact with the fuel rod itself. The protective grid is placed below the bottom structural grid of the fuel assembly, basically resting on top of the bottom nozzle. The configuration of the protective grid is shown in Figure 3. There are two key features which trap debris. First, the grid straps of the protective grid are positioned so that they intersect the flow holes in the DFBN (Figure 4). The grid straps reduce the flow area by either a factor of 2 or 4, dependent upon whether a grid strap intersection or just a grid strap itself is located over the flow hole. Experience has shown that the majority of all debris which enters the fuel assembly is trapped in or below the bottom grid. The debris would now be trapped either below the protective grid or in the grid springs in the protective grid itself.

Also included with the protective grid are long, solid end plugs on fuel rods. The end plug length is specified so that it extends to the top of the springs in the protective grid. Therefore, any debris which gets trapped in the protective grid will fret against the solid end plug and will not be able to penetrate the fuel rod.

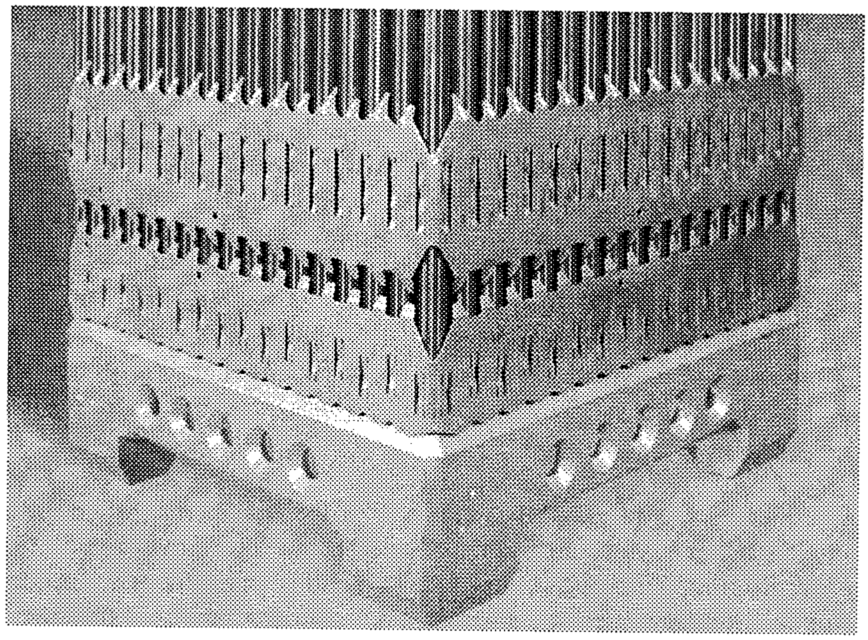


Figure 3. Protective Grid Design

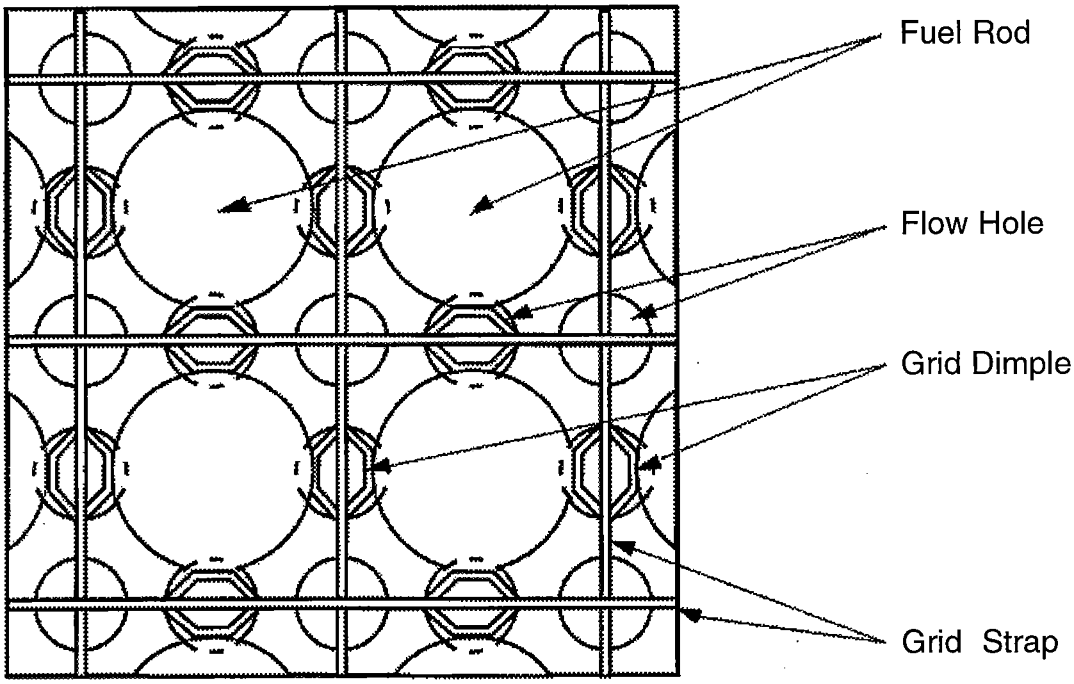


Figure 4. Protective Grid/Bottom Nozzle Interface (Top View)

Flow testing was again performed, to validate the effectiveness of the protective grid. The test focused on its effectiveness in trapping very small debris capable of passing through the DFBN. Several different sizes and shapes of debris were chosen for use in this test program. Much of the debris tested was very small, determined to be capable of passing through the DFBN in a separate test. The tests showed that approximately 60% of such debris was trapped by the protective grid. The debris which passed through the protective grid was so small that it would not be likely to lead to a fretting defect. Combined with the DFBN, the overall effectiveness of the DFBN/protective grid combination is nearing 100%.

One of the advantages of the protective grid over other concepts considered is that it also provides additional protection against grid-to-rod fretting. Very infrequently, grid-to-rod fretting defects in the bottom structural grid span are observed, when a grid cell of one fuel rod is damaged during handling. The bottom span is the most critical, since normal cross-flows at the very bottom of the core are strong enough to cause an unsupported fuel rod to be damaged by grid-to-rod fretting. Since the protective grid also supports the fuel rod at its bottom, handling damage to only one of the two bottom grids (the bottom structural grid or the protective grid) would not lead to grid-to-rod fretting damage.

The protective grid also allowed another fuel assembly design change to be made in order to provide additional margin to possible defects caused by fluid-elastic instability. This mechanism is only a concern at certain plants which have specific cross-flow anomalies. The use of a protective grid, which supports the fuel rod at the bottom, allows the bottom structural grid to be moved up and the second structural grid to be moved down to reduce the span length in the bottom span of the assembly. This changed the vibrational characteristics of the fuel rod sufficiently to eliminate such a flow-induced mechanism. Without the protective grid, the bottom grid could not have been moved up since the length of rod which would extend below the bottom structural grid would increase. It would not be desirable to further increase this unsupported length.

The protective grid was first delivered in 1993. Currently, there are 23 regions of fuel consisting of approximately 1700 assemblies with this feature in operation. To date, only 1 fuel rod has been found to be leaking due to debris-induced fretting in an assembly with a DFBN and a protective grid. The debris defect on this fuel rod was found immediately above a grid approximately 99 inches from the bottom of the fuel rod. This is one of the few cases where debris-induced fretting has been observed anywhere other than the bottom of a fuel rod. It is likely that in this case the debris actually entered the fuel assembly from the top or the side and did not pass through the DFBN and protective grid.

5. Summary

Debris-induced fretting was observed in the early 1980s and has been seen to varying degrees in the LWR industry since then. Initial corrective actions were to clean debris out of the system and to prevent its further introduction. While successful in reducing the number of debris defects observed, these practices did not eliminate the mechanism. It was recognized that the fuel assembly needed to be more resistant to debris-induced fretting. Westinghouse developed the Debris Filter Bottom Nozzle (DFBN) as a first step. The DFBN had reduced-diameter flow holes which restricted the size of debris which could enter the fuel assembly. This change was effective in reducing the number of debris-related leaking fuel rods, but did not eliminate the mechanism. As a second step, a protective grid was developed to be used in combination with the DFBN. The protective grid, located directly above the bottom nozzle, acts as a filter by reducing the effective size of the flow holes in the DFBN. The use of a solid end plug below the grid spring in assemblies with protective grids means that any debris trapped by the protective grid will only be able to fret against a solid end plug and will not, therefore, penetrate the fuel rod. The protective grid also provides additional resistance for other sources of grid-to-rod fretting. The combination of the protective grid and the DFBN provides protection against debris-induced fretting which is approaching 100%.

Development and Experience of Debris Resistant Lower Tie Plates for BWR and PWR Fuel

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1. Introduction

One of the causes for fuel failures during the operation of Pressurized Water Reactors (PWR) is the presence of debris in the cooling system. The debris is typically inadvertently introduced into the reactor during repair or maintenance operations. Boiling Water Reactors (BWR) have significant less incidents by debris. However, it is anticipated that as these reactors age, the amount of debris problems will also increase.

Bits of debris are picked up by the coolant flow and may pass through the lower tie plate into the fuel assemblies. These particles of debris can be trapped between fuel rods in a spacer, usually the bottom spacer. The debris particle will then vibrate under the influence of the coolant flow and can cause fretting corrosion of the fuel rod cladding.

Starting in 1990, Siemens developed debris resistant lower tie plate designs for PWR and BWR fuel assemblies. The tie plate designs have been tested for their various functional requirements. State of the art development methods were applied to optimize the design with respect to performance and manufacturing cost.

Up to now about 2900 Siemens PWR fuel assemblies with debris resistant lower tie plate designs have been delivered and operated with a maximum burnup at 54 MWd/kgU. On the BWR side, 56 assemblies with a maximum burnup of 35 MWd/kgU are under operation. More experience will be gained with the delivery of the first ATRIUM™* 10 reloads in 1996, which incorporates a small hole lower tie plate as a standard feature.

Table 1 provides a summary of the debris fretting failures for Siemens PWR and BWR fuel assemblies since 1990. The debris filter concepts demonstrated their high effectiveness. There was no fretting damage observed attributed to debris passing the filter.

* ATRIUM and FUELGUARD are trademarks of Siemens.

	Operating Year (BOC/EOC)			
	92/93	**93/94**	**94/95**	**95/96**
Fuel assemblies with debris filter				
Number of plants	2	9	21	23
Number of irradiated fuel assemblies	84	600	1.397	2.100
Number of rods failed by debris	0	0	0	2
Fuel assemblies without debris filter				
Number of irradiated fuel assemblies	5.103	4.466	3.790	3.260
Number of rods failed by debris	15	8	12	?

Table 1. Debris Failures on Siemens Fuel Assemblies (BWR and PWR)

2. PWR debris resistant lower tie plate designs

Debris resistant lower tie plates prevent debris from entering the fuel assemblies. A simple and cost effective way of improving the capture efficiency is the reduction of the flow hole size in the tie plate grid. As a standard debris filter, Siemens offers today the Integrated Debris Filter (IDF). This type of filter consists of an array of small square holes arranged in 3 x 3 patterns which are machined with an advanced process technique (Electro Chemical Machining) into the tie plate. Predecessor of the IDF was the Double Grid Debris Filter, a strip grid which was welded to the top of the tie plate.

For very high capturing efficiency demands, Siemens started in 1990 the development of the FUELGUARD™*, a curved blade design. The tie plate consists of a series of blades with curved portions in the middle, which are arranged in parallel. There is no straight line of sight through the curved blade grid, resulting in a highly efficient debris filter for small particles. The FUELGUARD lower tie plate was developed for application in new fuel assemblies as a stand-alone tie plate design as well as an insert into irradiated fuel. In 1993 a batch of 121 irradiated 17 x 17 fuel assemblies was retroactively equipped with FUELGUARD inserts. The total number of FUELGUARD lower tie plates that have been in operation as of April 1996 is 970 including designs for 14 x 14, 15 x 15 and 17 x 17 fuel assemblies (Table 2).

Both debris resistant lower tie plate concepts are presented in Figure 1.

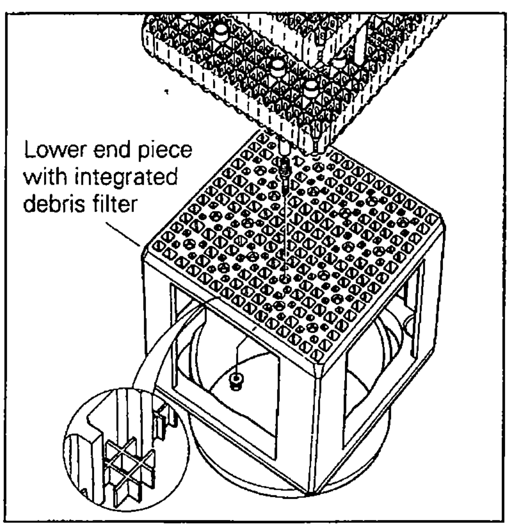
Both PWR debris filter designs meet strength and hydraulic compatibility requirements, the two major design constraints. Strength tests were performed at room temperature and operating temperature simulating seismic and handling loads. Tie plate pressure drop tests were conducted on full scale fuel assemblies.

The measured fuel assembly pressure drop for the IDF is 2 % to 5 % lower and for the FUELGUARD 2 % lower than standard lower tie plate designs (Figure 3).

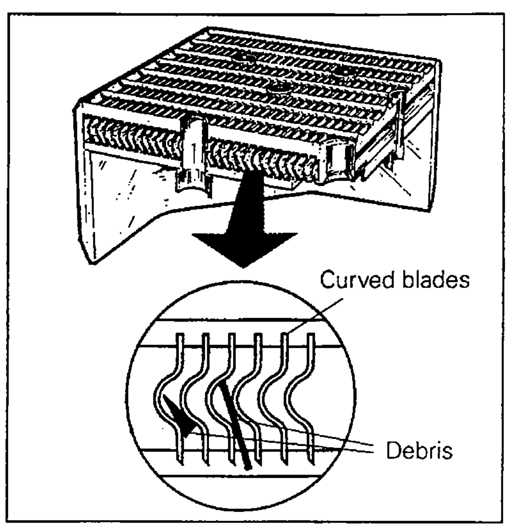
Tie Plate Design	Fuel Design	Number of Fuel Assemblies		Max. Burnup
		in core	cumulative	(Mwd(kgU)
BWR FUELGUARD	9 x 9 10 x 10	20	20	23
BWR Small Hole	10 x 10	36	36	35
PWR FUELGUARD (1)	14 x 14 15 x 15 17 x 17	913	970	29
PWR Small Hole	15 x 15 17 x 17 18 x 18	240	412	51
PWR Double Grid and Integrated Debris Filter	14 x 14 15 x 15 16 x 16 17 x 17 18 x 18	1267	1514	54

(1) includes FUELGUARD inserts

Table 2. Operating Experience with Debris Resistant Siemens Lower Tie Plates for BWR and PWR Fuel (Status 04/96)



PWR 16x16
Integrated Debris Filter



PWR 17x17
FUELGUARD

Figure 1. Siemens Debris Resistant Lower Tie Plate Designs for PWR Fuel

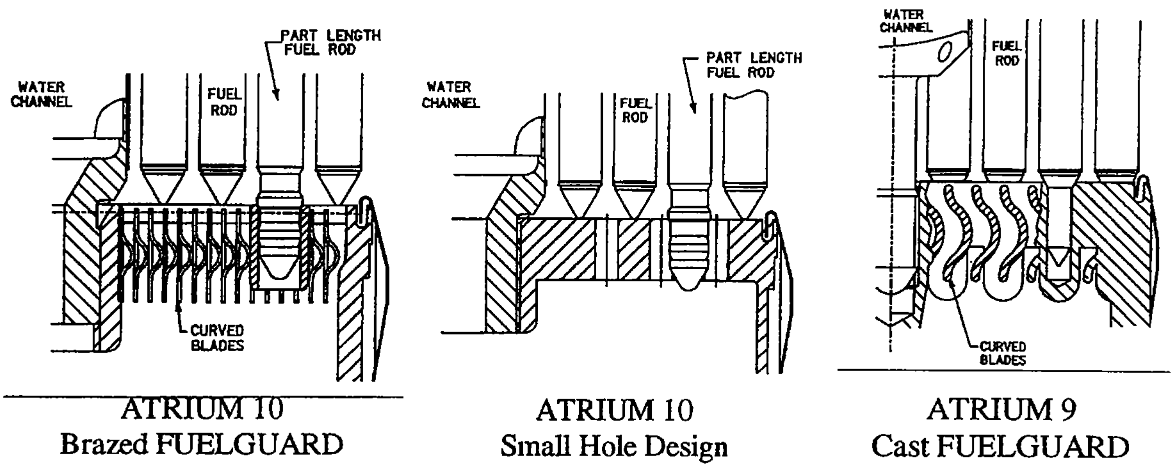


Figure 2. Siemens Debris Resistant Lower Tie Plate Designs for BWR Fuel

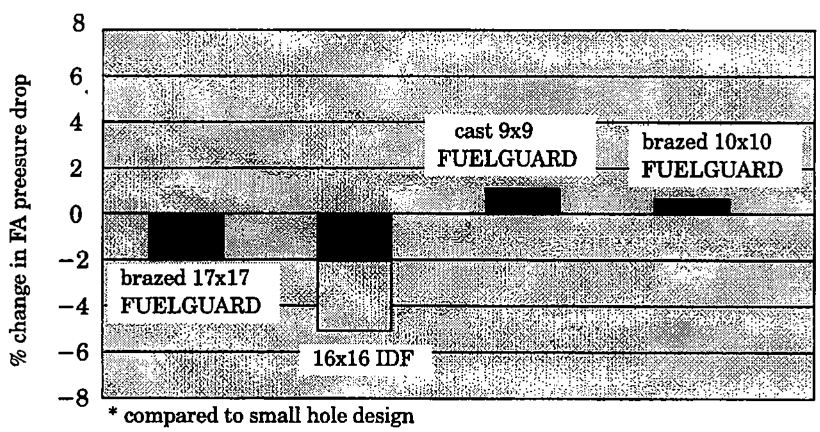


Figure 3. Change in Fuel Assembly Pressure Drop Compared to Standard Designs for Siemens Debris Resistant Lower Tie Plates

In order to determine the **debris capture efficiency** of the two designs, Siemens performed a series of tests at ambient temperature and pressure. Typical debris was identified in cooperation with a US utility and used in a close circulation test loop. Debris capture efficiency was then determined as the percentage of debris pieces captured by the lower tie plate. While the FUELGUARD tie plate achieved an efficiency of more than 90 %, a standard tie plate could filter only about 20 % of the debris in the test loop. For the IDF design, the debris capture efficiency was determined to be 80 % (Table 3).

3. BWR debris resistant lower tie plate designs

For the ATRIUM 10 BWR, two base designs were developed taking credit from previous PWR experience. The brazed FUELGUARD design was further optimized to account for the ATRIUM 10 specific fuel design features. Fuel rod support is provided through large diameter top cross grid rods with flat ground surfaces. Bottom cross grid rods could be omitted without violating strength requirements or significantly reducing debris capture efficiency.

		Lower Tie Plate Design			
Debris Type	Debris Size	Standard BWR or PWR Designs	Small Hole Designs BWR or PWR	Integrated Debris Filter PWR	FUELGUARD BWR or PWR
large particles	diameter > 11mm	X	X	X	X
medium size	diameter > 6mm	O	X	X	X
curled chips and wires small particles	diameter > 3.5mm	O	O	X	X
straight wires and chips	diameter > 2.5mm length > 12mm	O	O	O	X
straight wires and chips	diameter < 2.5mm length < 12mm	O	O	O	O
Total Average Filtering Efficiency		**20%**	**70%**	**80%**	**90%**

O-not filtered **X-filtered**

Table 3. Debris Capture Efficiency of Siemens Lower Tie Plates for DWR and PWR Fuel

In addition an economic small hole design was developed with round machined holes. The hole pattern has been arranged in such a way that sufficient fuel rod support is ensured.

For the ATRIUM 9 BWR a cast FUELGUARD design was selected to avoid excessive Electro Discharge Machining (EDM) work necessary to providing sufficient fuel rod support. The high pressure drop of the cast grid was compensated by optimizing the flow nozzle upstream resulting in only 1 % fuel assembly pressure drop increase compared to an assembly with a standard lower tie plate and nozzle design. Figure 2 compares the three debris resistant lower tie plates for BWR fuel assemblies.

The performance of the BWR lower tie plates are summarized in Table 3 and Figure 3. In BWR applications, the FUELGUARD lower tie plate increases the fuel assembly pressure drop slightly, but not enough to cause flow compatibility concerns. Comparing the debris capture efficiency of the FUELGUARD and small hole design to the standard cast grid, similar results as in PWR applications were achieved.

4. Design optimization for the FUELGUARD using Taguchi experimental methods

The optimization method applied is part of the Total Quality Management (TQM) within Siemens. Starting with a Quality Function Deployment (QFD), the functional requirements of the lower tie plate design are systematically identified. Customer expectations, manufacturing constraints, as well as performance requirements are treated to be equally important. From these functional requirements, a base design is derived and various design factors selected for final optimization. The optimum design factor levels are determined in a series of experiments. For that purpose, levels are assigned to each design factor in a manner, that the expected optimum level is bound. Taguchi's method (Ref.3) provides a method to reduce the number of tests significantly. Instead to testing every combination of factor levels, an orthogonal test array is suggested with well specified factor level combinations. Each specimen, as defined in the array, is tested for each quality attribute (i.e. strength, cost, weight, etc.) and the sensitivities, called „signal to noise ratio" (S/N), are extracted. Large S/N values represent increase in performance. By weighing conflicting quality attributes, the total S/N is maximized leading to an optimum design.

Using the ATRIUM 10 FUELGUARD lower tie plate as an example, this optimization process can be described in more detail. The ATRIUM 10 incorporates some unique design features:

- part length fuel rods attached to the lower tie plate grid
- central water channel as the load bearing structure
- fuel rods resting on the grid surface instead of penetrating the grid

Although credit could be taken from PWR experience (Ref.4), a FUELGUARD grid had to be developed addressing these unique features. As the base design, a brazed grid similar to the PWR design was chosen. Optimum tolerances for brazing, blade height, and blade thickness were selected from this proven PWR design. The optimization goal was then to minimize production cost without a signifi-

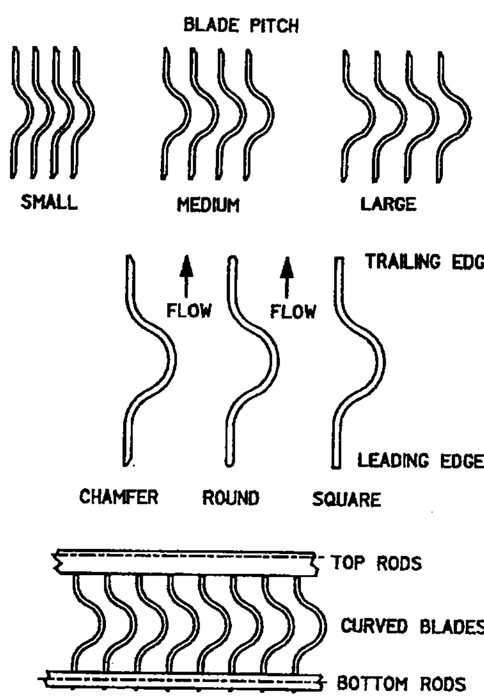


Figure 4. Siemens Debris Resistant Lower Tie Plate Designs for BWR Fuel

cant loss in performance. The design factors for the ATRIUM 10 FUELGUARD lower tie plate are shown in Table 4 and illustrated in Figure 4. With the number of design parameters chosen for the ATRIUM 10 FUELGUARD lower tie plate, a total of 64 experiments would have been necessary to cover all parameter combinations. The Taguchi method reduced the number of experiments to 9 without a significant loss of accuracy. The orthogonal array (test matrix) is presented in Table 5. Each test sample was tested for three quality attributes:

- pressure drop
- debris capture efficiency
- strength

Figure 5 presents the resulting S/N ratio gain for each design factor and quality attribute. It can be seen, that the blade pitch is the dominant factor for all three quality attributes. A small pitch lowers pressure drop and improves debris capture efficiency and strength concurrently. The S/N for all other design factors is small (<1), meaning that the sensitivity of all quality attributes with respect to these factors is relatively small. These other design factors were treated as economic factors resulting in a final ATRIUM 10 FUELGUARD lower tie plate design with significantly reduced manufacturing cost and only little loss in performance:

Factor	Level 1	Level 2	Level 3
Blade Pitch	small	medium	large
Blade Leading Edge	square	chamfer	round
Blade Trailing Edge	square	chamfer	round
Number of Bottom Rods	3	4	7

Table 4. ATRIUM 10 FUELGUARD Design Experiment Factors

Test Sequence Number	Blade pitch	Blade Leading Edge	Blade Trailing Edge	Number of Bottom Rods
1	small	square	square	3
2	small	chamfer	chamfer	4
3	small	round	round	7
4	medium	square	chamfer	7
5	medium	chamfer	round	3
6	medium	round	square	4
7	large	square	round	4
8	large	chamfer	square	7
9	large	round	chamfer	3

Table 5. ATRIUM 10 FUELGUARD Detail Experiment Configuration

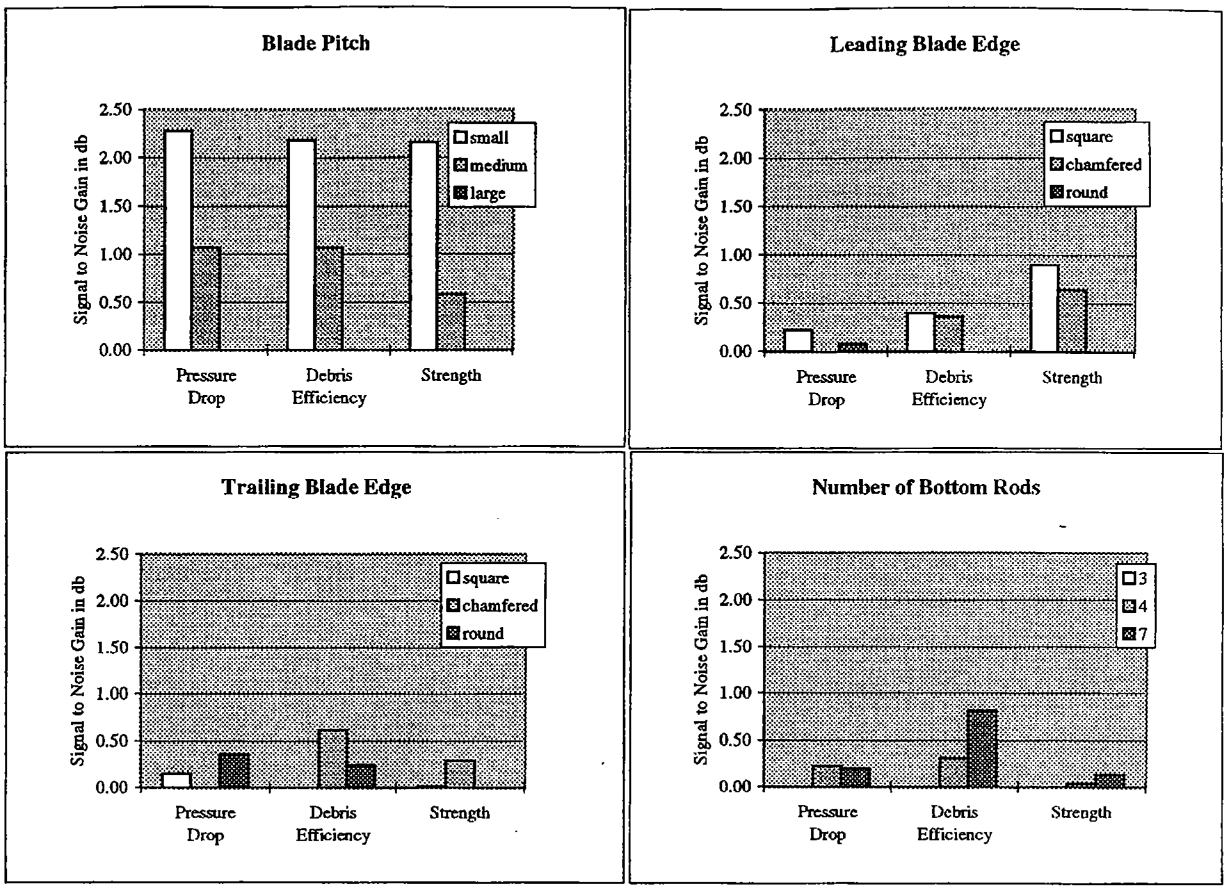


Figure 5. Test results as Signal to Noise Gain for Each Design Factor Level

- small blade pitch for lowest pressure drop, best debris capture efficiency, and highest strength
- square leading and trailing edge for minimum manufacturing cost
- elimination of bottom rods for lower manufacturing cost (accepting a slight disadvantage in debris capture efficiency)

Recent fuel failures in the bottom spacers of PWR fuel assemblies sensitized fuel assembly development at Siemens to any changes in the lower tie plate and bottom spacer region. For that reason, a special test setup was developed **to detect possible flow induced vibrations** on the lower fuel rod end. The test setup was successfully benchmarked against failures in 17x17 fuel by measuring rod vibration amplitudes and correlating them to fretting wear positions. The same test setup was used to verify the ATRIUM 10 FUELGUARD in comparison to the small hole design. The test setup as shown in Figure 6 consists of a full scale 7x7 ATRIUM 10 inlet region including the bottom spacer. One rod location at a time is equipped with a calibrated „flex-tip" with a well known stiffness and dampening. Flex-tip movements in an air flow are recorded through a video camera system simultaneously in two directions. Figure 7 illustrates how debris resistant lower tie plates reduce fuel rod vibrations and actively participate in minimizing the chances of fuel rod fretting in the bottom spacer region.

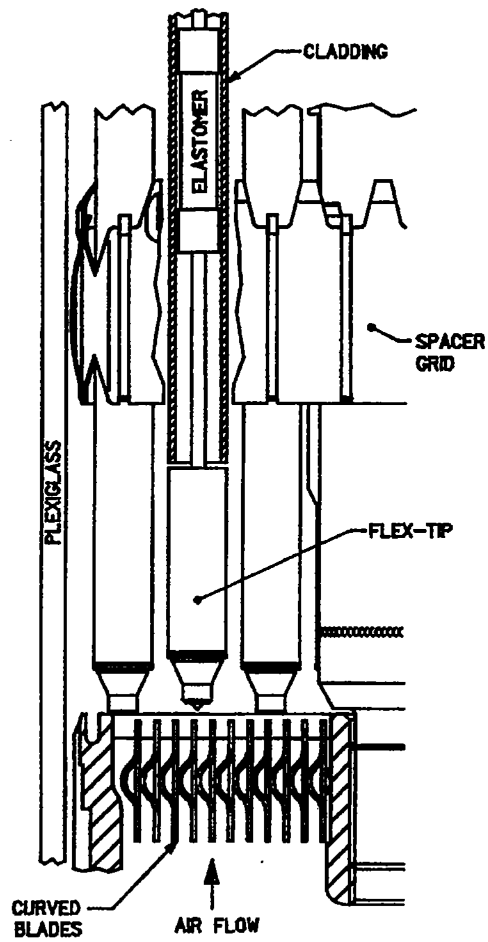


Figure 6. Test Setup for Fuel Rod Vibration Measurements

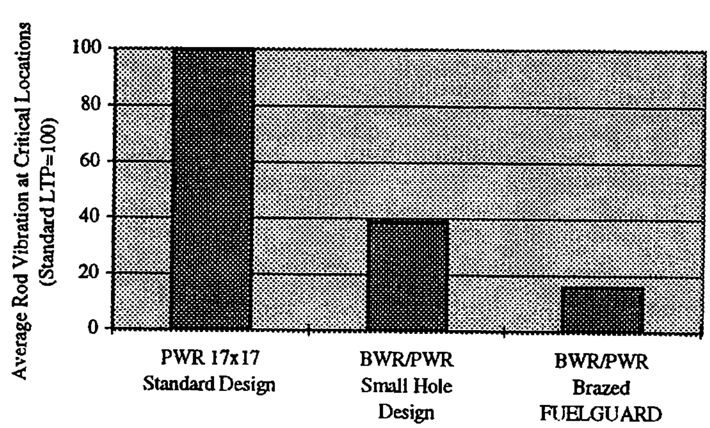


Figure 7. Effect on fuel Rod Vibration for SIEMENS Debris Resistant Lower Tie Plates

5. Summary

In conclusion, Siemens successfully developed small hole and curved blade (FUELGUARD) fuel assembly lower tie plate designs. The small hole designs and the Integrated Debris Filter (IDF) provide good debris capture efficiency at relatively low manufacturing costs while the highly efficient FUELGUARD for PWR and BWR applications reached a debris capture efficiency of approximately 90%. All tie plate designs fulfill the functional requirements of strength and pressure drop. It was shown, that the debris resistant lower tie plates, in addition to their filtering function, also effectively reduce fuel rod vibrations and can consequently help in prevention of fretting wear in the bottom spacer.

For the FUELGUARD lower tie plates, a customer oriented development process was chosen. The resulting optimum design resembles high performance for the various quality attributes, economical manufacturing costs, and robustness against manufacturing process variations.

Experience with about 2900 debris resistant lower tie plates show, that these concepts are effective and provide excellent protection against debris fretting failures.

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The influence of non-penetrating cladding cracks on rod behaviour under transient operating conditions-Data from the international TRANS-RAMP IV project at STUDSVIK

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J. Joseph (Framatome Nuclear Fuel) and
M. Grounes (Studsuik Nuclear AB)

Abstract

The TRANS-RAMP IV (TRIV) Project started in 1989 and was completed in 1993. It was co-sponsored by eleven contracting organizations and was managed by STUDSVIK NUCLEAR AB, Sweden. The TRIV program was backed by results from earlier STUDSVIK ramp projects, especially the DEMO-RAMP II, TRANS-RAMP I and TRANS-RAMP II projects. In these projects it was possible to catch incipient (nonpenetrating) cracks in various stages of penetration of the cladding wall and to suggest a pellet-clad interaction (PCI) failure progression diagram. The time to through-wall crack penetration was short compared to the time for out-leakage of fission products to the coolant water. The results obtained indicated that short fast power reactor transients may lead to formation of incipient cracks in the cladding.

The objectives of the TRIV Project were to study the effects of incipient cracks on rod behaviour, their liability to further propagation due to an irradiation at the same power level as before the transient or to another transient, and above all to study the rod propensity to failure due to pellet clad interaction/stress corrosion cracking (PCI/SCC) of rods containing or not containing incipient cladding cracks when the rods are subjected to a second power transient.

The source of the test fuel rods was full-size PWR fuel rods, which had been irradiated in the French reactor plant GRAVELINES 3 to a bumup of about 28 MWd/kgU. Seven test fuel rods were refabricated as rodlets from the full-size rods by CEA at Saclay, France, using the FABRICE process. The rodlets underwent non-destructive examination at Saclay and also at Studsvik after transport to Sweden.

Four of the test fuel rods were power ramped in the R2 loop N.° 1, using a ramp rate of about 1000 W/cm-min, to get information on the failure boundary curve and on the ramp test data needed to produce incipient cracks in the cladding of the filel rods. The postramp examination results agreed with those from earlier transient test projects, e.g. incipient cracks in failed fuel rods were indicated at pellet-to-pellet interfaces corresponding to linear heat ratings (LHR) close to 40 kW/m and fission product deposits were seen down to a LHR of 35 kW/m. Scanning electron microscope studies showed

that 25 to 30 % penetration of the cladding wall was needed to give a reliable eddy current indication of incipient cracks.

The remaining three test fuel rods were subjected to very similar power transients in the R2 loop N.° 1 at PWR conditions. The ramp terminal level was above the failure threshold established previously and the time to interruption was short (about 40 seconds). The examinations after ramping showed that incipient cladding cracks had been formed only in one of the rods. This rod also showed the largest amount of pellet-topellet dish filling as a result of the transient.

The three rods were then irradiated at PWR conditions in a boiling capsule (BOCA) rig in the R2 test reactor to get a gain in burnup of about 4 MWd/kgU. This irradiation caused a small diameter decrease and development of small secondary ridges. Dish filling was the same prior to and after the irradiation. Comparing the eddy current recordings taken before and after the BOCA irradiation it was found that the changes of the signals were small enough for all the three rods to be classified as insignificant.

The second power ramping of the three rods in the R2 loop N.° 1 resulted in a time to failure for the rod with incipient cracks that was much shorter than for the other two rods. The post ramp examination showed that for all the rods there were a large number of eddy current indications of defects. For the rod containing incipient cracks already after the first ramp the axial position of the largest eddy current indication of defects was the same after and before the second ramp.

From a plot of linear heat rating versus time to failure - or to deliberate interruption of the test - it was seen that the time to failure of the rod, which contained incipient cracks prior to the second power ramping, was shorter than would be expected for a rod going through a first transient, but that the time to failure of the other two rods was comparable to the expected value for a first transient.

1. Introduction

The TRANS-RAMP IV (TRIV) Project was an intemational fuel research project, managed by STUDSVIK NUCLEAR AB, Sweden, and sponsored by eleven separate organizations representing national research centres, safety authorities, fuel manufacturers and power utilities, see Table 1. The project was started in 1989 and its experimental phase was finished in 1992.

The TRIV program was backed by results from earlier intemational STUDSVIK ramp projects, especially the DEMO-RAMP II (DRII), TRANS-RAMP I (TRI) and TRANSRAMP II (TRII) projects [1-3]. In the DRII and TRI projects short length 8x8 BWR fuel rods were base irradiated in the Wuergassen BWR in F R Germany, after which they were individually power ramped in the Studsvik R2 reactor. In the TRII project nominally identical 14x14 PWR fuel rods were base irradiated in the Zorita PWR in Spain, after which they were subjected to rapid power transients in the Studsvik R2 reactor. The transients in the TRII project were aimed to simulate PWR power transients considered typical for a steamline break.

Some information about the fuel rods used in the three projects and the irradiations of the rods are given in Table 2. By the use of intentionally interrupted very short power transients in the R2 reactor it was possible in each of the projects to catch incipient (nonpenetrating) cracks in various stages of penetration of the cladding wall. A summary of the pertinent failure progression data obtained in the DRII, TRI and TRII projects is given in Figure 1. The PCI failure progression diagram shows that incipient cladding cracks form during certain short power transients and may propagate to penetration (failure) of the cladding within about one minute. The fission product out-leakage to the coolant may be delayed, which implies that cladding failures that occur during certain short transients may not be detected until on a later occasion.

2. Project Objective

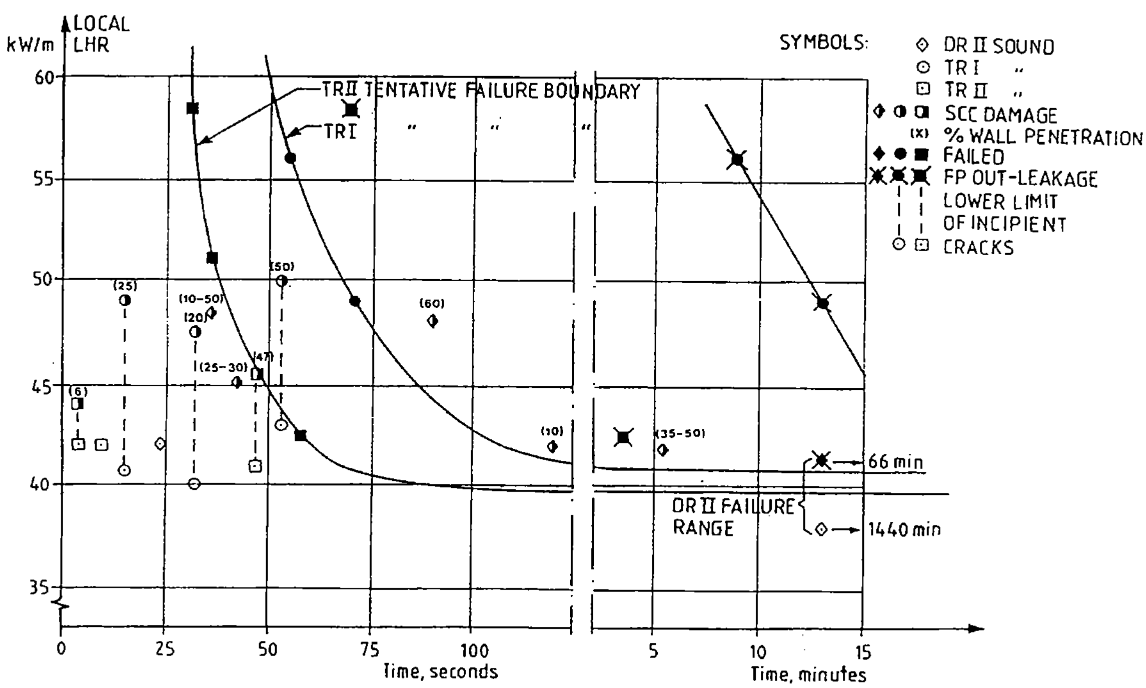
The principal objective of the TRANS-RAMP IV Project was to investigate the rod propensity to failure due to pellet clad interaction (PCI) / stress corrosion cracking (SCC) after continued irradiation of PWR test fuel rods containing non-penetrating (incipient) cladding cracks caused by an earlier power transient.

Belgonucléaire (BN)	Belgium
British Nuclear Fuels plc (BNFL)	UK
Commissariat à l'Energie Atomique (CEA)	France
Electricité de France (EdF)	France
Fragema (FGA)	France
Institutt for Energiteknikk (IFE)	Norway
Nuclear Electric plc (NE), formerly CEGB	UK
Studsvik AB (Studsvik)	Sweden
Swedish Nuclear Power Inspectorate (SKI)	Sweden
Swedish State Power Board (SSPB)	Sweden
US Nuclear Regulatory Commission (NRC)	USA

Table 1. Contracting Parties of the STUDVISK TRANS-RAMP IV Project

Project	DRII	TRI	TRII
Type of rods	8x8 BWR	8x8 BWR	14x14 PWR
No. of rods	8	5	6
PELLET			
Type/powder av. grain size/density	UO_2/7.6 µm/96 % TD		UO_2/5 - 7 µm/93.9 % TD
Outer diameter/height/dish depth	10.6 mm/ 12.0 mm. 0.25 mm		9.3 mm/15.2 mm./0.34 mm
CLADDING			
Material/heat treatment	Zircaloy-2/CWSR		Zircaloy-4/CWSR
ROD PARAMETERS			
He fill pressure	1 - 1.2 bar		500 psia (34 bar)
Outer diameter/clad thickness	12.5 mm/0.86 - 0.87 mm		10.75 mm/0.64 mm
Diametral gap	0.20 mm		0.16 mm
Fuel column length/rod length	314 mm/390 mm		2357 mm/2643 mm
BASE IRRADIATION			
Reactor	Wuergassen		Zorita
LHR average (kW/m)	16 - 30	20 - 22	20 - 22
Burnup (MWd/kgU)	25 - 29	18 - 21	31
RAMP TESTING IN R2			
Conditioning LHR (kW/m)	30	30	20
Conditioning hold time (h)	24	24	6
Ramp rate (W/cm·min)	40 - 220	2000	900 - 1080
Ramp terminal level (kW/m)	38.0 - 48.5	47.5 - 56	42.0 - 58.5

Table 2. The DRII, TRI Projects. Data on Fuel Rods and Irradiations



Note: For reasons of comparison the time of zero is defined as the time when rod LHR exceeds 40 kW/m during up-ramping.

Figure 1. PCI Failure Progressiom Diagram for Some BWR and PWR Test Fuel Rods.

3. Project Scope

The TRIV program embraced the following main activities:

- Design, fabrication and characterization of fuel rods for irradiation in the French PWR plant GRAVELINES 3. This was performed under the direction of the French fuel supplier FRAGEMA prior to the start of the project.

- Base irradiation of fuel rods in the reactor GRAVELINES 3. This part of the program was performed by the French utility EdF.

- Transport of fuel rods from the reactor to Saclay, in France, where CEA refabricated the filel rods into seven rodlets (test fuel rods), using the FABRICE process, and characterized the rodlets.

- Transport of the test fuel rods from Saclay to Studsvik, Sweden.

- Non-destructive examinations of the test filel rods prior to irradiation in the R2 reactor at Studsvik.

- R2 irradiation phase A. Four test fuel rods were used for an attempt to determine the conditions for producing suitable incipient cladding cracks by power transients. The three remaining rods were then subjected to short transients.

- R2 irradiation phase B. The three last ramped rods were irradiated under PWR conditions to get an additional burnup of about 4 MWd/kgU at roughly the same power level as during the base irradiation in the GRAVELINES 3 reactor.

- R2 irradiation phase C. The same three rods were power ramped to about the same peak power level as in the phase A power transient and were held to failure at the ramp terminal level.

- Non-destructive examinations after each of the irradiation phases which the respective rod went through.

- Destructive examinations on some of the test fuel rods using the facilities at the STUDSVIK Hot Cell Laboratory.

- Data processing, reporting of test results and compilation of observations by the organization performing the work for each segment of the work scope.

4. Test Fuel Rods

Seven rodlets (test fuel rods) were supplied to the TRIV Project by CEA, EdF and FRAGEMA. The rodlets had been fabricated by CEA from three long rods (father rods) supplied by FRAGEMA and base irradiated in the EdF power reactor GRAVELINES 3.

The father rods M17, Qll and Q12 were standard 17x17 PWR fuel rods supplied by FRAGEMA. The rods were not individually characterized prior to irradiation, but the rods and the components were controlled and some sampling was performed in accordance with the technical specifications.

After the base irradiation (see next chapter) the father rods were withdrawn from their assemblies and transported to Saclay, where CEA performed non-destructive examinations consisting of visual examination, eddy current testing and axial gamma scanning.

The test fuel rods were then refabricated using sections of the father rods cut out between the positions of the fuel assembly grids. The numbering of the test fuel rods was according to their axial positions, see Figure 2. The design of the test fuel rods is shown in Figure 3, using rod Q12/1 as an example. Rod data are given for the test fuel rods in Table 3. The characterization of the rods included the following examinations, performed at Saclay: axial gamma scanning, diameter measurements at two angular orientations, eddy current testing and visual inspection. The diameter measurements showed the existence of primary ridges (maximum height 20 - 27 µm) and hints of formation of secondary ridges. The diameter values were in the range of 9.42 to 9.45 mm, which, compared with the nominal value of 9.50 mm, implies a clad creepdown in the range of 0.5 to 0.8 %. The eddy current testing and the visual inspection showed the rods to be sound.

The subsequent examinations after transport to Studsvik, outlined in Table 4, agreed with the results obtained at Saclay. From the results it was concluded that the test fuel rods were all very similar and that the gap between fuel and cladding was probably entirely closed already at power in the base irradiation.

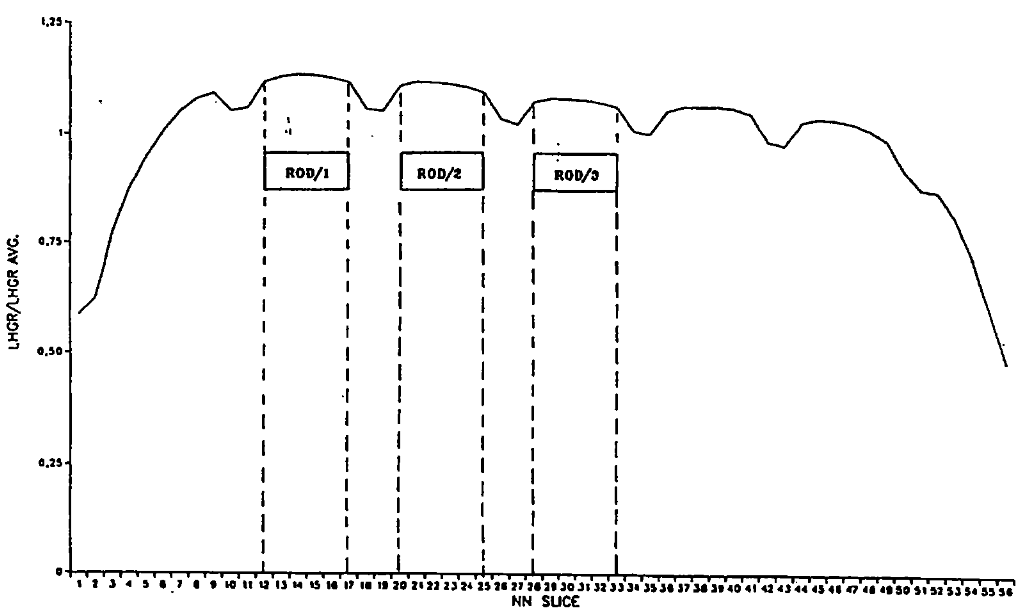


Figure 2. Sections of FRAGEMA (father) Rods Used for Re-Fabrication of Test Fuel Rods

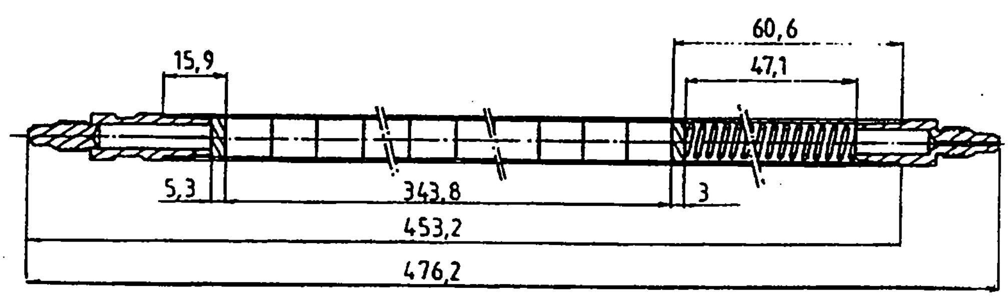


Figure 3. Design of the Test Fuel Rod Q12/1

PELLET	
Type/powder/ grain size/density	UO_2 / IDR / 11 - 15 µm[1)] / 95 % TD[1)]
Outer diameter/height/dish depth	8.19 mm[1)] / 13.46 mm[1)] / 0.3 mm at both ends[1)]
CLADDING	
Material/heat treatment	Zircaloy-4 / Cold worked and stress relieved
ROD PARAMETERS	
Type of rods/He fill pressure	17x17 PWR / 25 bar[1)]
Outer diameter/cladding wall thickness	9.50 mm[1)] / 0.57 mm[1)]
Diametral gap/rod overall length	0.165 mm[1)] / 476.2 - 477.1 mm
Fuel column length	337.2 - 357.4 mm
BASE IRRADIATION	
LHGR average cycle	20.3 - 25.2 kW/m
Burnup (rod average)	23.4 - 29.2 MWd/kgU
Fast fluence (E>1 MeV)	3.94 - 4.86 10^{21}n/cm^2

[1)] Nominal data

Table 3. TRIV Test Fuel Rod (FABRICE) and Base Irradiation Data

5. Base Irradiation

The base irradiation was performed in the standard EdF 900 MWe PWR GRAVELINES 3 during the cycles 3 and 4 from November 20, 1983 to August 31, 1985. The father rod M17 was positioned in the assembly FF06El and the rods Ql 1 and Q12 in the assembly FF06E0. In-core monitoring measurements constituted the base for the calculation of power and fast flux distributions for the rods. The calculations were performed by EdF, using classical reactor physics computation, to give local and average values of linear heat generation rate (LHGR), fast fluence and burnup. The average power history of each of the three rods is presented in Figure 4.

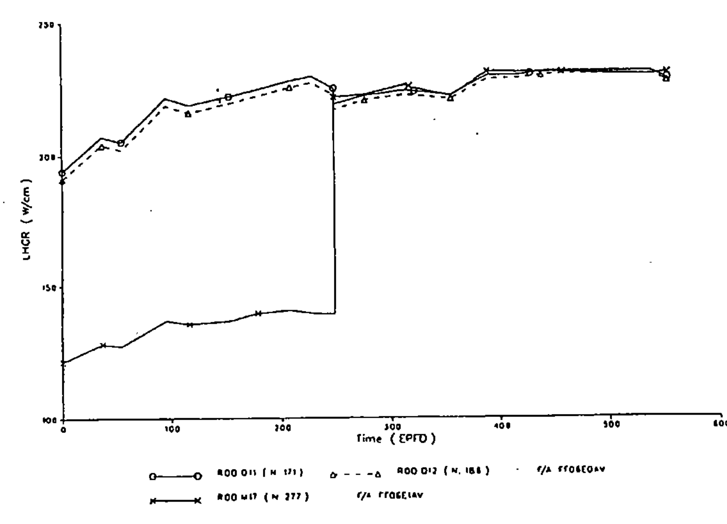


Figure 4. Base Irradiation Power Histories of the FRAGEMA Rods M17, Q11 and Q12

The base irradiation data of the refabricated test fuel rods were deduced from the histories of the father rods and are summalized in Table 3.

6. Irradiations in the R2 Reactor and Examination of Test Fuel Rods

6.1. General information

The three phases of irradiation in the R2 reactor were described under Project Scope. The power rampings were performed using the R2 reactor and the following facilities:

- The loop N.° 1 system
- A sample exchange device
- A ramp test rig with a ramp capsule
- A He-3 absorber system for power control
- Instruments for power measurements, fission product monitoring and rod elongation measurements.

Details of the facilities and the test techniques used have been given elsewhere [4, 5].

The exanmination of the test fuel rods made use of facilities in the R2 pool and in the Hot Cell Laboratory. Standard experimental methods were used for the examinations. The program for the test fuel rods is given in Table 4, which lists the various examinations performed prior to or after each irradiation phase.

Due to the fact, that the results of some of the post-ramp examinations in many cases were vital for the understanding of how the ramp testing had influenced the state of the test rod, this chapter will for each phase of the R2 irradiation combine rod irradiation data and results with supporting examination results.

6.2. R2 Irradiation Phase A

The objectives ofthe R2 irradiation phase A were:

- to get some information on the failure boundary curve by ramp testing in the usual manner to various power levels to get failed or non-failed rods
- to produce suitable incipient cladding cracks in some ofthe test fuel rods by exposing the rods to deliberately interrupted transients to a predetermined power level.

The ramp tests (transients) were all performed in the R2 pressulized water loop N.° 1 with forced circulation cooling at a loop pressure of 146 bar and a coolant inlet temperature of 311 to 314 °C. The ramp rigs used for the tests were He-3 rigs equipped with an axial elongation detector.

The general ramp scheme involved a moderate power ramp rate to a conditioning power level of 25 kW/m held for normally 6 to 7 hours, followed by a rapid ramp to a predetermined ramp terminal level

Rod identification	M17/3	Q11/1	Q11/2	Q11/3	Q12/1	Q12/2	Q12/3
<u>Before R2 irradiation phase A</u>							
Visual examination	x	x	x	x	x	x	x
Gap measurement	x	x	x	x			
Eddy current testing	x	x	x	x	x	x	x
Profilometry	x	x	x	x	x	x	x
Neutron radiography	x	x					
Noise analysis	x	x	x	x	x	x	x
<u>After R2 irradiation phase A</u>							
Visual examination	x	x	x				
Gap measurement	x	x		x			
Eddy current testing	x	x[1)]	x	x	x	x	x
Profilometry	x	x[1)]	x	x	x	x	x
Neutron radiography	x	x[1)]	x	x	x	x	x
Fission gas release				x			
Clad inside inspection	x	x		x			
SEM crack depth	x[2)]						
Metallography/ceramography				x			
Radial gamma scan/scan of Cladding				x			
Noise analysis					x	x	x
<u>After R2 irradiation phase B</u>							
Eddy current testing					x	x	x
Profilometry					x	x	x
Neutron radiography					x	x	x
<u>After R2 irradiation phase C</u>							
Eddy current testing					x	x	x
Profilometry					x	x	x

1) Performed after ramp and re-ramp
2) Performed on two samples

Table 4. STUDSVIK Examination Program for TRIV Test Fuel Rods

and holding at that level either to the occurrence of coolant activity increase due to rod failure, or for a predetermined short time. The irradiation histories were documented by computer plots and chart recordings. The computer plots contained the following data:

- Coolant pressure
- Coolant inlet temperature
- Linear power density at axial power peak
- Rod elongation
- Coolant activity level.

The chart recordings showed the time variation of:

- Reactor power
- Calorimetric rod power Δt (difference between outlet and inlet rig water temperature)
- Rod elongation (arbitrary zero point)
- Coolant activity.

Examples of computer plots are given in Figures 5 and 6 and of chart recordings in Figures 7 to 9.

The ramp test data and results are summarized in Table 5.

Rod M17/3 was ramped to failure as planned. A sudden rod shortening and an increase of coolant water activity were registered during the ramp and failure was also confirmed by the subsequent neutron radiography examination and visual inspection. The eddy current (EC) testing, dad inside inspection and investigation by scanning electron microscope (SEM) revealed indications of defects at regions which had seen a linear heat rating (LHR) above about 41 kW/m. Fission product deposits were seen at both ends corresponding to a LHR of about 39 kW/m. The result of the SEM investigation agreed with the earlier STUDSVIK experience that 25 - 30 % penetration of the cladding is a practical limit for a reliable EC indication of incipient cracks.

Rod Q11/1 was planned to be ramped to the same linear heat rating as M17/3, but reached only 40.8 kW/m at the time of interruption. The subsequent neutron radiography and eddy current testing gave the same result as before the ramp. It was therefore decided, that the rod should be used for an additional ramp test.

The **second ramp test of rod Q11/1**, see Figure 5, resulted in failure, which was later confirmed by neutron radiography and visual examination. Eddy current testing and clad inside inspection showed incipient cracks at pellet-to-pellet positions corresponding to a linear heat rating of 39 kW/m, but large incipient cracks only for positions where the linear heat rating had been above about 41 kW/m.

Rod Q11/3 was ramped by bringing the rod linear heat rating to 46.5 kW/m and interrupting the test 40 seconds after the start of the ramp. There were no indications of failure during the ramp test, and no indications of incipient cracks were found in the eddy current testing or the clad inside inspection.

Rod Q11/2 was ramped according to data in Table 5. Details of the power are shown in Figure 6. The eddy current testing gave indications of defects - the largest one at a bottom position corresponding to a linear heat rating of 43.3 kW/m, and others over the peak power position up to a top fuel position corresponding to a linear heat rating of 40.0 kW/m. The position for the largest indication was considered abnormal. The neutron radiography showed the appearance of the fuel typical for a failed rod containing moisture and with no abnormalities at the position for the largest EC indication of defect. Visual inspection of the rod gave indications of cracks at two axial positions but no clear indication at the position for the largest EC indication. Profilometry of the rod showed, however, a large ridge at this position. Due to these somewhat contradictory data the failure of rod Ql 1/2 may possibly be considered to be atypical.

Rods Q12/1. Q12/2 and Q12/3 were planned to be ramped to a ramp terminal level of 45 + 1 kW/m and to be interrupted 40 seconds after the start of the ramp. The ramp terminal level obtained was in

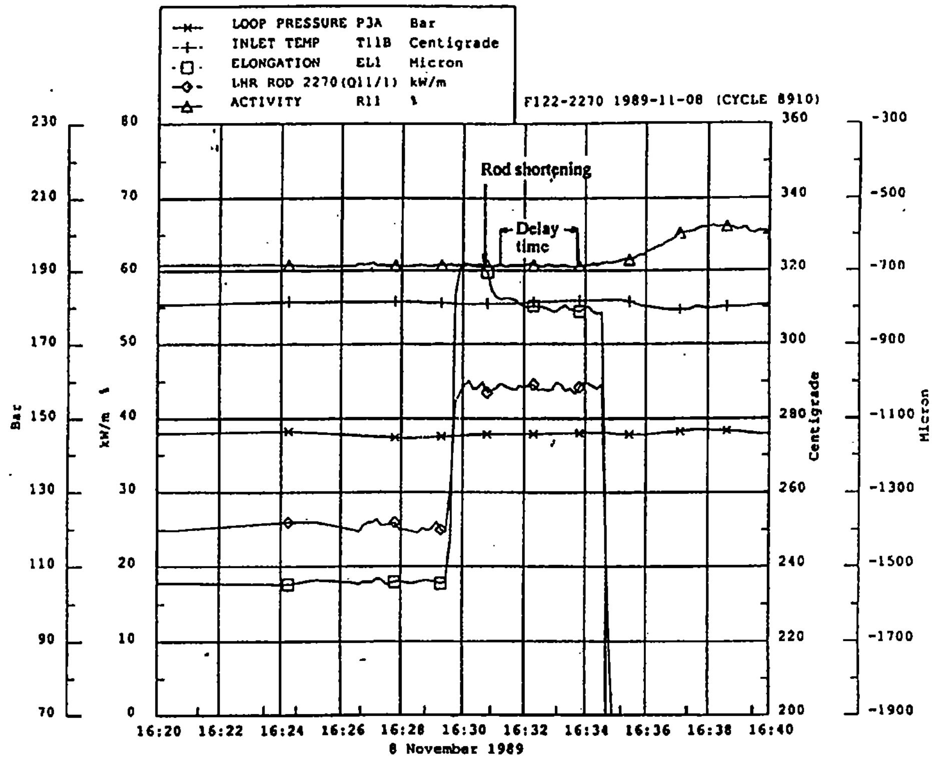


Figure 5. Ramp Test N.° 2 of Rod Q11/1.

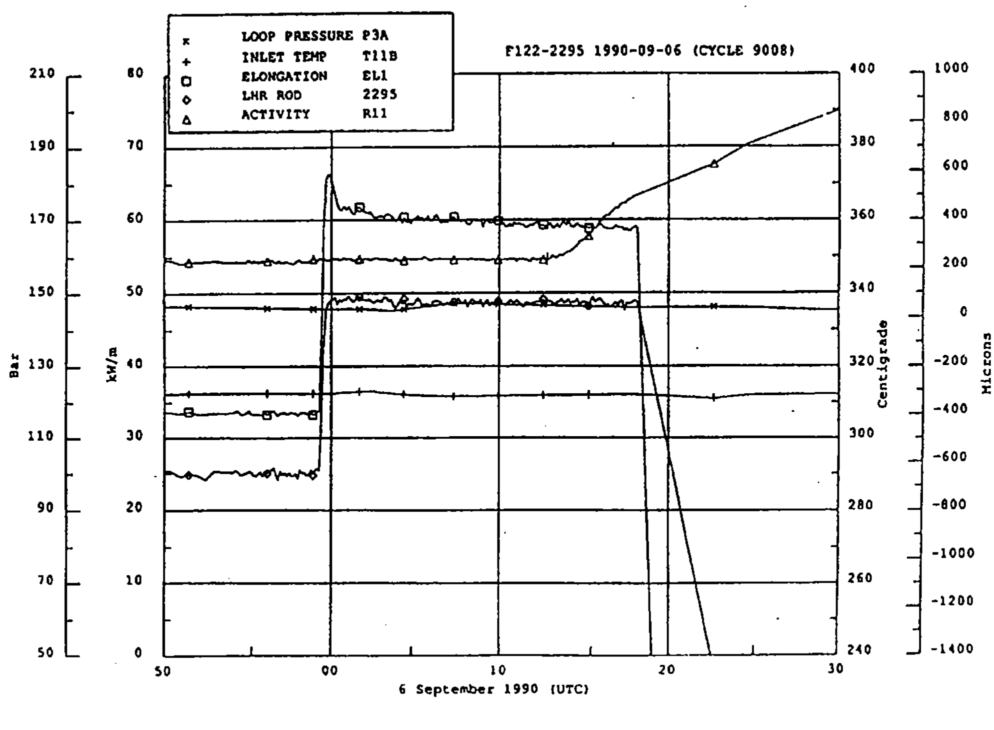


Figure 6. Ramp Test of Rod Q11/2. Details of Power Ramp.

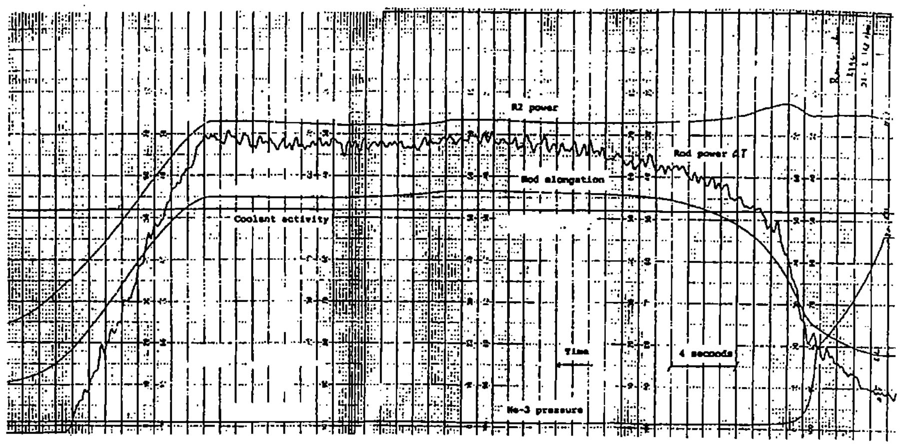


Figure 7. Ramp Test of Rod Q12/1.

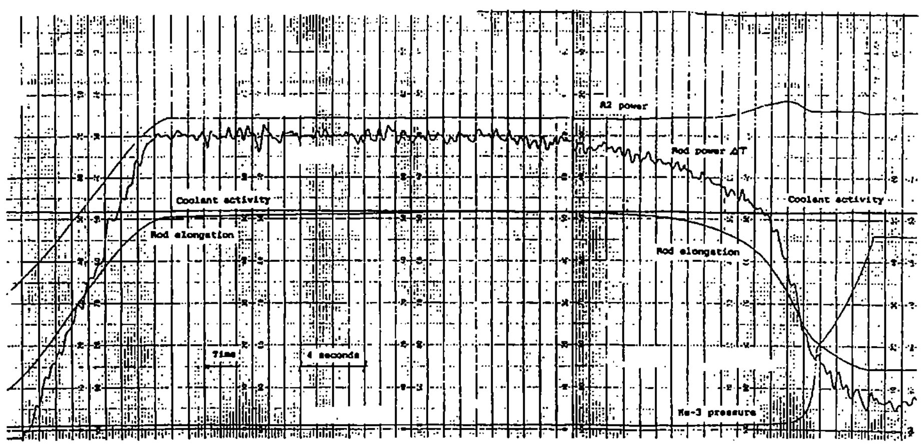


Figure 8. Ramp Test of Rod Q12/2.

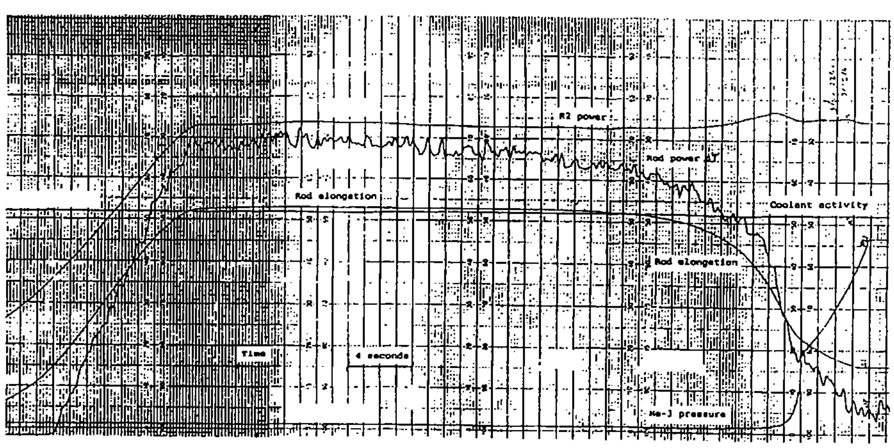


Figure 9. Ramp Test of Rod Q12/3.

Irradiation Phase		A								C		
Project Rod No.		M17/3	Q11/1	Q11/1	Q11/3	Q11/2	Q12/1	Q12/2	Q12/3	Q12/1	Q12/3	Q12/2
Date of Ramping		891017	891018	891108	900601	900906	910212	910213	910214	911107	911108	911112
First/Second Ramp			First	Second			First	First	First	Second	Second	Second
LHR[1] at Conditioning	(kW/m)	25.0	25.0	20.3/ 25.4	25.0	25.0	26.2/ 30.6 25.0	25.0	25.2	25.5	25.8	25.9
Conditioning Hold Time	(h)	6.5	7.0	1.4/ 1.5	3.8	6.1	0.4/ 1.1/ 4.8	6.0	6.2	4.0	2.1	1.4
Position of Axial Power Peak[2]	(mm)	200	200	185	145	140	180	180	185	160	165	180
Ramp Rate[3]	(W/cm.min)	1000	900	900	1100	600	1200	1200	1200	700	750	750
LHR[1] after Ramping[4]	(kW/m)	43.3	40.8	44.1	46.5	49.3	43.6	43.5	44.5	42.5	44.5	45.7
Failure/No Failure		F	NF	F	NF	F	NF	NF	NF	F	F	F
Time from Start of Ramp to Sudden Rod Shortening	(s)	880		70		39				118	78	35
Time from Start of Ramp to Increase of Coolant Activity	(s)	1730		100		665				235	155	170
Time from Start of Ramp to Interruption of Test	(s)	2270	38	300	40	1135	41	43	41	485	665	660

1) LHR = Linear Heat Rating
2) Above bottom of fuel stack
3) Average calorimetric value during the time for He-3 pressure to go from the initial value to 1,2 bar
4) Maximum LHR attained after ramping prior to failure or in case of no failure at interruption

Table 5. Summary of Ramp Test Results of TRIV Test Fuel Rods

the range of 43.5 to 44.5 kW/m and the time to interruption was 41 to 43 seconds, see Table 5. Chart recordings of the ramps are reproduced in Figures 7 to 9.

The neutron radiographs of the three rods after ramping showed differences regarding the magnitude of the remaining pellet to pellet dish depth and the number of interfaces for which the dish had decreased. The result of the microdensitometer scannings of the neutron radiography film is shown in Figure 10. Rod Q12/2 was influenced to the greatest and Q12/1 to the least extent by the ramping.

The eddy current measurements after ramp compared to those before ramp showed insignificant changes for the rods Q12/1 and Q12/3, however, for rod Q12/2 the eddy current testing indicated the presence of incipient cladding cracks after ramp.

Views for the understanding of the differences between the rods after ramping may be offered as follows:

- There may have been unknown differences between the rods prior to ramping which were manifested by the ramping.

- Taking into account the stated accuracy ofthe rod power determination, 2.3 % [4], it does not seem possible to know for certain about the order of the ramp terminal level of the rods.

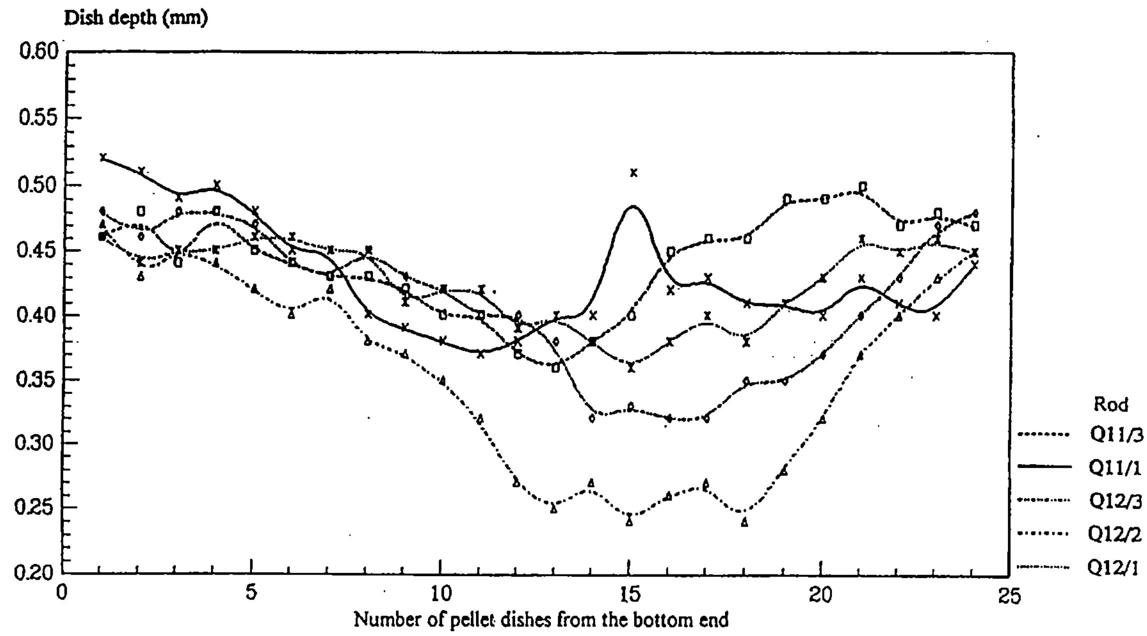


Figure 10. Pellet to Pellet Dish Depth for the Rod Q11/1 after the Second Ramp, for Rod Q11/3 after Ramping, and for the Rods Q12/1, Q12/2 and Q12/3 after the First Ramp.

6.3. R2 Irradiation Phase B

The objective of the R2 irradiation phase B was to study the possible influence of an irradiation at constant power (about the same as before the transient) on the state of the test fuel rods, especially the behaviour of incipient cracks.

The irradiation of the three rods Q12/1, Q12/2, and Q12/3 was performed in a boiling capsule (BOCA) rig positioned in the reactor core. The BOCA system pressure was 146 ± 1 bar. The rod surface temperature was considered to be controlled by subcooled surface boiling, which implied a surface temperature of 338 ± 2 °C over the axial region 80 to 280 mm from the bottom of the fuel stack, whenever the peak linear heat rating was at least 23 kW/m.

The BOCA irradiation took place during five R2 cycles to get an additional burnup of about 4 MWd/kgU. Irradiation data for the test fuel rods are summarized in Table 6.

The examination of the rods after the BOCA irradiation gave the following results:

- Neutron radiography confirmed that the rods were intact, and measurements of the remaining dish depth showed no significant changes when compared to measurements before the BOCA irradiation.

- Eddy current recordings of the rods after the BOCA irradiation were compared to the recordings after the ramp. The changes of the signals were small enough for all the three rods to be classified as insignificant.

R2 Cycle	Time (h) for PR2 > 30 MW	Average LHR (kW/m)			Burnup Gain (MWd/kgU)			Fast Neutron Fluence ($10^{24}/m^2$)		
		Rod Q12/1	Rod Q12/2	Rod Q12/3	Rod Q12/1	Rod Q12/2	Rod Q12/3	Rod Q12/1	Rod Q12/2	Rod Q12/3
9103	427	22.4	22.3	20.9	0.80	0.80	0.75	1.05	1.04	0.98
9104	419	22.6	22.5	21.2	0.81	0.80	0.76	1.04	1.04	0.98
9105	410	24.1	23.9	22.3	0.84	0.83	0.78	1.10	1.08	1.01
9107	401	25.1	24.9	23.9	0.89	0.89	0.85	1.17	1.17	1.11
9108	433	23.3	23.2	21.9	0.86	0.85	0.81	1.12	1.11	1.05
Total	2090				4.20	4.17	3.95	5.48	5.44	5.13

Note: The data refer to the axial peak power position of each rod

Table 6. BOCA Irradiation Data of the Test Fuel Rods

- Profilometry of the rods showed that the BOCA irradiation had resulted in a diameter creep-down of about 5 μm for each of the three rods.

6.4. R2 Irradiation Phase C

Since there were strong indications that rod Q12/2 contained incipient cladding cracks and that there were no such cracks in the rods Q12/1 and Q12/3, the objectives ofthe R2 irradiation phase C could be stated as follows:

- Study if incipient cracks are liable to further propagation when the fuel rod is subjected to a second transient.

- Study if the failure threshold or the time to failure is different for rods subjected to a second transient compared to rods subjected to only one transient and if the presence of incipient cracks influences the rod behaviour.

It was planned to perform the ramp tests for the R2 irradiation phase C as fast ramps to 45 ± 1 kW/m for all the three rods, with a hold at the ramp terminal level long enough to learn about the occurrences of sudden rod shortening and increase of coolant activity.

The same ramp rig with elongation detector was used for the rods Q12/1, Q12/2 and Q12/3 in phase C as in phase A of the irradiation. The loop data were also the same during the two irradiation phases.

The irradiation histories were documented by the same type of computer plots as for the phase A ramp tests and in addition PC plots based on a PC data sampling system for a detailed description of the up-ramping. The PC plots showed the time variation of:

– R2 local power in the vicinity of the loop N.° 1 (SPND)

– He-3 pressure

– Calorimetric rod power DT (difference between outlet and inlet rig water temperatures)

– Rod elongation.

The data contained in the PC plots were only relative, absolute figures were given in the computer plots.

The ramp test data and results are summarized in the three last columns of Table 5.

Rod Q12/1 was ramped to a linear heat rating of 42.5 kW/m which was lower than planned. This was due to the difficulty to achieve the correct rarnp terminal level for the first rod tested by very fast ramping in a new R2 core loading. The other two rods were ramped to ramp terminal levels within the specified range.

The elongation behaviour of the three rods during the first and the second ramp test was studied. It was found that the elongation versus linear heat rating was very similar for the three rods during the

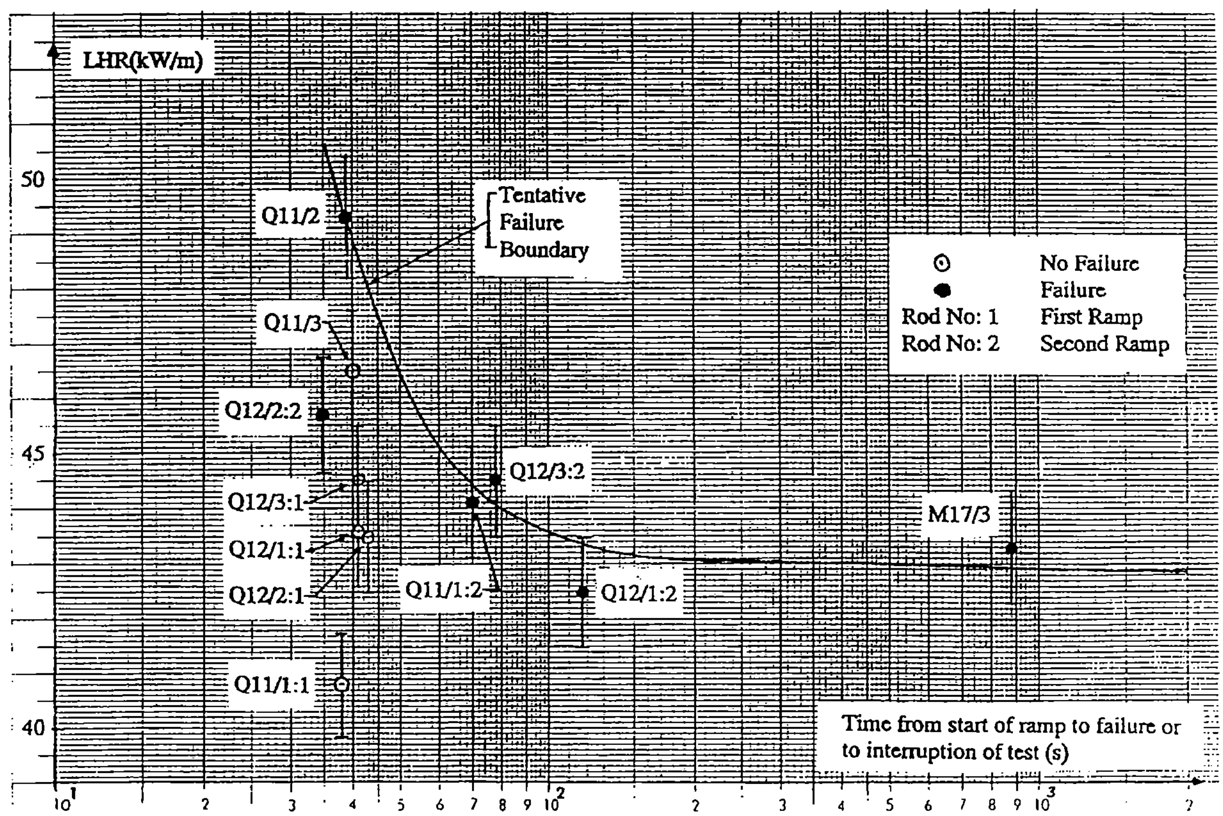


Figure 11. Ramp Testing if TRIV Test Fuel Rods. Linear Heat Rating versus Time from Start of Ramping tod Indication of Failure by Rod Shortening or to Deliberate Interruption of Test.

first ramp and for the rods Q12/1 and Q12/3 during the second ramp. Rod Q12/2 showed, however, a different elongation behaviour already from the start of the second ramp.

The eddy current recordings of the three rods after the second ramp all showed some very large indications of defects. For rod Q12/2 the axial position of the largest indication before and after the second ramp was the same.

In the profilometry after the second ramp a maximum diameter increase of 25 llm was found for the rods Q12/1 and Q12/2 and 40 μm for rod Q12/3. The height of most primary ridges had increased by about 5 μm and with up to 10 μm for the secondary ones. For all the three rods there were some much larger primary ridges - in most cases at the same axial positions as the largest eddy current indications of failure.

For a discussion of the results of the second ramp tests reference is made to Figure 11, where the linear heat rating obtained in the ramp has been plotted versus the time from start of the ramp to indication of failure by a sudden rod shortening or to deliberate interruption of the test. Lacking an established failure boundary curve for rods ramped only once it is not possible to judge if the first ramp has influenced the rod behaviour during the second ramp for the rods Q12/1 and Q12/3. However, from Figure 11 it is seen that within the uncertainties the results of the rods Q12/1 and Q 12/3 may be part of a tentative failure boundary curve for the TRIV test fuel rods. The result for rod Q12/2 shows that the time to failure was shorter during the second ramp than would be expected.

7. Summary of Observations

The main observations resulting from the TRANS-RAMP IV Project are listed as follows:

- Based on the results of the examinations performed prior to power ramping it was deemed that the test fuel rods were all sound and very similar and that the gap between fuel and cladding was probably entirely closed already at power during the base irradiation.

- The power ramping of the first four rods and the subsequent examinations gave only data for a tentative failure boundary curve.

- The examinations of the rods which failed (M17/3, Q11/1 after the second ramp, and Ql 1/2) showed that incipient cracks were indicated at pellet-to-pellet interfaces corresponding to linear heat ratings (LHR) from 39 to 41 kW/m, and fission product deposits were seen down to a LHR of 35 kW/m. SEM examinations showed that 25 to 30 % penetration of the cladding wall was needed to give a reliable eddy current indication of incipient cracks. The differences obtained in the dimensional changes of the rods may be explained as due to differences in the ramp data for the rods (ramp terminal level, hold time and re-ramp or not).

- The examinations ofthe non-failed rods (Q11/1 after the first ramp and Ql 1/3) showed no indications of incipient cracks by the eddy current or the clad inside examination, and there were no signs of fission product deposits in rod Q11/3.

- The last three ofthe seven tests rods (Q12/1, Q12/2 and Q12/3) were power ramped with only small differences what regards ramp terminal level and time to interruption. The examinations after ramping revealed that dish filling had taken place in the rods (a small amount in Q12/1, larger in Q12/3, and largest in Q12/2), and indication of incipient cracks was found in the cladding of Q12/2 but not in Q12/1 or Q12/3.

- The irradiation ofthe three rods Q12/1, Q12/2 and Q12/3 in a boiling capsule (BOCA) rig at PWR conditions and at linear heat ratings in the range of 21 to 25 kW/m to a burnup gain of about 4 MWd/kgU brought about a diameter decrease of about 5 μm, no change of the primary ridges and development of secondary ridges to a maximum height of 5 μm. Neutron radiography and eddy current testing performed after the BOCA irradiation showed no significant changes compared to the data after the power ramping.

- The second power ramping of the three rods (Q12/1 to 42.5 kW/m, Q12/2 to 45 7 kW/m and Q12/3 to 44.5 kW/m) with hold to failure, resulted in a much shorter time to failure for rod Q12/2 than for the other two rods. It was observed that for the rods Q12/1 and Q12t3 the elongation behaviour was thesame as during the first ramp but rod Q12/2 showed a different elongation behaviour already from the start of the second ramp test. For rod Q12/2 the axial position of the largest eddy current indication of defects was the same after and before the second ramp.

- From a plot of linear heat rating versus time to failure -or to deliberate interruption of test- it is seen that for the second ramp test the time to failure of rod Q12/2 was shorter than would be expected for a rod going through a first transient. However, the time to failure of Q12/1 and Q12/3 was comparable to the expected value for a first transient.

- From the results of this program it can not be concluded that the presence of incipient cracks before a power transient increases the propensity for failure.

8. Acknowledgements

The effects of the individuals in the organizations performing the work for each segment of the work scope are gratefully acknowledged. Thanks are forwarded to the members of the Project Committee for a successful supervision of the program.

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The Automatical Reactor Trip with Neutron Flux High Signal during the Earthquake

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1. Introduction

At Onagawa nuclear power plant Unit-1(524MWe BWR,Tohoku Electric Power Co.,Inc.),an earthquake which occured in November 1993 caused the automatical reactor trip with an average neutron flux high signal. The maximum acceleration during the earthquake was about 120gal in horizontal-direction and the Magnitude was about 6.

In order to investigate the reactor trip mechanism, a vibration test with four mock-up fuel assemblies(actual dimensional size) was performed. Based on the vibration test results,the neutron flux transient during fuel assemblies vibration was analyzed and the main cause of the reactor trip during the earthquake was clarified.

2. Description of the transient

On November 27,1993, with Onagawa nuclear power plant Unit-l at 100 percent power, the earthquake struck at 3:10 p.m. and the reactor tripped automatically with average neutron flux high signal. The distance between the seismic center and the plant site was approximately 1 15km (Fig.l) and the earthquake magnitude was 5.9. The maximum accelerations measured on the basemat are presented in the following table.

DIRECTION	EAST& WEST	NORTH & SOUTH	UP & DOWN
ACCELERATION (gal)	121	37	50

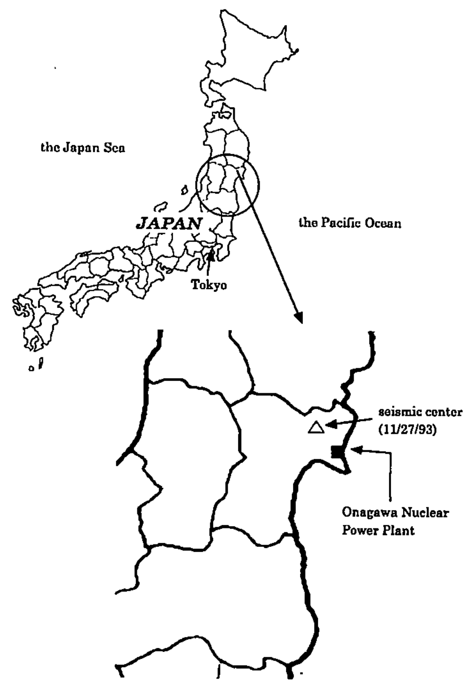

Figure 1. Location Map

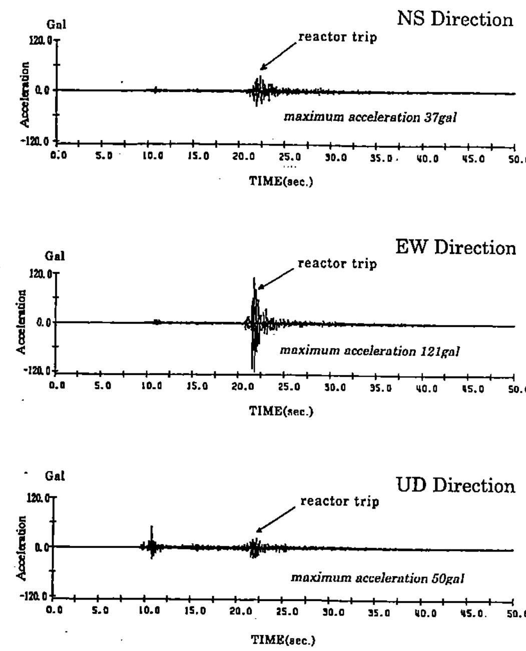

Figure 2. Acceleration Histories on Onagawa Unit-1 Basemant

An acceleration of 121 gal in an EW direction was the largest values which we had ever observed. The acceleration histories ,in which the reactor tripped after the peak had appeared in UD direction, are shown in Fig.2. And Fig.3 gives the main plant parameter trends from PLADIS(Plant Diagnostic Instruction System) which deals with a large amount of plant data very fast(minimurn sampling time:l0msec). PLADIS showed us that the main plant parameters had changed little just before the earthquake. During the earthquake, the neutron flux increased resulting in a reactor trip, a rapid negative reactivity insertion due to control rod scram. A remarkable point is all LPRM(Local Power Range Monitor) and APRM(Average Power Range Monitor) signals increased during the earthquake. This suggests that a positive reactivity was actually inserted to the core. (Before Onagawa Unit-1 tripped, Fukushima-1 nuclear power plant Unit- 1,3 and 5(BWR,Tokyo Electric Power Co.,Inc) also had tripped with average neutron flux high signal on April 23th,1987. But they didn't have PLADIS at that time.)

3. Analysis of the cause

Fig.4 gives probable causes of neutron flux high signal. Some causes are rejected in terms of parameter trends obtained from PLADIS (e.g.,surge current in Reactor Protection System). And other impracticable causes aren't considered either (e.g., resonance of shock waves; shock waves hardly spread in two-phase flow like coolant of BWRs). The following are assessments of causes that could add reactivity to the core.

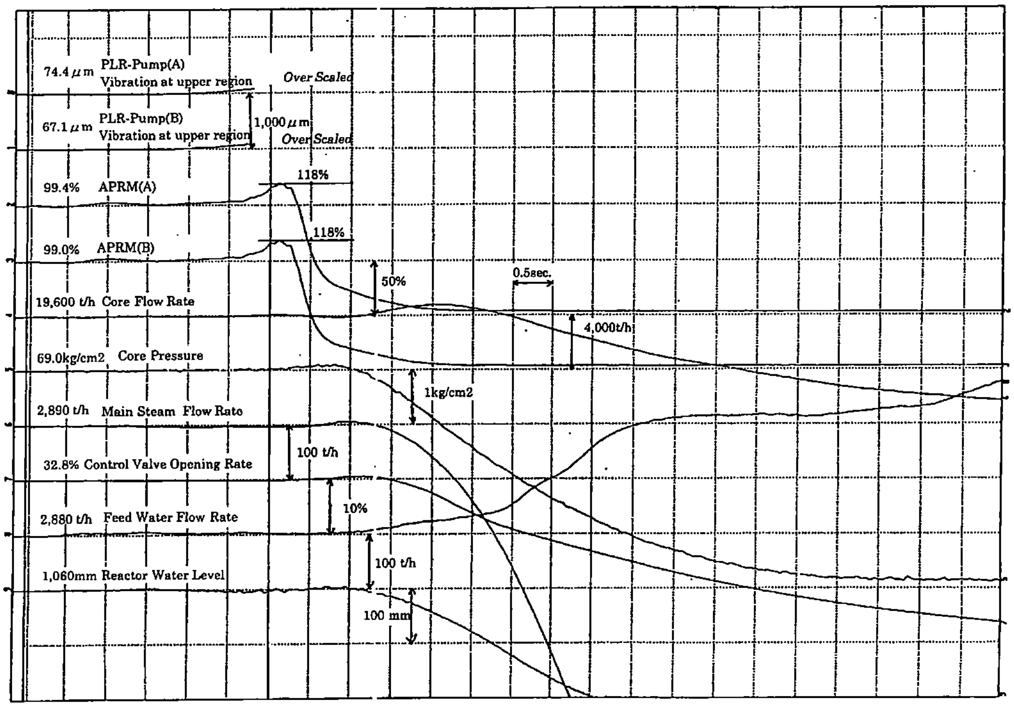

Figure 3-1. Main Plant Parameters Trends from PLADIS

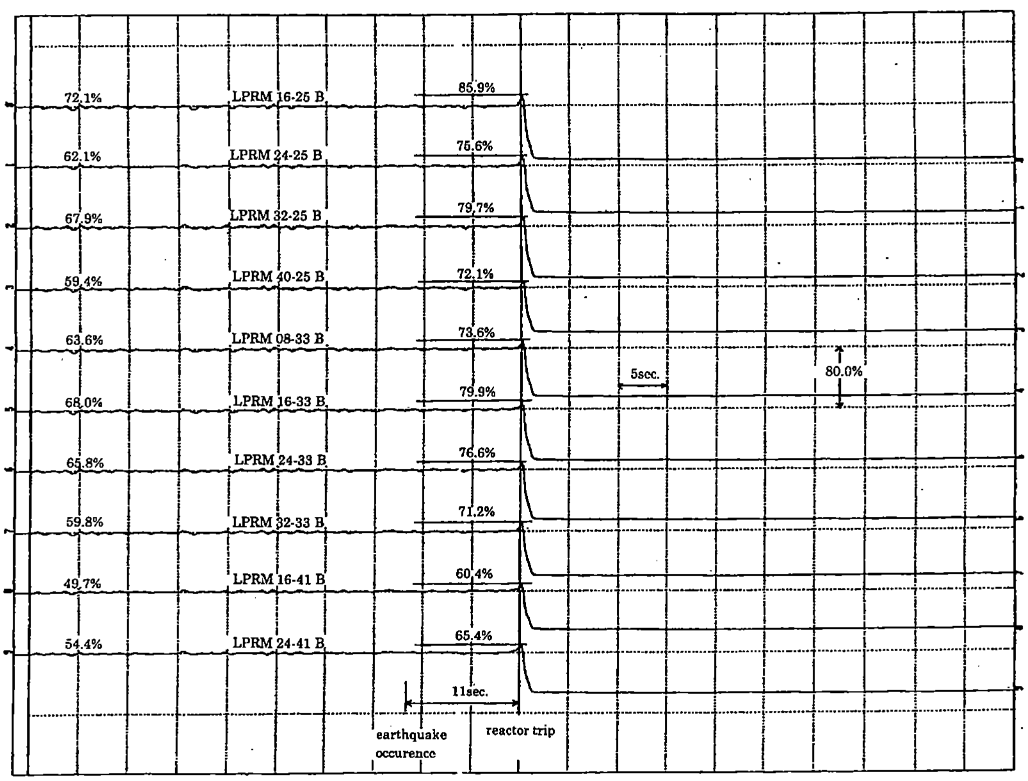

Figure 3-2. Main Plant Parameters Trends from PLADIS

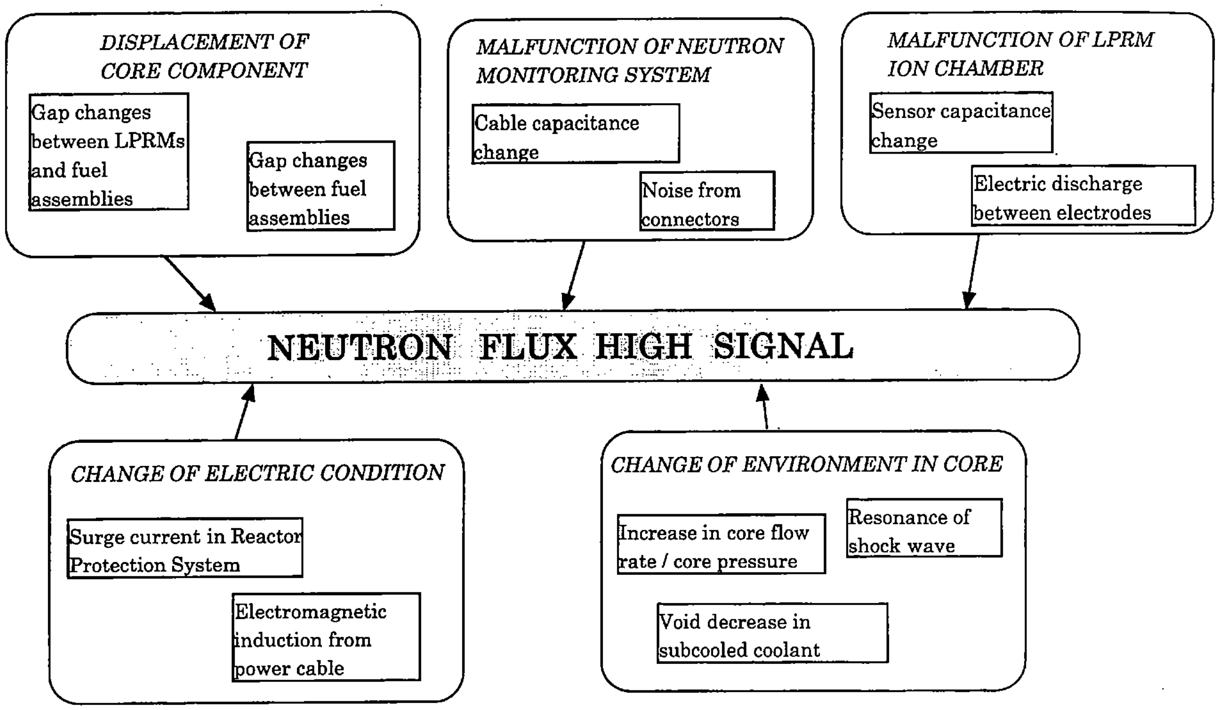

Figure 4. Causes of Neutron Flux High Signal

3.1. Increase in core flow rate or reactor pressure

The fission reaction is advanced when the core flow rate or the reactor pressure increases due to void reduction in the core as to BWRs. But as Fig.3 shows, the two parameters had made stable transition just before the trip of the reactor. The seismic acceleration didn't have influence upon these parameters.

3.2. Void decrease in subcooled coolant

For BWRs, the recirculation pumps are used to feed coolant into the core from the bottom, creating the void in the subcooled coolant at the lower region of the core. In the subcooled coolant, the water temperature does not reach saturation. It is possible for the void in subcooled coolant (hereafter referred to as subcool-void) to depart from the surface of the fuel claddings due to the acceleration, and disappear in subcooled water. The disappearance of the void may cause the positive reactivity insertion particularly at the lower part of the fuel.

When Onagawa Unit- I tripped, we could not seize the axial neutron flux behavior, because PLADIS was connected with LPRMs of level B (four vertical ion chambers form an LPRM string, the lowest being level A and the highest being level D). After the incident, level A,C, and D were added in the PLADIS with two LPRM strings. On August 14, 1994, the neutron flux increased again, but the reactor did not trip, during the earthquake. The obtained LPRM's data (See Fig.5) showed that the higher the LPRMs were, the bigger the increase rate. The mechanism of subcool-void decrease cannot explain the phenomenon, because the subcool-void exists only at the lower region of the core.

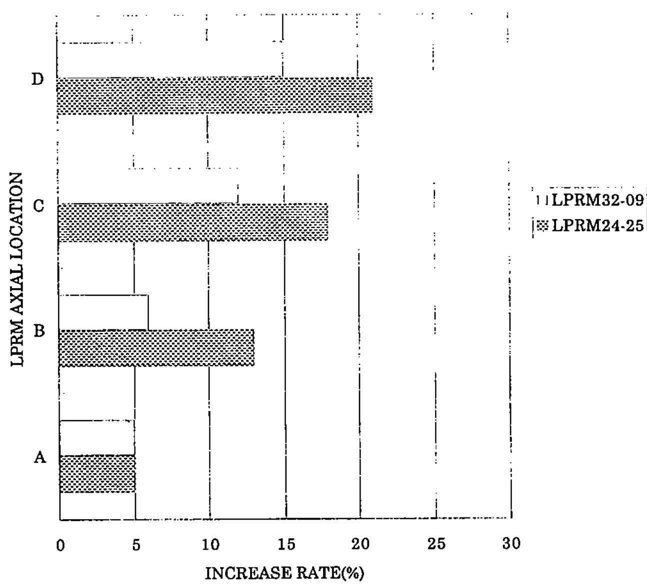

Figure 5. LPRM Axial Indication Increase Rate

However the subcool-void behavior under the state of vibration had not been verified correctly. Therefore we carried out a basic experiment as described in section 4. 1.

3.3. Gap changes between the fuel assemblies[1]

Fig.6 gives four fuel assemblies loaded in the core from above. As four assemblies can move only in the upper fuel support grids, the earthquake acceleration causes the gap changes between the assemblies. A behavior of assemblies under the state of vibration is depicted in Fig.7.

Onagawa Unit-1 has the core called D-lattice which has a feature of different water gap widths. The larger gap is on the cruciform control rod insertion side (See Fig.8). Therefore enrichment of fuel rods in fuel assembly is split in order to flatten horizontal power distribution. Enrichment of fuel rods adjacent to the wide water gap is designed to be lower and enrichment of fuel rods adjacent to the narrow gap is higher. Fig.8 also gives K-inf curves according to H/U as for each lattice. H/U is equivalent to displacement of assembly in this case. The fuel assemblies for D-lattice have different curves in region adjacent to the wide water gap and opposite side. This is a reason that the increasing rate of reactivity for widening narrow water gap (water gap changing is equal to H/U changing at local view point) is larger than the decreasing rate of reactivity for narrowing fuel wide water gap. Consequently the average neutron flux increases when the gap between fuel assemblies changes due to acceleration of earthquakes. On the other hand, the fuel assemblies for C-lattice which has identical water gap widths, do not have the enrichment split. The K-inf is therefore represented by one curve. The figure shows that the average neutron flux does not change or decreases even slightly when the fuel gaps change.

As mentioned in section 2., before Onagawa Unit-1 tripped, Fukushima- 1 nuclear power plant have had similar experiences several times. Fukusima-1 has 6 reactors, the Unit-1 to 5 have D-lattice core and the Unit-6 has C-lattice core. The average neutron flux behavior measured by the APRM recor-

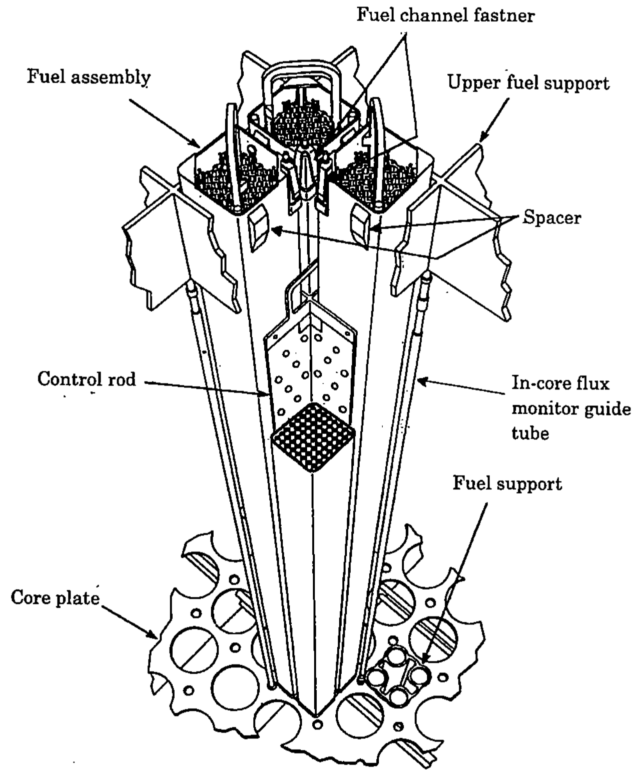

Figure 6. Fuel Assemblies Loaded in the Core

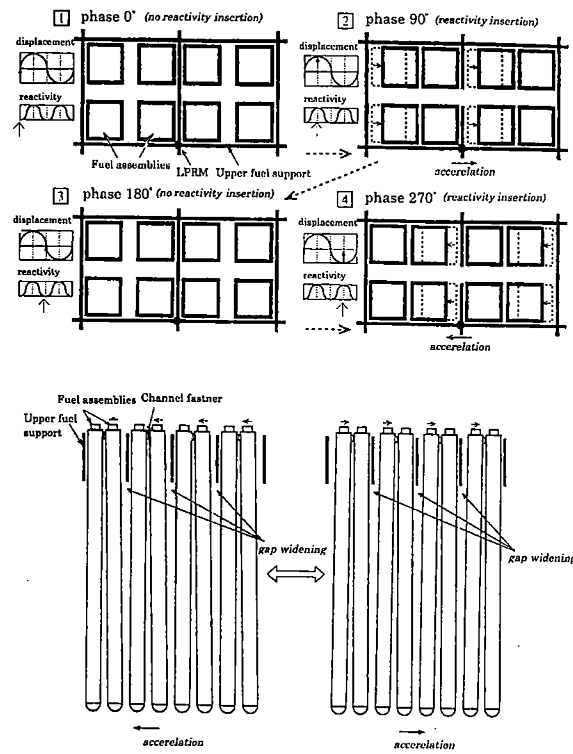

Figure 7. Behavior of Fuel Assemblies Under the State of Vibration

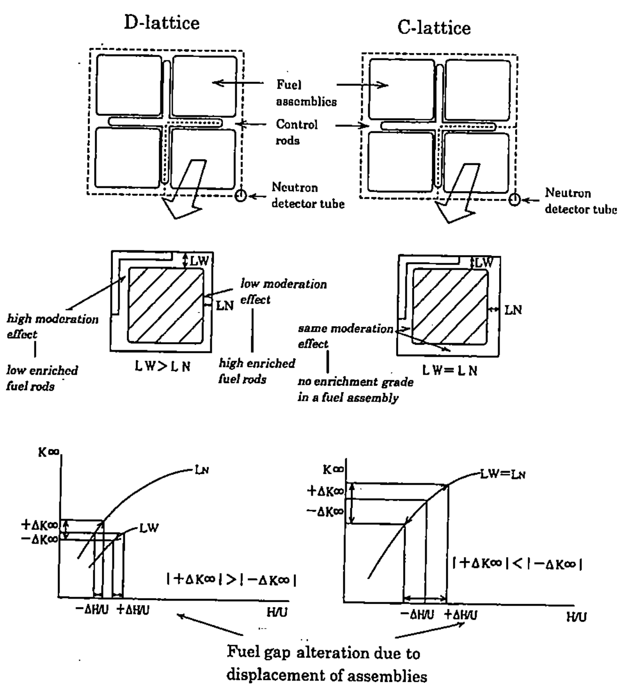

Figure 8. Characteristics Comparison D-lattice with C-lattice

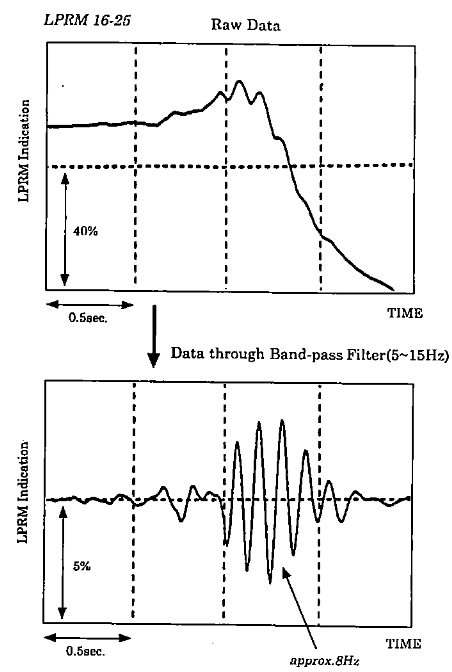

Figure 9. LPRM Signal through Band-pass Filter

ders for several years in each unit was surveyed and it was clarified that only the Unit-6 having C-lattice has never experienced the increase of neutron flux while in other D-lattice units, the neutron flux has increased at least twice. This is a strong circumstantial evidence for the mechanism of the gap changes between the bundles.

There are two other facts supporting the mechanism.

(1) Fig.7 shows the water gap widens and a larger amount of reactivity is inserted at the upper part of the assemblies than the lower part, when the gap between fuel assemblies changes. It explains the data obtained on August 14, 1994 in Onagawa Unit- 1 mentioned in section3.4. very well.

(2) LPRM data through a band-pass filter(from S to 1 SHz) are shown in Fig.9. It is known that a natural frequency of fuel assemblies is approximately 4Hz, and the figure gives the LPRM signals contain harmonic frequency of 8Hz.

This mechanism is more descriptive in actual performance characteristics than in others. We carried out some vibration experiments and analyses to evaluate this mechanism more quantitatively.

4. Vibration experiments

4.1. Void behavior in subcooled coolant[2]

We performed basic vibration tests to observe the subcool-void behavior with equipment consisting of a vessel, a vibrator and measurement systems, as illustrated by Fig.10. In the vessel, a heater and a cooler were equipped. Adjusting power ofthe heater and flow rate of the cooler generated subcool-void under the state of vibration. We recorded behavior of subcool-void by a high speed video camera.The experiments were carried out under condition of atmospheric pressure and natural circulation.

With time, changes in diameters of voids were observed by the camera. Parameters such as subcooling, vibration frequency and acceleration varied. The obtained data were applied to Bankoffequation given by,

$$D/Dm=1-2^{K}(abs(0.5-(T/Te)^{N}))^{K}$$

D = void diameter
Dm = amaximum void diameter
T= time
Te= life span of a void
K, N= constant

One of the results is plotted on the Fig.11. This figure shows that vibration does not have influence on the mechanism of void generati on and disappearance. The same thing can be said about other test cases. The temperature boundary-layer is thinner in the actual reactor environment such as high heat flux and forced coolant circulation than conditions in which the experiments were performed. In other words, subcool-void diameters tend to be small in the reactor in operating, therefore the influence of the vibration on the subccol-void behavior is negligible.

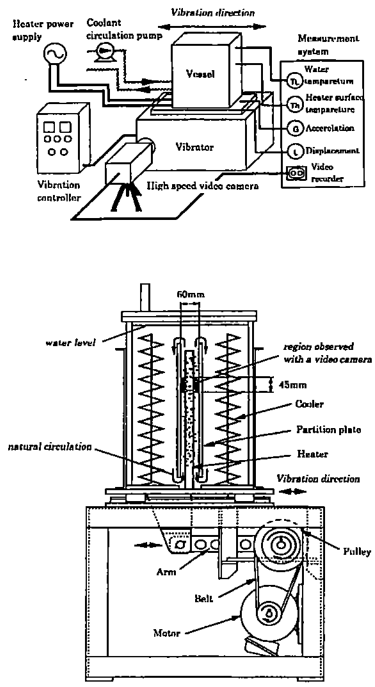

Figure 10. Experimental Apparatus (Void behavior under vibration)

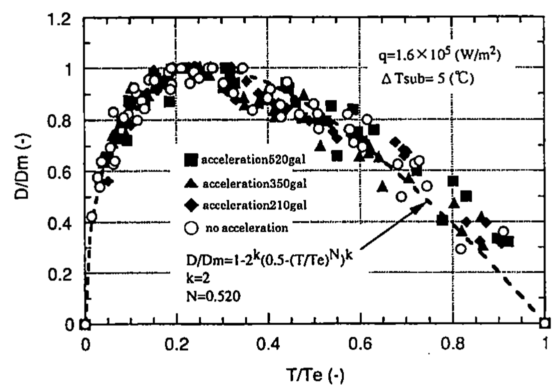

Figure 11. Experimental Results (Void behavior under vibration)

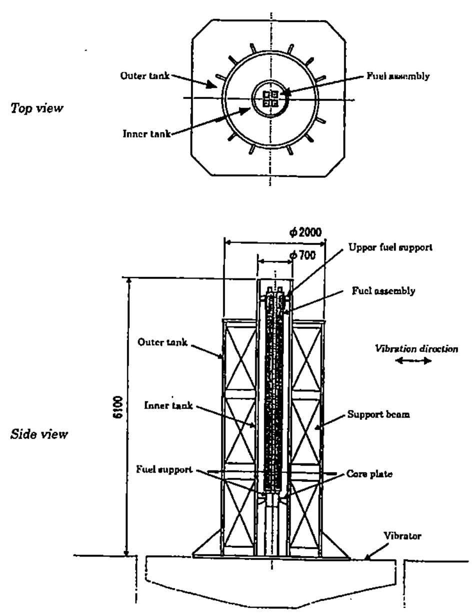

Figure 12. Experimental Equipment (Fuel gap change

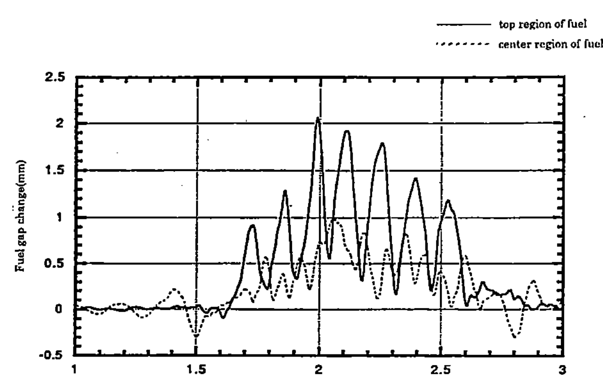

Figure 13. Fuel Gap Change obtained from Vibration Experiment

4.2. Gap changes between the fuel assemblies

To investigate the interval changes of fuel assemblies during the earthquake, vibration experiments were carried out.

Four mock up fuel assemblies were loaded in an inner tank filled with water under the atmospheric pressure condition and a tank was located on the vibration test facility as shown in Fig.12. The inner tank components simulated D-lattice core.Horizontal accelerations and fuel gaps were measured at seven axial positions of fuel assemblies with strain gages and eddy-current coils. The vibration tests were carried out based on the recorded acceleration time histories.

The experiments showed nonlinear behavior of the fuel assemblies motion such as detaching of the fuel assemblies from the upper fuel support caused by deformation of channel fasteners (See Fig.13). It was verified that the nonlinear behavior caused fuel gap changes. And the fuel gap changes at the upper side of the fuel assemblies were larger than that at center region. Maximum values of fuel gap change at the upper side and the center region were approximately 2mm and lmm respectively.

5. Evaluation of the mechanism by three-dimensional transient code: ARIES

Based on the vibration test results mentioned above, the transient in the reactor core during the earthquake was analyzed by a three-dimensional numerical code, ARIES[(3)]. The ARIES code is a three-dimensional dynamic code, which has a threedimensional neutron diffusion model and a multi channel thermal-hydraulic model. It takes the coupling effect of neutronics and thermal-hydraulics into account. Numerical simulation results were compared with the measured data during the earthquake as shown in Fig.14. A good agreement was obtained as for the transient of average neutron flux signal. The analyzed average neutron flux signal increased close to 120% rated power during the simulated earthquake,which could initiate the automatical reactor trip with average neutron flux high signal.

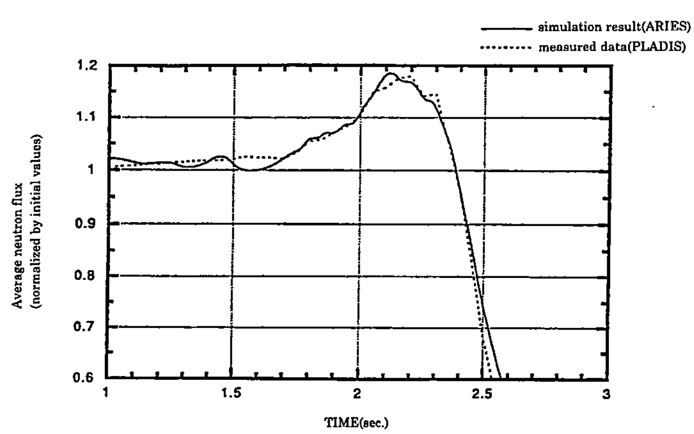

Figure 14. Numerical Simulation Result

6. Conclusion

Some experiments and analyses clarified that the fuel gap changes cause the reactor trip experienced at Onagawa Unit-1 in November '93. The data obtained from PLADIS gave important information to make clear the cause of the reactor trip. And according to preliminary evaluations, the reactivity insertion and its effects by that mechanism are smaller than main transient events evaluated in Establishment Permission. Therefore there is no safety-significant concern.

This study is a part of the results from co-operated work with Tokyo Electric Power Co., Inc.,Chubu Electric Power Co., Inc., Hokuriku Electric Power Co., Inc., Chugoku Electric Power Co., Inc., The Japan Atomic Power Co.,Inc., Tohoku Electric Power Co., Inc.,Toshiba Corporation and Hitachi Ltd..

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The fuel failure in Hamaoka nuclear power station unit 1

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Abstract

The 11th annual inspection of Hamaoka Unit 1, one of the nuclear plants of Chubu Electric Power Company, Inc. (BWR; rated power of 540 MW), was conducted from June,1990 to June,1991. Since the concentration of radioactivity in the reactor coolant had increased slightly during the cycle 11, sipping inspections (inspections to identify leaking fuel bundles) were conducted during this annual inspection period. Five bundles were identified to be leaking.

In addition, as a result of visual inspections on the five leaking bundles, it was revealed that the crud and oxide layer on the cladding surface of some fuel bundles were partly spalled. Thus, the following investigations were carried out to make the cause of the leakage;

- On site investigation such as oxide thickness measurement, crud analysis

- Investigation of reactor water quality and fuel cladding material etc.

Furthermore, the post irradiation examination (PIE) was conducted at the hot laboratory in Japan in order to explore the situation of the leaking bundle in detail and to clarify the oxidation mechanism. This report summarizes the results of the series of investigations, cause, countermeasure, the results of PIE and so on.

1. Outline of Plant and Fuel Bundle

1.1. Outline of plant

Name:	Hamaoka Nuclear Power Station Unit 1
Reactor Type:	Boiling Water Reactor
Primary Containment Vessel:	MARK-I
Thermal Power:	1593MWt
Electric Power:	540MWe
Commercial Operation:	March 17, 1976

1.2. Outline of Fuel Bundle

Fuel bundle loaded in the cycle 11 core was New 8x8 fuel bundle(8x8RJ) and 8x8 zirconium liner type fuel bundle (8x8BJ).

2. Outline of the Event

2.1. Change of I-131 Concentration in the Reactor Coolant Water during Operation

After the 10th annual inspection, Hamaoka Unit 1 started the cycle 11 on August, 1989.

On March 27, 1990 the indication of the outlet gas monitor of the off gas condenser increased from a normal value of about 1.9×10^{-11} amperes (A) to approximately 2.5×10^{-11} amperes at rated power. Also the concentration of I-131 in reactor coolant water increased from a normal value of about 0.15Bq/g to about 1.5Bq/g on March 30,1990. However, since there was sufficient margin to the operational limit for the I-131 concentration in reactor coolant water (7700Bq/g), reactor operation was continued with more careful monitoring. The I-131 concentration in the reactor coolant water reached approximately 20Bq/g at the end of the cycle 11. The chart of off-gas and I-131 concentration was shown in Fig.1.

2.2. The State of Leaking Fuel Bundles In-core and out-core sipping

inspections were conducted on all of the 368 fuel bundles. 5 fuel bundles of all were identified to be leaking and were confirmed to be spalling by visual inspection. These were new 8x8 fuel bundle loaded in the cycle 9 and were irradiated for three cycles about 24~25 GWd/t.

3. On-site examination for Fuel Failure

3.1. Visual Inspection of Fuel Bundles (Using Underwater TV camera and Fiberscope)

Since spalling was observed by visual inspection on five leaking fuel bundles, the same inspections were carried out on other fuel bundles loaded in the cycle 11. As a result, it was found that 78 fuel bundles loaded in either the cycle 8 or 9 and therefore irradiated for three or four cycles was spalling. In most of the spalling areas, as crud and oxide layer were peeled off from cladding (spalling area), oxide layer appeared white in color at the spalling surface. Spalling size and shape was various and surface of some rods looked like pit (Fig.2).

There was no spalling on the fuel bundles loaded in either the cycle 10 and 11.

3.2. Ultrasonic Inspection

The 5 fuel rods included in 4 bundles of 5 leaking fuel bundles were detected as a failed rod by ultrasonic inspection on site. These fuel rods were observed spalling.

3.3. Oxide Thickness Measurement

The fuel rods of bundles loaded in the cycle 8 or 9 looked like white in color and uniformly oxidized. And the maximum oxide thickness was more two times thicker than that of sound fuel rods measured

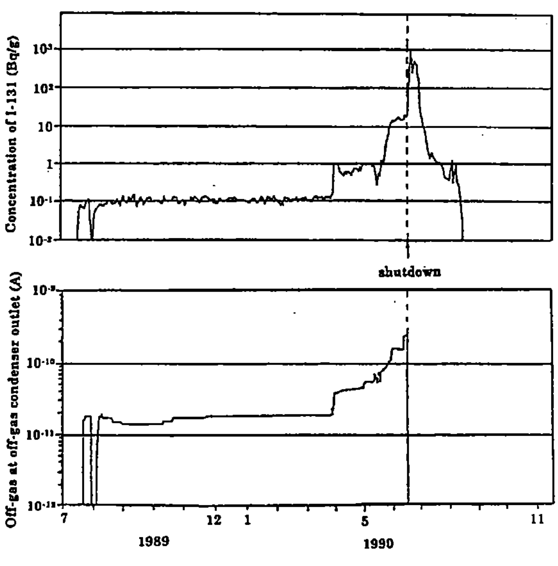

Figure 1. The Cart of I-131 Concentración and Off-gas During the 11 Cycle

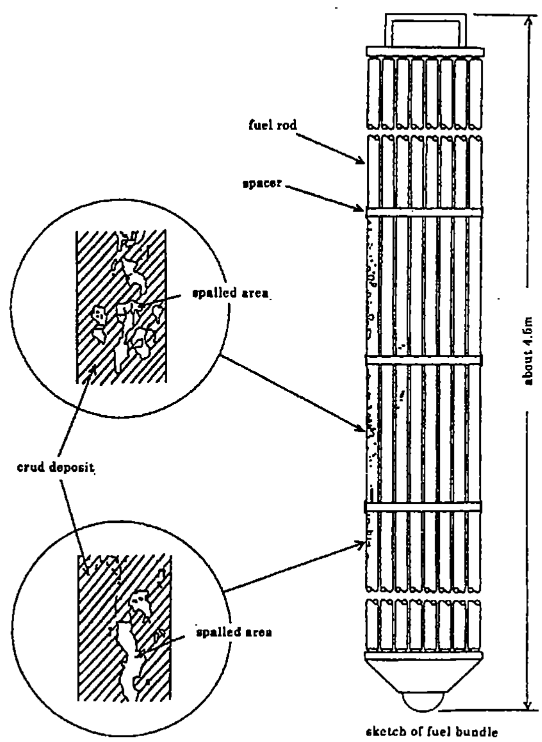

Figure 2. Typical Appearance of Leaking Fuel Bundle

before (l). The rods of fuel bundles loaded in the cycle 10 or 11 did not looked like white in color and were observed nodular corrosion.

The oxide thickness of these fuel rods was as thick as that of sound fuel rods(Fig.3).

3.4. Crud Analysis

The crud was sampled and analyzed. According to this analysis, the crud was composed mainly of iron, as was crud on sound fuel rods.

4. Investigation of Cause of Abnormal Oxidation[2]

Based on the above investigation a formation of abnormal oxide was observed. It was explained that the formation of thick oxide abnormally occurs under the condition of a combination of the peculiar coolant water chemistry and the relatively high corrosion susceptible cladding ,as reported in foreign event.

4.1. Examination of Reactor Coolant Water Chemistry[3]

The reactor coolant water quality was satisfied with the specified control requirements, but examined in detail to identify the peculiar characteristics that led to corrosion.

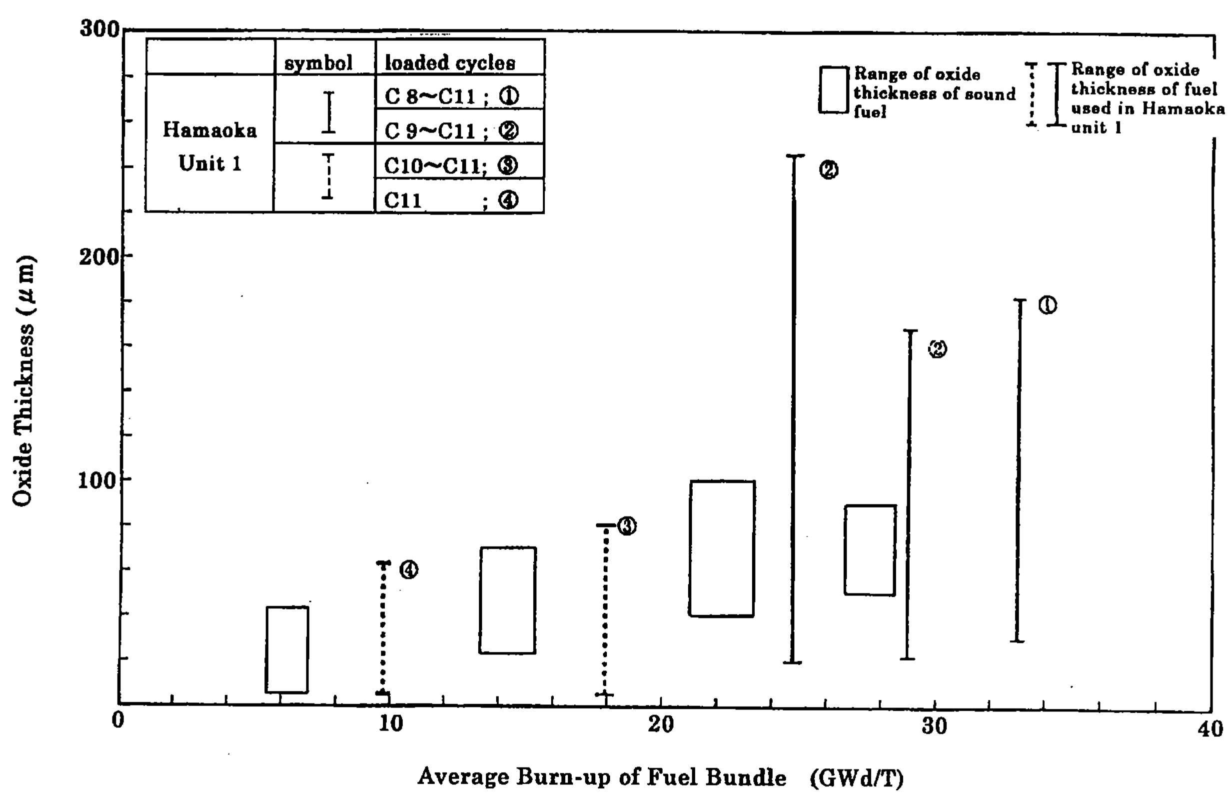

Figure 3. Measured Oxide Thickness of Fuel Bundle

• At the Time of Reactor startup

The electrical conductivity of reactor coolant water increased to 1-1.2 μS/cm and sulfate ions (SO_4^{2-}) were also detected at the startup of the cycle 9 and 11. No increase had been observed before the cycle 9. It was thought that this increase of the electrical conductivity come from the presence of chemical species which were generated by decomposition of organic materials by heat and radiation in the reactor. The Organic materials, which were usually monitored as Total Organic Carbon (TOC), dissolved from ion exchange resin in the condensate demineralizer (CD) and flowed into the reactor (Fig. 4, 5).

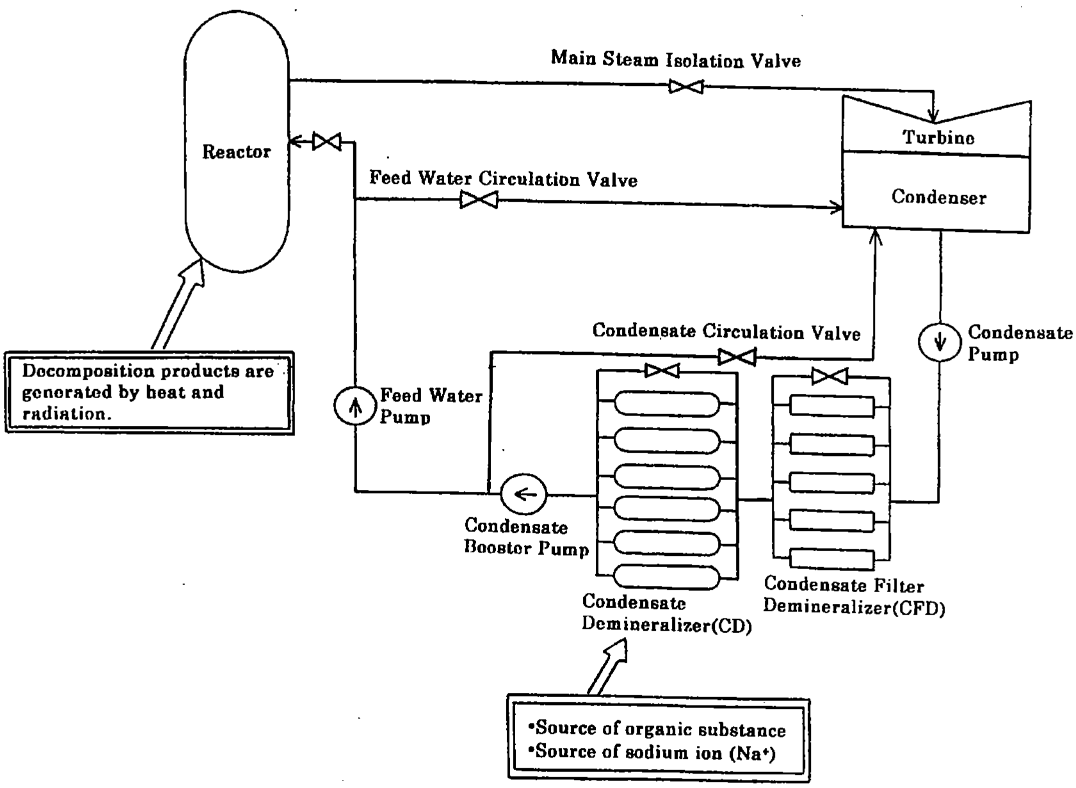

Figure 4. Feed Water/Condensate System Diagram

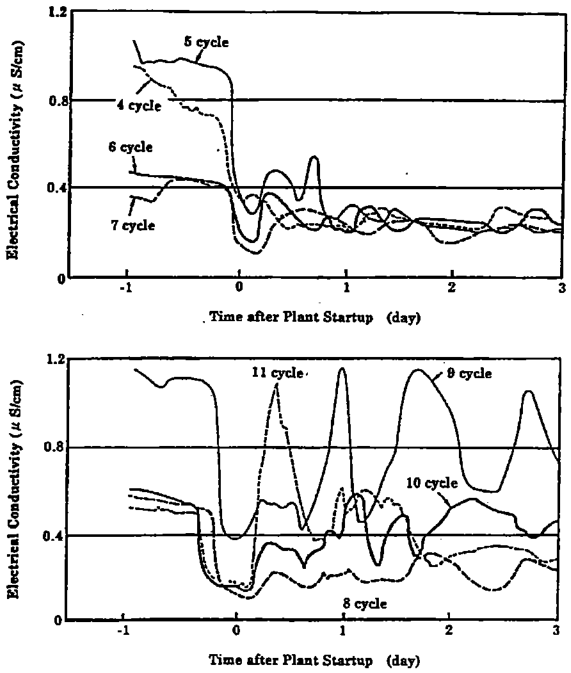

Figure 5. Change of the Electrical Conductivity of Reactor Coolant Water at Plant Startup

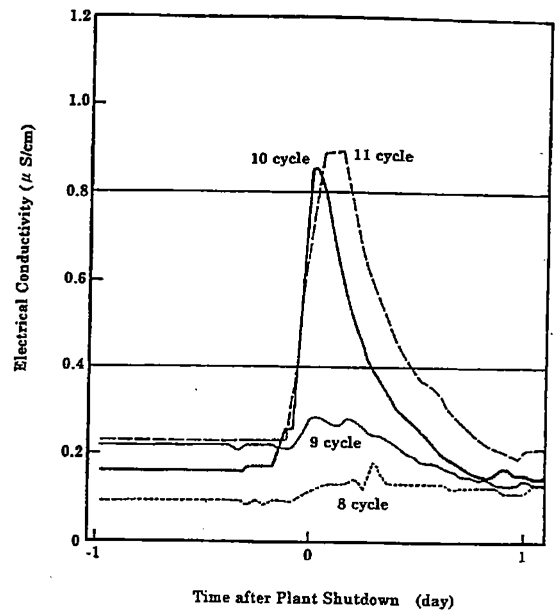

Figure 6. Change of the Electrical Conductivity of Reactor Coolant Water at Plant Shutdown

- During Rated Power Operation

The average concentration of sodium ions (Na^+) in reactor coolant water during rated power from the cycle 8 to the cycle 11 was 15ppb and the average pH value was about 7.7. These values were slightly higher than those in other domestic BWRs. The higher value could be explained by the following process. A part of cation exchange resin resided in the anion exchange resin and was regenerated in Na form by caustic soda(NaOH) (inverse regeneration) when resin was separated from the CD for chemical regeneration. As a result, a small amount of Na^+ immigrated from the CD into the reactor and caused the high Na ion concentration in the reactor coolant water.

Organic materials flowing out of the CD resin also entered through the feed water system into the reactor and generated a small amount of decomposition products(SO_4^{2-},CO_3^{2-},etc) by heat and radiation (Fig.4).

- At the Time of Reactor Shutdown

The electrical conductivity of reactor coolant water at shutdown of the cycle 10 and 11 was higher than that of previous cycles and Na^+ and SO_4^{2-} were detected at the shutdown of the cycle 10 (Fig.6) It was thought that these ions dissolved from Na_2SO_4 which had adhered to the oxide layer on the cladding (hideout return) with decreasing plant power.

4.2. *Examination of Cladding Tube Material*

Though the specification of cladding tube material was satisfied, manufacturing process of cladding tube was surveyed in detail to find the peculiarity in regards to the corrosion susceptibility.

We know that the corrosion susceptibility of BWR decreases with decreasing accumulated normalized annealing time after the ß-quenching process (annealing parameter: Σ Ai) used as an index for evaluating the influence of heat treatment on corrosion. The annealing parameter of all of the cladding tube that had been loaded in the cycle 11 was evaluated. Annealing parameter of the cladding loaded in either the cycle 8 or 9 were found to be larger than that of others (Fig.7).

A high temperature corrosion test was performed on the archive tube materials of the fuel cladding. Corrosion weight gain of the cladding tube loaded in the cycle 8 or 9 was relatively large and widely distributed.

5. Cause of Failure

It was supposed that decomposition products (SO_4^{2-}, CO_3^{2-}) of CD resin resulted in the increase of the electrical conductivity at plant startup could influence initial oxidation of the fuel claddings. And it was presumed that during rated power Na^+ and decomposition products could be condensed in oxide layer by boiling into Na_2SO_4 and similar compounds. These could have an influence on the acceleration of the cladding oxidation. It was thought that abnormal oxidation was attributed to the presence of such species in reactor coolant water.

It could be determined that some materials used in cladding loaded in the cycle 8 or 9 had relatively high susceptibility of corrosion due to the examination of cladding tube material.

Thus, it was thought that this fuel failure of Hamaoka Unit 1 could be caused by a combination of the peculiarity of water chemistry and relatively high corrosion susceptibility of cladding.

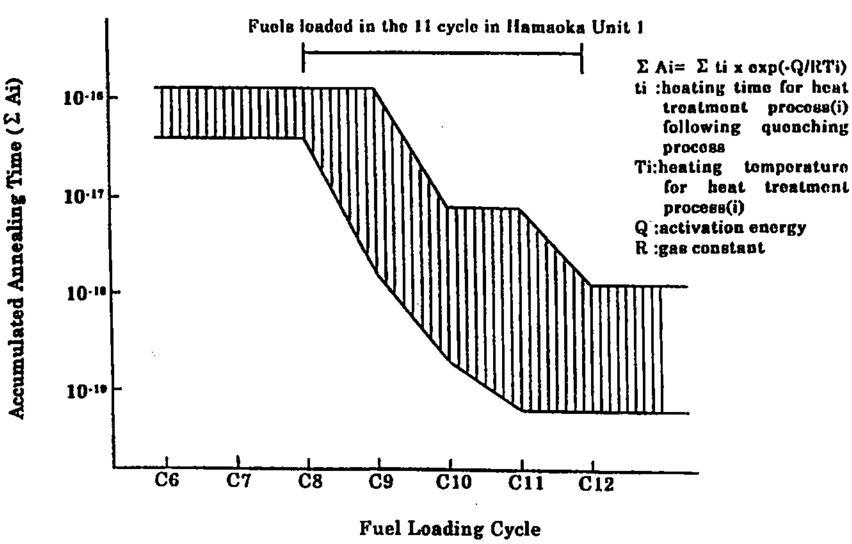

Figure 7. Change in Σ Ai for Fuel Cladding Material

6. Countermeasures

6.1. Improvement and Control of ReactorCoolant Chemistry

1. Prevention of Sodium Ion Inflow

Used resins were replaced with new resins of CD. And regeneration of CD resin was not operated from the cycle 12 to prevent the cation resin from inversely regenerating to Na^+ form at the chemical regeneration of CD resin.

2. Suppression of Electrical Conductivity Increase at Startup

To suppress the increase of the electrical conductivity due to organic compounds discharged from CD, discharge of organic materials from CD before starting of CD operation, bypassing reactor coolant water from CD in cleanup operation before plant startup and improvement of cleanup capacity of reactor coolant cleanup system (CUW) were carried out.

3. Enhancement of Control of water chemistry

Levels of organic materials (TOC) before startup, of the electrical conductivity of reactor coolant water at startup and of Na^+ during operation were controlled strongly.

6.2. Fuel Bundle

The cycle 12 core was composed of the sound fuel bundles that were confirmed by sipping, visual inspection and oxide thickness measurement on site. The improved corrosion susceptibility cladding tube was used in the fuel bundles loaded after the cycle 11.

7. The Plant Status after Countermeasures

The oxide thickness of 8x8 zirconium liner cladding that was improved in corrosion susceptibility was examined for several cycles and was found to be not more than about 40 μm. The fuel bundles has been sound and reactor coolant water chemistry has also sustained to be good, as shown in Table 3, under the countermeasures.

Measurement items		11 Cycle	12 Cycle	13Cycle	14 Cycle
Electrical Conductivity (μ S/cm)(25℃)		0.20	0.11	0.08	0.12
pH	(25℃)	7.7	6.9	7.0	6.5
Na	(ppb)	17	<1	<1	<1
Cl	(ppb)	2	2	<1	<1
NO_3	(ppb)	6	5	1	3
SO_4^{2-}	(ppb)	1	1	<1	1
SiO_2	(ppb)	390	230	430	307

notes : An increase of the electrical conductivity was not found at the plant start-up and shutdown of 12, 13, 14cycle.

Table 1. A Change of Reactor Water Quality in Hamaoka Unit 1

8. Post Irradiation Examination

One of the failed bundles was transported from Hamaoka Nuclear Power Station to the Nippon Nuclear Fuel Development Co., Ltd. (NFD) for the detailed PIE. [4] The bundle was irradiated for three cycles and this exposure was around 25GWd/t.

This bundle is shown in Fig.8. and failed rod was located in A1 position.

8.1. Visual Inspection of the Fuel Bundle

The fuel bundle was observed through a window installed in the pool wall. The fuel rod surfaces at the areas where crud was removed is shown in Fig.9. The results is following;

- The fuel rod surfaces (spacer N.° 13) were covered with reddish brown crud and white oxide with partial spalling was on most fuel rod surfaces. The failed rod A1 was especially observed much spall.

- Gap distance between rod A1 and B1 changed due to bowing of rod Al. (spacer N.° 2~3: narrow, spacer N.° 3~4:wide)

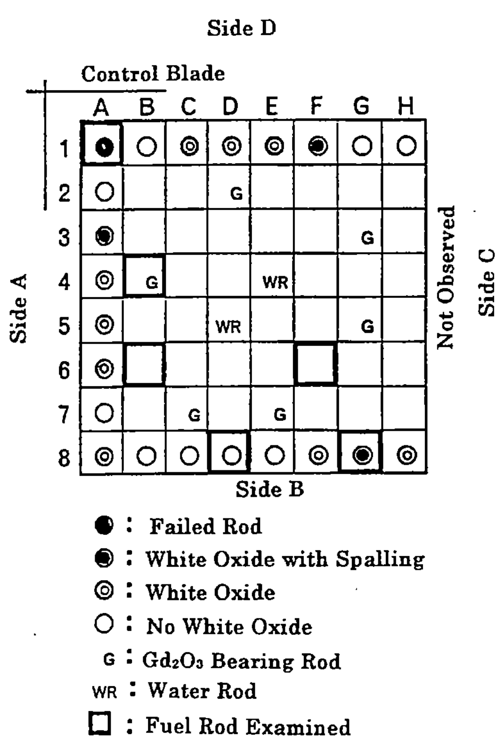

Figure 8. State of Rod Surface Oxide from BundleVisual Inspection

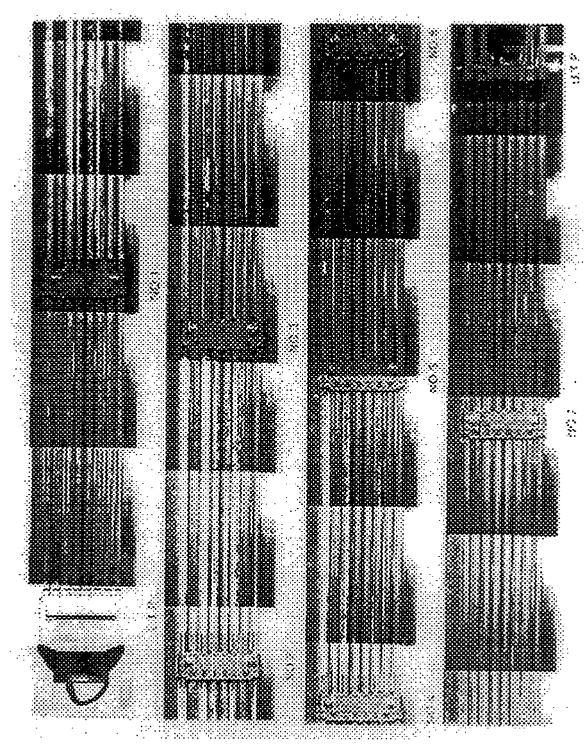

Figure 9. Appearance of the Fuel Bundle (Crud removed at Some Area)

8.2. *Non-Destructive Examination of the Fuel Rods*

Six fuel rods were selected for nondestructive examinations based on the bundle inspection results as listed in Table 2.

1. Visual Examination

   - White oxide and its spalling on the rod A1 and G8 were observed only on the surfaces facing the bundle outside. By contrast, those surfaces facing the bundle inside exhibited no white oxide.

   - Rod A1 also showed irregular rod bowing and bulging near its bottom, both of which are attributed to the secondary hydriding as mentioned later.

   - The other rods showed no white oxide on any surfaces.

2. An Eddy Current Flaw Inspection

   A significant signals were detected at the middle and bottom position of Rod A1 corresponding to the crater-like spalling and bulged portions. No defect signals were identified from the other rods.

3. Oxide Thickness Measurement

   Oxide thickness of cladding were measured by both an eddy current method and metallography as described later. Only the failed rod A1 was not examined by the eddy current method due to its bowing which caused measuring difficulty. Oxide thickness profiles are shown in Fig.10.

   The maximum oxide thickness for the rod G8 was almost 150 μm on the surface facing the bundle outside but around 50 μm on surface facing the bundle inside. The others showed maximum oxide thickness of around 50-90 ,μm. The oxide thickness tended to become thicker longitudinally at a location between spacer N.° 1 and 3 and circumferentially on surfaces towards the bundle outside. This tendency was notable for the rods located at the bundle outermost row.

4. Determination of the Defect Location

   The defect locations were identified by the bubbling method in which helium gas was injected into the rod which had been immersed in water to allow the released bubbles to be observed. Three dif-

Location in the bundle		NDE[1)]	DE[2)]	Remarks
Outermost row	A1	○	○	Failed rod
	G8	○	○	White oxide and its spalling
	D8	○	—	No white oxide
Inner row	B6	○	○	Maximum burnup
	F6	○	—	Minimum burnup
	B4	○	—	Gd_2O_3 bearing rod

1) Non-Destructive examination 2) Destructive examination

Table 2. Fuel Rods for non-Destructive and Destructive Examinations

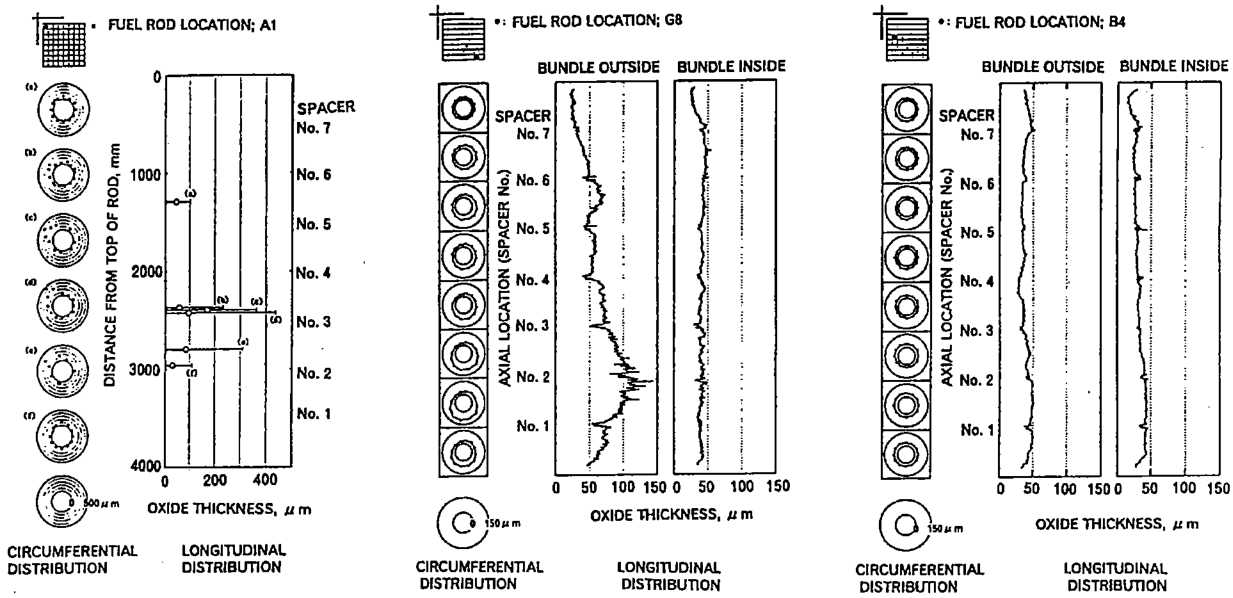

Figure 10. Cladding Oxide Thickness of Rods A1 (by metallography), G8 and B4 (by eddy current method)

ferent locations, two at the oxide spalling (primary defect), about 2350 mm from top of the rod, and one at the bulge (secondary defect), about 3850 mm from top of the rod, were identified as leaking.

8.3. Destructive Examination

Three rods (Al, G8, B6) were selected based on the non-destructive examination results as shown in Table 2.

1. Fission Gas Measurements

   Two sound rods (G8, B6) were subjected to fission gas measurements by puncture method. Fission gas release rates were about 20% and 17%, respectively. They were within the previous reported range.

2. Cladding Hydrogen Analysis

   The pellets and cladding was oxidized by water flowed in so that the localized massive hydride precipitates were formed in rod A1. A specimen from it showed a very large hydrogen content of about 2500ppm, while other specimensrangedbetween 130-200ppm. The sound rods G8 and B6 had hydrogen amounts between 70-90ppm.

3. Crud Analysis

   The soft crud was about 90% iron with small amounts of nickel, copper, zinc, and other impurities. Major elements detected in the hard crud, which was sampled with the cladding oxide by

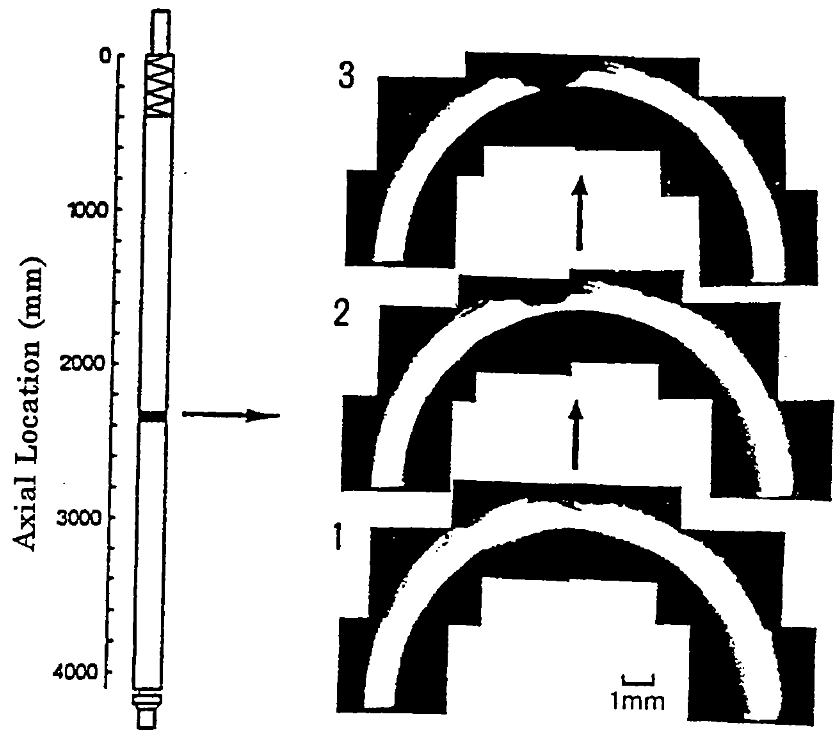

Figure 11. Cladding Cross Sectional Observations at Primary Defect

grinding the cladding surface after the removal of the soft crud, were iron, chromium, zinc, silicon, and nickel.

4. Metallography

The results was follows;

- Rod Al (taken from a vicinity of primary defect)
  - Cladding cross sections at one of the primary defects were observed at intervals across the defect, as shown in Fig.11. It was revealed that the primary defects were caused by a localized progression of the oxide within the thick white oxide layer, finally resulting in a breach of the cladding wall.
  - Many cracks were observed circumferentially in oxide layer.
  - Massive hydride precipitates (sunburst) was found at circumferential location opposite the areas of thick oxide.
  - The greatest density of hydrides was seen at the bulged location near the rod bottom with the secondary defect. In this location there were also large cracks which were passed through cladding wall.

• Rod G8

Oxide thickness tended to increase on the surface facing the bundle outside. Many cracks were observed in oxide layer circumferentially.

• Rod B6

No circumferential distribution of oxide thickness was observed.

• Oxide thickness of Rod A1 measured by metallography

Oxide thickness profile of rod A1 is shown in Fig.10. The maximum oxide thickness for rod A1 was more than 400 μm at the location which was a vicinity of defect location and 300 μm at the location observed white and thick oxide.

5. Cladding EPMA

Analysis Deposits on the cladding surface were analyzed by utilizing EPMA on the cladding cross section at the primary defect locations. Characteristic X-ray images are shown in Fig.12. No significant deposit of copper was observed but chromium or silicon deposited in the cladding oxide cracks.

6. Cladding Hardness Measurement

Micro Vickers hardness was measured on cladding tubes of rods A1 and B6 at various circumferential locations. Results are illustrated in Fig.13.

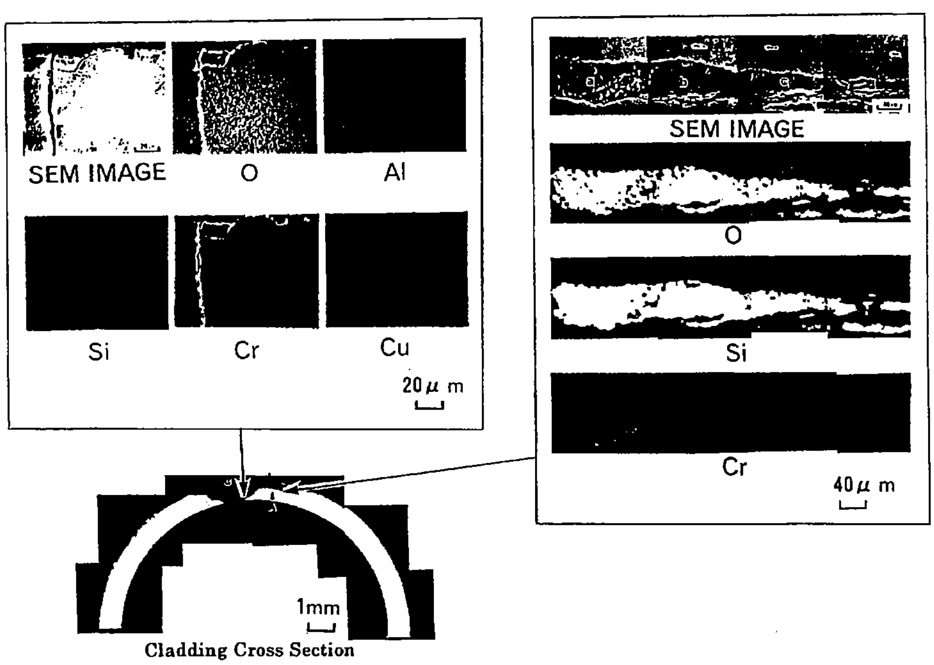

Figure 12. Characteristic Images by EPMA on the Cladding Cross Section at Primary Defect

- The cladding hardness of the rod A1 varied circumferentially and the lowest hardness at the thickest oxide location is equal to non-irradiated Zr-2 hardness. This meant that the recovery of irradiation hardening took place at the thick oxide portion due to the increase in cladding temperature during irradiation.

- The rod B6 showed the typical hardness of irradiation hardened Zr-2 and no circumferential variation.

8.4. Result of PIE

On the based of PIE we investigated the main cause of this affair. The status which could be clear by PIE was as following;

- The fuel rods located in the bundle outermost row showed characteristic features of cladding waterside corrosion in its appearance and its azimuthal variation of thickness. A thick and white oxide layer, with partial spalling, was abnormally formed on the surfaces facing the bundle outside.

- The failed rod was characterized by both the formation of the thick and white oxide layer and the progression of oxidation locally resulting in a breach of the cladding wall.

- No significant deposit of copper, as reported in CILC failures, was observed, but chromium or silicon deposited in the oxide cracks at the primary defects. The role of such elements in the oxidation behavior is not clear.

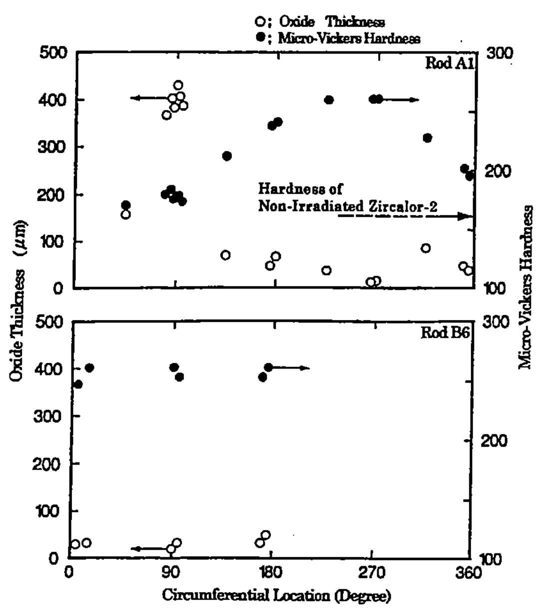

Figure 13. Circunferential Distribution of the Cladding Hardness on the Rods A1 and B6

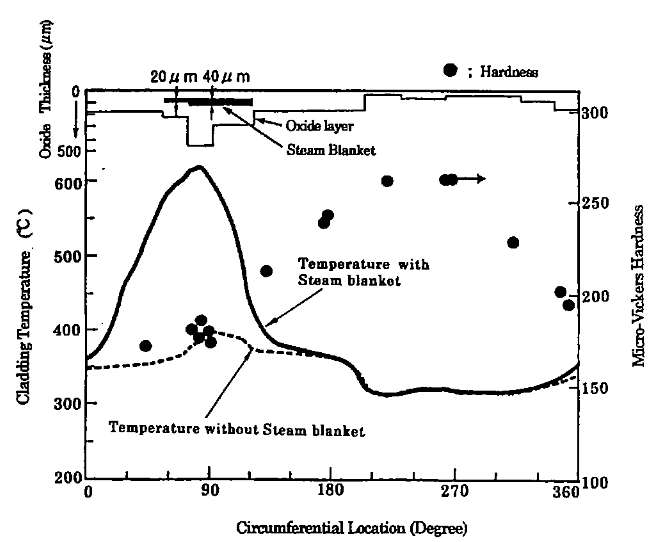

Figure 14. Results of Model Calculation of the Cladding Temperature Taking into Account the Oxide Effects, with and without the steam Blanket

We reconfirmed that the main cause was the abnormal and localized formation of a thick white oxide layer.

8.5. Discussion on the Formation of Abnormal Oxide Layer

With regard to the locally progressed oxide, cladding temperature during rated power operation was discussed. The cladding hardness measurements revealed a marked recovery of irradiation hardening beneath the thick oxide layer. This indicated that the cladding experienced abnormally high temperature causing annealing of irradiation defects.

The oxide layer involved many cracks. When these gaps were isolated from coolant water to form a steam blanket, the cladding temperature could be increased by its thermal insulation effect under the heat flux. Fig. 14 shows a circumferential cladding temperature profiles was calculated with the measured oxide thickness. Unless the steam blanket effect was considered, the maximum cladding temperature was around 400 °C. However, assuming a steam blanket thickness of 40 μm in the thick oxide layer, the maximum temperature increased to around 600° C, which was consistent with the observed hardness recovery. At such high temperature the zircaloy corrosion rate was enough high to penetrate the cladding wall within the order of several days. Although this temperature analysis involved some uncertainties such as heat transfer at the boiling surface, it qualitatively pointed out the possibility of significant cladding temperature rise, leading to a localized accelerated oxidation, based on the thermal barrier effect induced by steam blanket.

9. Conclusion

According to the investigation during the 11th periodical inspection, it was presumed that the peculiar water quality triggered the abnormal oxidation of the fuel cladding material with relatively high corrosion susceptibility, which led to the spalling and eventual leakage of some fuel rods in Hamaoka Unit 1. The results of PIE conducted after the investigation also revealed that fuel failure was caused by the localized oxide progression in the thick oxide layer. The mechanism of progression could be explained that the oxidation of cladding excessively accelerated due to the raise of cladding temperature by the thermal insulation effect after the formation of the thick oxide layer.

Accordingly, it was thought that the cause of fuel failure could be the abnormal oxidation occurred under the condition of the combination of the water quality factor and the cladding material factor.

Before the cycle 12 start-up, the following countermeasures were taken. First, only fuel bundles confirmed soundness were loaded together with fresh fuel and the cladding improved in corrosion resistance was used. Secondary, the reactor water quality was improved and the monitoring of water quality enhanced. These countermeasures continue to be carried out until today. Thus, the reactor water quality sustains to be good and the fuel cladding maintains to be sound.

10. Reference

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SICOM
An equipment for very accurate dimensional and corrosion inspection of irradiated fuel assemblies

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J.R.Fernández (TECNATOM), J.Serra (IBERDROLA), J.Vallejo (D.T.N.)

1. Introduction

Nuclear fuel undergo evolution in its properties and characteristics as result of the operation in the nuclear reactors. Knowledge of this evolution is necessary to guaranty the behaviour and safety in the operation, as well as to optimize the economic performance of the fuel cycle. For this reason, fuel manufacturers develop fuel behaviour models readily validated.

On the other hand, when with the objective of improving the performance of fuel, some important modifications are made, it is necessary to prove the effect on the behaviour and evaluate the influence in each of the critical characteristics of the fuel assemblies. In this cases, before a complete refuelling with new designed fuel is inserted, it could be necessary, in accordance with the importance of the modification, to burn a reduced number of fuel assemblies of the new design. This kind of fuel assemblies are known as demonstration assemblies. During the demonstration process, some characterization of the behaviour of the fuel is made, usually at the end of each cycle. The information obtained would be used to asses the criteria for commercial operation of the new designed fuel.

An important part of this characterization is made using NDE methods, applied at the nuclear plant during the refuelling outages. The features to be checked in this circumstances use to be the following: general condition of the assembly and its main parts, dimensional variations, corrosion layer thickness and fuel rod tightness.

Within the Electrotechnical Research and Development Program (PIE), and with participation by IBERDROLA, TECNATOM and ENUSA, an inspection system (SICOM) has been developed for spent fuel assemblies from pressurized water plants, the aim being to check: general condition, apply dimensional controls and measure the oxide layer on the peripheral fuel rods. This equipment was qualified at Tecnatom and Almaraz I NPP during the first quarter of 1995. Subsequently, in September, it was validated for the EDF P'4 plants at C.N.P.E. Belleville.

2. Equipment Configuration

Figure 1 is a general representation of the SICOM equipment. The main parts of the equipment are described in the following paragraphs.

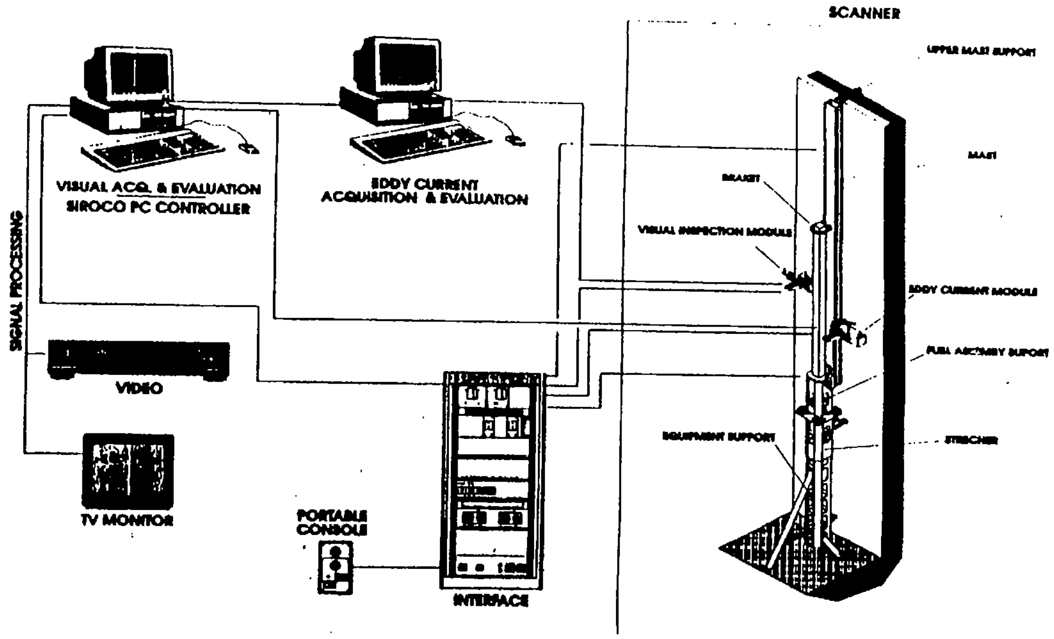

Figure 1. General Presentation

2.1. Mechanical Equipment

From the moment this equipment must be installed in spent fuel pools of different plants, it is necessary to adapt the equipment to fit in the different existing configuration.

Apart from the elements required to fit the equipment to each particular site, the mechanical equipment is designed to safely hold the fuel assembly, make it turn so each face can be inspected and provide precise axial movement of the inspection modules.

The main sub-assemblies of the mechanical equipment (fig. 4) are as follows:

- **Upper anti-seismic assembly**: This assembly, from which hangs the rest of the mechanical equipment, is anchored outside the pool and allows the overall assembly to be levelled. It is designed such that the SICOM is capable of withstanding an earthquake, even though a fuel assembly is being inspected at the time.

- **Mast**: This hangs from the upper support and, supported by it, rests on the floor of the pool. The inspection modules are displaced along its faces via linear guides and rack and pinions.

- **Mast support**: This provides the base for support of the equipment on the floor of the pool.

- **Fuel assembly support**: This supports and rotates the fuel assemblies in order for each face to be lined up with the inspection modules.

- **Clamp**: This prevents the fuel assembly from falling, ensuring the security of the assembly during the inspection. The clamp has no contact with the fuel assembly.

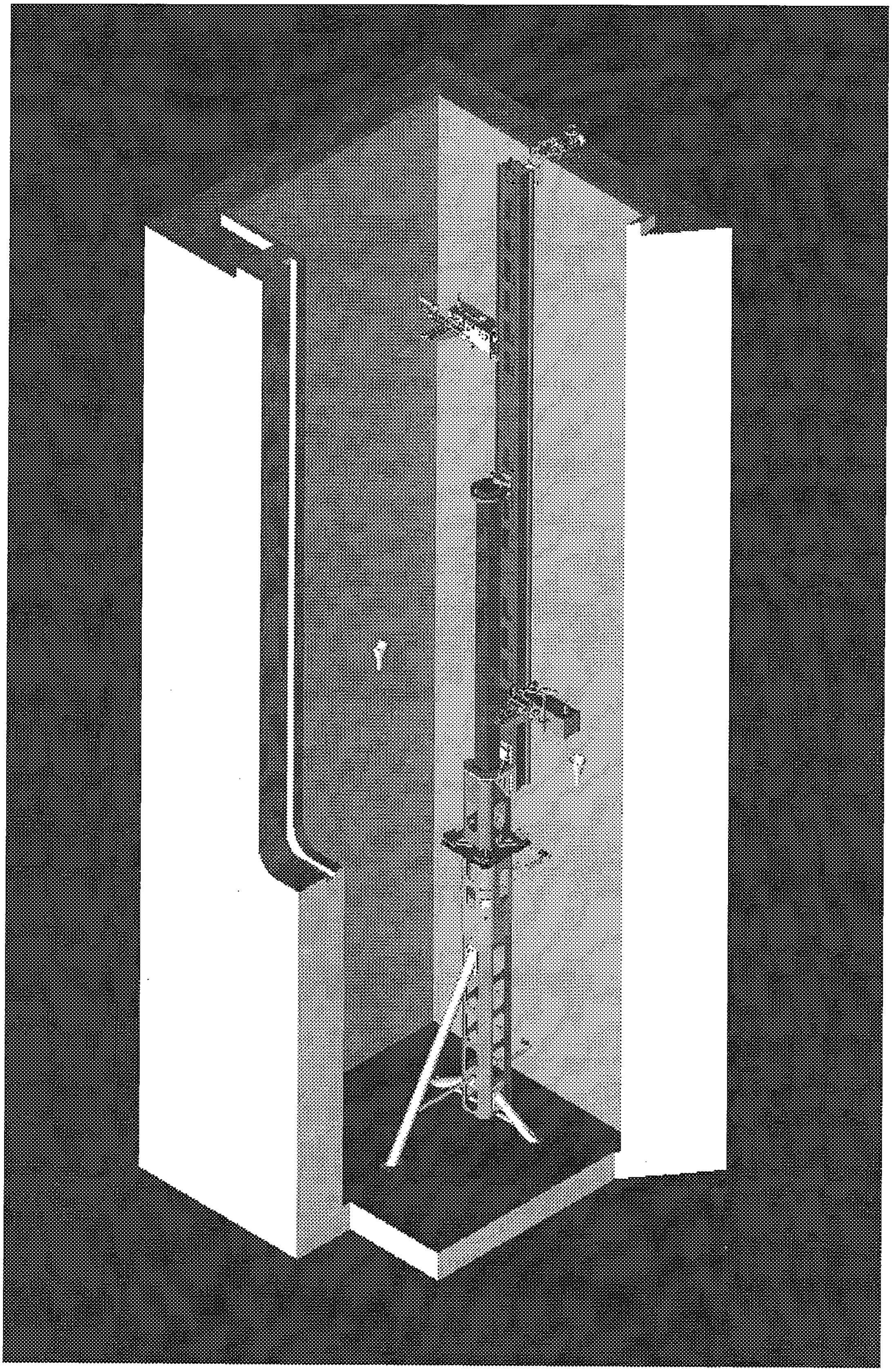

Figure 4. Main Sub-Assemblies

- **Displaceable modules**: These transport the eddy current modules, television cameras, lights, etc. to any position along the length of the fuel assembly faces.

The mechanical equipment is designed and manufactured taking into account the high levels of radiation to which it will be exposed and the fact that it will operate under water.

2.2. *Controller and Interface*

The controller allows operation of all the assemblies of the mechanical equipment to be accomplished, ensuring also indications of the position of the inspection module and of the fuel assembly during rotation.

The SIROCO-PC controller is made up of a computer equipped with a SIROCO-PC card. The hardware used to govern the equipment depends on the inspection to be carried out. To inspect fuel assemblies using the Eddy Current Module, the SIROCO-PC computer is used, while in the case of inspections performed using the Visual Inspection Module control is accomplished by means of the same computer used to acquire the visual inspection data.

The SICOM interface is in charge of providing the assemblies required for control to be accomplished from the SIROCO mechanical equipment. The characteristics of the equipment and the conditions encountered in operation under water require that the interface provide an infrastructure allowing the receptacles in which the equipment motors operate to be kept pressurized, also providing control of the pneumatic modules and electrical supply for the equipment lights. The interface is made up of a cabinet which is located at the edge of the pool and from which emerge four pressurized tubes which run to each of the equipment boxes.

2.3. *Visual and Dimensional Inspection System*

Dimensional measurements aims to measure variation in overall fuel assemblies geometry, namely that caused by the irradiation induced growth of the components, and the general distortion of the assembly.

Fuel assembly sketches showing the kind of dimensions are included as Figures 2 and 3. They represent the following characteristics:

a) Distance between top and bottom nozzles measured in the center of each face. Accuracy ±1 mm. (Characteristic L1, in Figure 2).

b) Length of all peripheral rods. Accuracy ±1 mm. (Characteristic L2 in Figure 2).

c) The rod-to-nozzle gap of all peripheral rods on each nozzle. Accuracy ±0,3 mm.

d) The gap between rods in the center of each span. Accuracy ±0,3 mm. (Characteristic A in Figure 3).

e) Assembly distortion, that is bow (Characteristic P in Figure 3), tilt (Characteristic D in Figure 3), and twist. Accuracy ±0,3 mm.

f) Height of the top nozzle springs. Accuracy ±0,3 mm. (Characteristics L4 and L5 in Figure 2).

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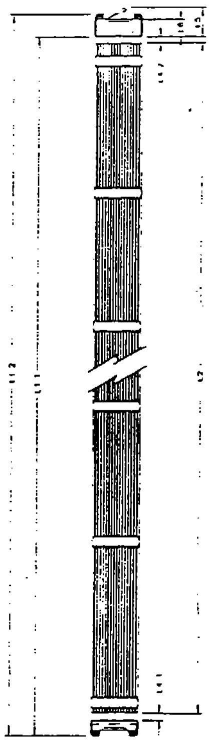

Figure 2. Dimensional Magnitudes
Length Measurements

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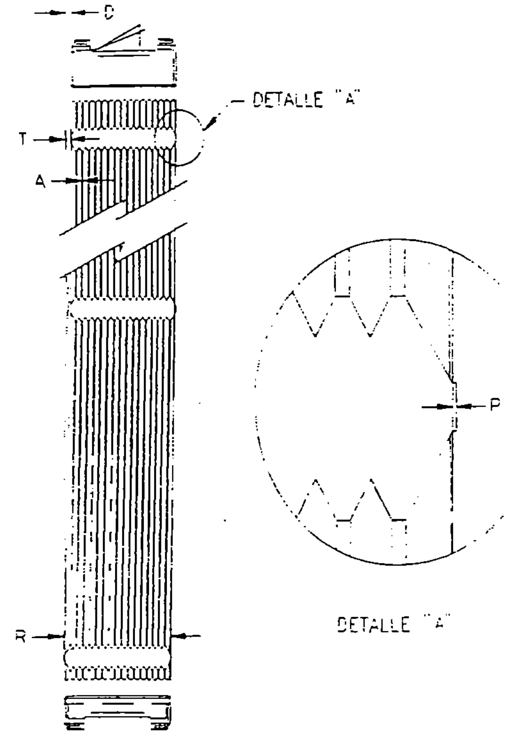

Figure 3. Dimensional Magnitudes
Horizontal Measurements

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g) Grids width (Characteristic R en Figure 2). Accuracy ±0,3 mm.

The basic method utilized by SICOM system to achieve such a dimensional measurements is a follows:

The camera is positioned at the edge of the characteristics to be measured. The edge coordinates are obtained by a digital image processing system that has been previously calibrated using a standard length gage. Then the camera is moved to the other edge and the coordinates will be obtained as above. Then the characteristic values is calculated substrating both coordinates. (This method is applied to the b), c), f) and g) measurements from above).

For a) measurement, the total length (characteristic L1) is given by the distance between the upper edge of the bottom nozzle and the lower edge of the upper nozzle, at which is added the nominal bottom nozzle height.

The rod-to-rod gaps, measurement d), are obtained directly with the digital image processing system.

For fuel assembly bow measurement, at each face two reference straight lines are drawn along the corresponding corners of the bottom and top nozzles. Then the horizontal distances between each grid corner edge and the respective reference line are determined at both corners of the grid (see characteristic P in Figure 3). The bow value at each grid is the average of the two measurements for that grid.

For tilt measurement, at each face a vertical reference straight line containing the bottom nozzle corner edge is established. Then, the distances between top nozzle edge and the reference line are measured (see characteristic D in Figure 3).

For twist measurement a vertical reference straight line containing the bottom nozzle edge is established. Then, the distances between each grid and top nozzle edges and the reference line are measured. The twist value is obtained by subtracting the measurements at both corners of the grid or top nozzle.

The visual inspection system is used to check the general conditions of the fuel assembly (physical integrity, fuel rods, grids, nozzles, springs).

The system is made up of an inspection module and the data acquisition, processing and storage equipment. The inspection module, located on the mechanical equipment, includes a radiation-resistant television camera and four spotlights. The acquisition system is made up of a personal computer including a digitizer card and software for image processing and for automatic calculation of the corresponding measurements by means of artificial vision algorithms.

The movements to be made by the mechanical equipment for data acquisition are also automatically programmed from the visual inspection computer, along with those for the corresponding illumination. The SIROCO-PC controller executes the movements and actions generated. All the information from the visual system may be stored in the computer as well as in the video recorder.

2.4. Oxide Layer Measuring System

Measurement of the oxide layer on the peripheral fuel rods is accomplished using eddy current techniques based on measuring the separation between the sensor and the cladding material, since the oxide layer is not conductive. Under normal conditions the thickness of the oxide layer to be measured may be between 20 and 60 micra. The accuracy of the measurement to be performed is 3-5 micra.

The system consists of an inspection module and a data acquisition system for data processing and storage. The inspection module includes the sensor (coil) which moves in constant contact with the fuel rod, and a television camera for observation of sensor coupling. The acquisition system is made up of a personal computer including the eddy current equipment and corresponding specific software. The results of the measurements performed may be displayed on screen and printed out. Figure 5 shows an example of oxide layer thickness data display.

3. Qualification Test at Tecnatom

Qualification of the SICOM equipment at TECNATOM was performed using a mock-up of a 17x17 AEF-XL type fuel assembly previously characterized for this purpose at the ENUSA installations.

The inspections were carried out in an 8-metre deep trench full of water, in order to reproduce as closely as possible the inspection conditions existing at the nuclear power plants.

The different phases of the qualification process were as follows: assembly of the equipment, installation of the mock-up on the inspection equipment, inspection and disassembly of the equipment.

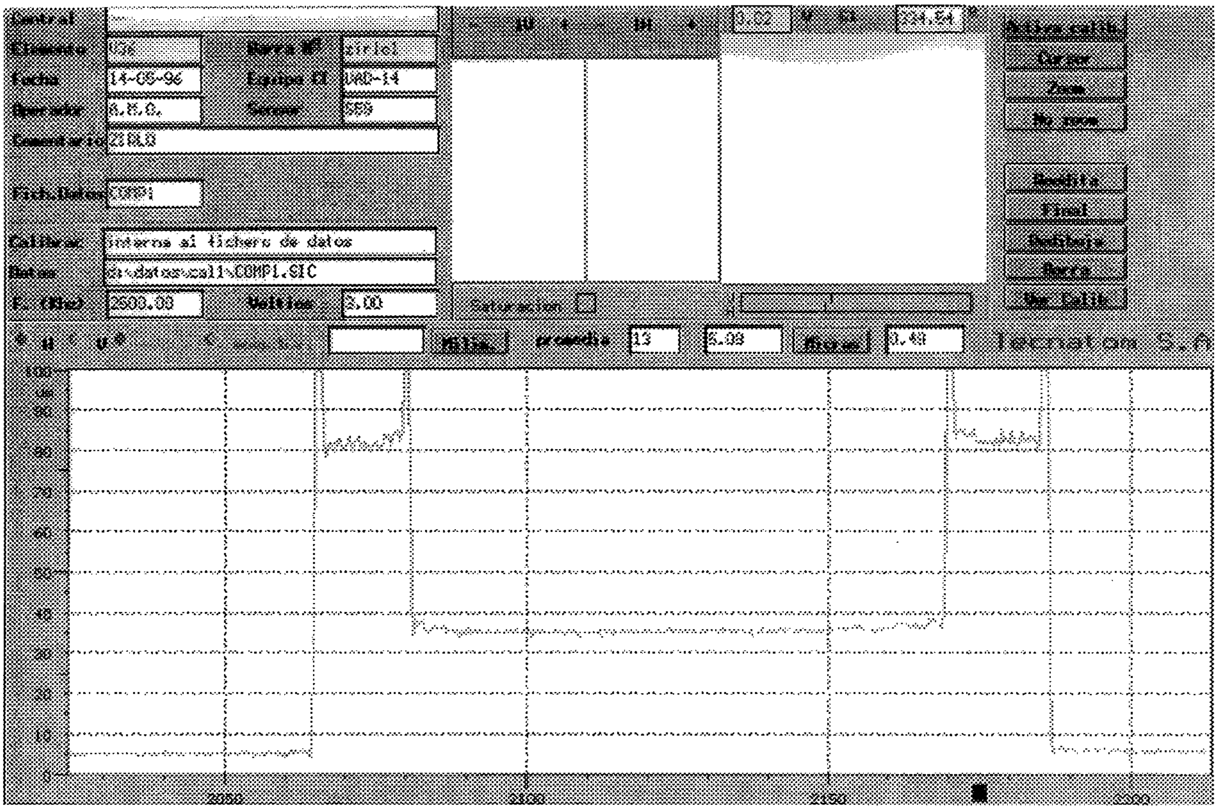


Figure 5. Example of Oxide Layer Thickness Data Display

During the assembly, mock-up installation and disassembly phases, checks were performed of the difficulties encountered in execution, the times involved and the safety conditions.

The inspection phase consisted of inspecting the following characteristics on the mock-up:

- Visual inspection of physical integrity.
- Dimensional measurements:
  - Channel spacing
  - Separation between peripheral rods and nozzles
  - Length of peripheral rods
  - Length of mock-up
  - Lateral width of grid
  - Height of upper nozzle
  - Bowing

  - Twist

  - Perpendicularity

- Corrosion measurements:

  Corrosion measurements were performed on four bushings (two of Zircaloy and two of Zirlo) located on four different mock-up rods.

The results of the qualification process were as follows:

- Visual inspection of physical integrity.

  The equipment demonstrated its capacity for the performance of this inspection with high quality imaging.

- Dimensional inspection.

  The equipment obtained the measurements with a degree of accuracy similar to or better than that specified.

  The bowing, twist and tilt measurements are qualified by means of a fuel prototype especially conceived for this purpose.

  The repeatability achieved in measurement was also very high, the accuracy obtained thus being guaranteed.

- Corrosion inspection.

  The variations in the values obtained by the equipment, with respect to the values obtained using metallographic methods, were lower than the specified degree of accuracy of ± 3 micra.

4. On-Site Validation Tests

4.1. Validation at Almaraz NPP

This phase of the qualification process consisted of repeating the process indicated in section 7 at the Almaraz I Nuclear Power Plant, with the equipment installed in the spent fuel pool.

This validation process also served to qualify equipment packaging and transport.

The main difference with respect to the qualification performed at TECNATOM was that an actual irradiated fuel assembly (17x17 STD) was used for the inspection.

The scope of the inspection was as follows:

- Visual inspection of fuel assembly physical integrity.

- Complete dimensional inspection of a fuel assembly.

- Corrosion inspection on 8 peripheral rods.

The results of this validation were as follows:

- Visual inspection:

  The equipment demonstrated its capacity for the performance of the inspection on an irradiated fuel assembly, with high quality images obtained.

- Dimensional inspection:

  The equipment demonstrated its capacity to perform the measurements on an irradiated fuel assembly.

- Corrosion inspection:

  The equipment demonstrated its capacity to measure corrosion on irradiated rods.

  In addition, in this case the values obtained were compared to those measured independently by another unit of equipment homologated for this type of inspection, no significant differences being appreciated.

This test also served for validation of the transport, assembly, inspection, disassembly and decontamination of the equipment at a 900 MW PWR plant (Almaraz).

4.2. Blank Test At C. N. P. E. Belleville I

The final blank test carried out at the 1300 MW Belleville plant consisted of demonstrating the full operability of the equipment at this plant, for the inspection of 17x17 AEF-XL fuel assemblies.

This test was performed, at Unit 1 of the aforementioned plant. Once the equipment had been assembled, the inspection was carried out on a fuel assembly mock-up supplied by the plant, along with different safety tests, all giving satisfactory results.

5. Services Performed and Future Undertakings

The SICOM Operations Committee was set up for management of the services provided, this including participation by D.T.N., TECNATOM and ENUSA.

During the month of February of 1995 the first inspection service using the SICOM equipment was performed at Almaraz II NPP.

The inspection was performed on two demonstration assemblies with Zirlo cladding with two burnup cycles and on a spent fuel assembly present in the fuel pool.

Visual physical integrity and dimensional inspections were performed on the three assemblies. In addition, a total 56 Zirlo fuel rods were inspected for corrosion, along with 32 Zircaloy-4 rods, on the two demonstration assemblies. On the other fuel assembly, 15 Zircaloy-4 rods were inspected for corrosion.

During the month of May of 1995 the second commercial SICOM service was carried out at the Vandellós II Nuclear Power Plant. Initially, within the segmented rod project, four fuel assemblies that had undergone their first burnup cycle were inspected, with general visual, dimensional and oxide layer corrosion controls being performed.

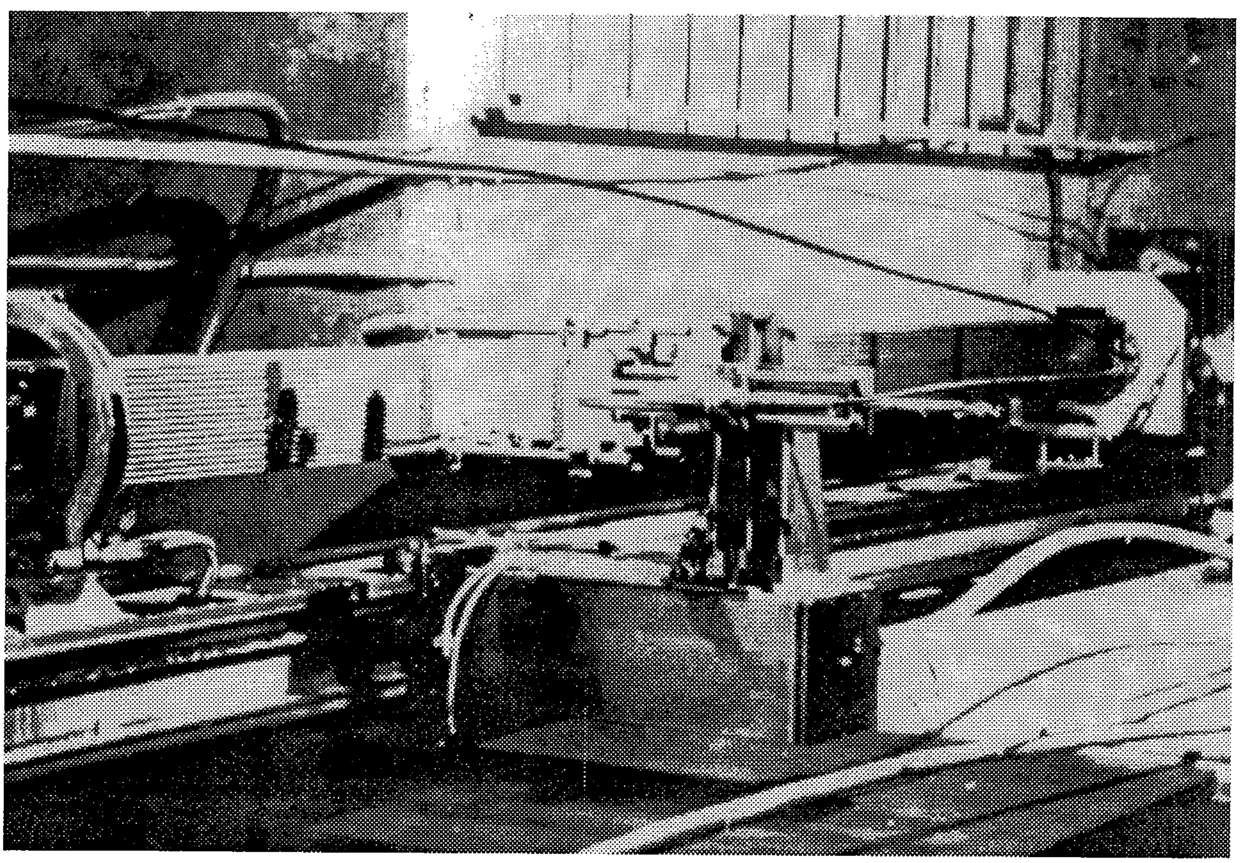
In view of the possibilities offered by the equipment, Vandellós later requested that the service be extended to include the performance of rod corrosion measurements on a further five spent fuel assemblies present in the fuel pool.

This inspection was performed with success in a short period of time.

In September of 1995 and following the final blank test, fuel assemblies were inspected at Unit I of C.N.P.E. Belleville.

The scope of the service included visual integrity control, complete dimensional control and measurement of the oxide layer on peripheral rods.

In 1996 the following services was carried out:



SICOM

- Almaraz II, may: Inspection of 18 fuel assemblies.

- Vandellós II, june: Inspection of 4 fuel assemblies for the Segments Bars Project and 8 fuel assemblies for the Collaboration Program with spanish NPP's.

There are currently undertakings to use SICOM in the medium term for the following projects:

- Segmented rods at Vandellós II.

- LFA program at C.N.P.E. Belleville I.

- ADA program at C.N.P.E. Belleville I.

- Gadolinium program at Ascó II.

- Zirlo program at Almaraz II.

- Fuel Assemblies Desmostration at Trillo NPP.

- Collaboration program with spanish NPP's.

6. Conclusions

The qualification and validation tests performed, along with the experience acquired in commercial use of the SICOM equipment, has made it possible to demonstrate the capacity of the equipment for the performance of quality inspections on irradiated fuel. Particularly outstanding is the high degree of accuracy achieved in dimensional measuring and the magnificent image resolution obtained in the visual inspections, along with the large number of rods on which corrosion measurements were obtained.

The advisability of having available an advanced item of equipment for fuel inspections, developed by Spanish industry, is confirmed by the significant medium-term demand foreseen for use of the equipment.

TECHNICAL SESSION III

EDF Experience about RCCA Behaviour

X. Thibault
EDF/SEPTEN

Abstract

The concerns which may affect the RCCA component are known today. The cladding wear, due to hydraulic forces in the guide tube, remains the first RCCA life limiting factor on the 900 MWe and the 1300 MWe reactors. The clad swelling induced by the buckling of the SIC (Silver Indium Cadnium) absorber material is also a potential problem but, up to now, no jam, occuring inside the core, may be attributed to this factor. A lot of monitoring campaigns have been performed since 1988. The inspections have allowed to identify the wear amplitude and the swelling when sufficient to induce cracks.

Three actions were initiated, mainly to deal with these problems:

The first concerned the immediate requirements and consisted in defining a RCCA management strategy based on mechanical calculations. This strategy includes periodical controls, axial repositioning and specific rejection criteria. The criteria are based on safety considerations like the mechanical integrity of the RCCA, the integrity of the absorber itself and the primary coolant activity.

The second is devoted to some design improvements of RCCA. Regarding the wear problem, various hardening methods of the clad have been investigated (nitride, chrome carbide, chrome). For the swelling, geometrical modifications should delay the interaction between the absorber and the clad.

The third action, is a long term work with a more ambitious approach. It consist in a large R&D program concerning the study of the phenomena involved in the wear process. The aim is to build a full set of physical models and to introduce them in a computer code, in order to be able to predict RCCA wear.

The present status of these actions is the following:

The management strategy is now in current use on all the reactors and allows to operate them in safe conditions, even with standard RCCA. This strategy will require some adjustments when reactor operating conditions are modified.

Out of pile tests have showed the potential efficiency of various remedies. These remedies are progressively introduced in the reactors. Now about 1000 new design RCCA are in reactor and the oldest ones have yet experienced 8 cycles. The results already obtained confirm the better behaviour of these RCCA compared to the standard one.

The third action is still in progress. EdF has planned to qualify the code in 1997. Then it will be possible to perform analytical studies to investigate modifications on the design of RCCA or internal guide-tubes.

1. Introduction

Two important safety functions are ensured by the RCCA system: quick insertion of anti reactivity during incident or accident conditions and safety state in all the shutdown conditions. These functions has to be preserved in any case. So, no adverse damage, on this system, are acceptable.

Wear and swelling problems has been encountered in all EDF reactors and appear to be potential limiting phenomena. An important program of pool side and hot cells examinations allowed to understand the main causes and the kinetics of the phenomena.

In order to address these problems EDF has set up since 1988 a management strategy for the RCCA system, based on industrial controls and rejection criteria. This action allowed to operate the reactors in safety conditions despite the existing problem.

Nevertheless the control and RCCA replacement costs are so important that a better solution is desirable. So, EDF has asked to the vendors in the same time to do their best proposal to reduce significantly the phenoma. By the end of 1988, the first experimental new design RCCA have been introduced in reactor.

Moreover a large R&D has started in conjonction with the two previous action. The aims of the program were to identify the various parameters which contribute to the RCCA wear damage and to quantify these parameters in order to develop a computer model to be able to predict the phenomena.

All these actions are presented in the next paragraphs.

The main design features of the EdF plants, internal guide tubes and RCCA's, which have some importance regarding RCCA problems, are gathered in table 1.

2. Feed back experience

At the very beginning RCCA clad wear has been considered as a potential problem but essentially because of the sliding against the internal guide tube during the step by step motions. As load follow operating conditions could emphasise the sliding wear (some RCCA could accumulate a total of steps motion exceeding 500 000) EdF decided to examine some RCCA to assess the actual amount of the wear. The measurements has performed by Eddy current and Ultrasonic technics which have given the worn area or worn depth. The very first RCCA control on EdF reactors had been performed in 1984 on BUGEY4 plant. On this plant, the wear appeared very low like in many others T hot reactors in the world.

However others controls done in 1986 on six T cold 900 MWe reactors showed a lot of wear particularly in front of the guide tube cards.This wear problem mainly affects the shutdown rods which are parked at the same axial position all along the cycle. On 1300 MWe plants if the amplitude of the wear phenomena appeared lower it affected quite all the RCCAs but essentially on the tip.

These observations indicate that the main wear mechanism is in fact the fretting induce by hydraulic forces inside the guide tube of the upper internals, and it has been confirmed by out of pile test (see § 5). As 900 MWe and 1300 MWe guide tube have different designs the hydraulics force and so the wear kinetic are not the same.

In 1988 - 1989 two incidents occurred respectively in DAMPIERRE 1 and GRAVELINES 4 (CPY plants) where one rod failed in front of 7th card (total number of cards is 8 on 900 MWe). Expertises, carried out on the two rods, revealed: the mechanism of the failures which is fatigue damage and the specific action of the SIC rod upper end which increases the bending stresses [1].

Since 1984, more than 150 RCCA control campaigns has been perform on the 900 MWe and 1300 MWe EdF reactor. The results shows a large scatter in wear from one reactor to an other or from one position in the core to an other one. Nevertheless some trends emerge concerning the worst rods positions in the cluster, the evolution of the kinetic and the various shapes of the wear. For example some core positions looks more sensitive than others and the single rod vanes inside the cluster are the more affected position. But one of the most important is the following; if the situation remain the same the wear kinetic decreases in time.

With the same equipment used for wear control some clad swelling has been identified in 1989 on 1300MWe reactors and in 1990 on 900 MWe reactors. Such swelling are located at the bottom of the rod near the tip where the fluence is maximum (figure 1). In fact the clad swelling is a consequence firstly of the gap closure due to SIC absorber creeping and packing down during the step by step motions and secondly of a very progressive SIC swelling induce by transmutation of indium into tin.

All RCCAs are affected by swelling after several years (at least 6 years for 900 MWe plants) but temperature and power control RCCA banks are more susceptible because of the higher insertion of SIC in the core. This phenomena has no impact on shutdown timing but should damage the assembly dashpot or cause RCCA jam. But such situation has never been experienced till now.

In addition the swelling is generally associated with clad cracks. The potential consequences of unwatertight clad are Silver 110 m contamination for all RCCA filling with a SIC rod. But in 1300 MWe reactors the shutdown RCCA's contain also B4C pellets which is a soluble material if irradiated [2]. Hot cells examinations, carried out on several swelling rods, demonstrate the absence of contact between water and B4C due to the strong binding of the absorber and the clad in the swelling zone.

3. RCCA management on 900 MWe and on 1300MWe reactors

3.1. Description

The numerous wears find on RCCA leads EdF to define a management strategy in order to avoid unacceptable incidents like those appeared in 1988 but also to take into account the swelling problem. For 900 MWe reactors a management strategy has been defined in 1988 and, after some modifications, fully established in 1990. For 1300 MWe reactors, as these reactors are quite new, it take more time to settle a management strategy. A full set of calculated criteria has been established in 1992. However, in 1993, EDF decided to convert all the 1300 MWe plants from T hot into T cold. So due to the negative effect of these new operating conditions it was necessary to define again the rejection criteria mainly regarding the wear in front of cards (the wear in the lower part of the RCCA should not be significantly modify). The lack of experience with the new operating conditions lead to a new preliminary management strategy. The study of the new set of criteria is still in progress and should be available in 1997.

In the two cases, 900 and 1300 MWe plants, the management strategy has three aspects:

- a full set of rejection criteria based on mechanical studies
- a frequency control
- a procedure for RCCA repositioning

Those three aspects are linked. In any cases shuffling is not permitted, this is due to the fact that wear kinetics used to define the criteria are based on wear measurements obtained cycle after cycle without any modification of the RCCA-guide tube. On the contrary axial repositioning is used systematically with three allowed level for 900 MWe and five for 1300 MWe. The RCCA repositioning is performed after every cycle. This procedure spreads on independent zones the wear and increase the life time of RCCA's but it does not improve the situation regarding wear on the tip (worst zone on 1300 MWe) because the assembly guide tube is a continuous element.

Frequency control is based on the wear and swelling kinetics. The control is performed every three cycles on 900MWe reactors and every two cycles on 1300 MWe reactors. The inefficiency of axial repositioning for wear tip, explain the difference between the two situations.

The rejection criteria are defined taking into account the calculated limiting wear or swelling, the kinetic of the phenomena before the next control and the axial repositioning procedure (just for wear in front a card).

For 900 MWe plants the absorber material is only SIC so a leak is not a safety concern but the criteria has to prevent from rod failure. But for 1300 MWe the potential risk of B4C dissolution requires to avoid any clad leak.

To determine the wear limits it is necessary to assess the actual forces acting on the component and to know the actual wear shapes. The first are measured on a specific mock up and the second come from hot cells examinations. An another important parameter is the maximum number of steps the various RCCA group can done. For RCCAs which move during load following (bank G) the criteria is relativly lower because of the leading fatigue mechanism.

From mechanical point of view, swelling of the clad in the lower part could increase significantly the over-pressure when the RCCA go into the dash-pot assembly during a shutdown. The RCCA speed at the entrance of the dash-pot and the axial shape as the location of the swelling are the main parameters governing the over-pressure.

3.2. Application

The figures 2 and 3 give the number of rejected RCCA by application of these management strategies. In fact, as far as swelling is concerned, in 1990 on 900 MWe reactors, the RCCA was rejected when it exceed about half of the criteria limit because of the quite unknown kinetic of the phenomena at that time. For 1300 MWe, as mentioned before the criteria are well established at the beginning of 1992, so previously just empirical criteria are used.

On CPY reactors (900 MWe T cold) the evolution of the rejection rate, after decreasing during three years, increases again to reach a rate excending the rate during 1990. This fact suggest that from the kinetic point of view the replacement of RCCA give a situation less favourable than the initial one. RCCA are rejected mainly for wear on the cards; about 50 % of the total rejection number (figure 4). The large number of rejected RCCA is due to the conservatism of the criteria but it confirm the high level of wear on this type of reactor.

It is a little more difficult to interpret the evolution of the rejection on 1300 MWe reactors because some plants are recent and the rejection criteria has been modify during this last period. However the trend seems to be the same than for 900 MWe with a continuous increase of the rejection number. The comparison of the rejection rate before and after 1993 indicates an increase of the wear card with the conversion in T cold of all 1300 MWe plants, but not a drastic one; as it could be feared. Note that the higher level in 1994 for wear card is mainly due to one specific plant.

For 1300 MWe plants swelling and wear tip appear to be the main causes of rejection even if some card wear are found since 1993 (figure 5).

4. RCCA new design

4.1. Wear remedies

In parallel to the establishment of the RCCA management strategies EdF has asked to FRAMATOME to develop new RCCA design to solve or reduce the wear and swelling problem.

In this purpose FRAMATOME set up a R&D program consisting in autoclave wear test in order to select the best wear remedy. Finally, FRAMATOME choose the ionic nitrogen treatment for the absence of any adherence problem, the cheap industrial process and its good efficiency regarding the wear even in very conservative fretting conditions.

Others remedies were also proposed by the suppliers as chromium carbide coating and hard chromium coating. Out of pile tests indicate also a good wear resistance for such coating.

EdF decided to test in reactor as soon as possible all these solutions, to assess their actual behaviour. They have been introduced progressively since 1988/1989 for nitrogen treatment and chromium carbide coating and since 1992 for hard chromium [table 2]. All these RCCA are inspected regularly as defined by the general management strategies. And for the very first one systematic ultrasonic measurements have been performed in order to detect the incipient wears.

The results show a very good wear resistance behaviour for nitrogen treatment and chromium carbide coating. On the sixteen nitrogen RCCA loaded in CRUAS 2 none wear has been detected after one cycle whereas some standard RCCA's have to be rejected for exceeding criteria value after the same time [figure 6]. The very limited wear measured on one rod of a nitrogen RCCA loaded in PALUEL 2 is just over the range of the uncertainty and is located on the tip where the treatment is lower due to end effects in the fabrication furnace. Improvement of nitrogen process has been adopted in 1991 and a very good homogeneity of this treatment is now obtained. In the same reactor the wear measured on chromium carbide plated RCCA is also at the lower end in an uncoated zone (few centimetre near the tip are uncoated zone).

Very few measurements have been done on chromium coating RCCA. The available results indicates a general good behaviour with also some wears on the tip in an uncoated zone. This problem should be solve by an increase of the coating length.

In 1996 the first decrease of the rejection rate for 900 MWe plants could be attributed to the beneficial effect of the various wear remedies. The same trend should see for the 1300 MWe plants in 1997 because the new design RCCAs have been introduced in a large scale more recently. Therefore, in the next year the experience feedback should increased as now the total amount of wear remedies RCCA in reactors exceed 1000.

4.2. Swelling remedies

To improve the situation regarding the swelling problem a very simple remedy has been adopted. It consists in an optimisation of the absorber-clad gap in order to delay the interaction between the two components. The increase of the gap is however limited because the neutronic worth of the control rod has to remain constant in addition a large gap could lead to a higher absorber temperature. This could be compensate by filling the rod with helium.

Since 1992, all the new RCCA put in the EdF's reactors have such improvement and no rejection for swelling has encountered up to now. But as there is a threshold before to see an incipient swelling (at least 2 cycles) this situation is not very surprising.

5. A model for wear prediction

The two previous actions allow to control the situation after the two incidents occurred in 88 and 89 and improve it by introducing gradually new design RCCA's. But the RCCA maintenance is very expensive, particularly if inspection operation is on the critical path of the shutdown period of the reactor. Using new design RCCA with wear remedy should improved significantly the situation but for how long? It could be more profitable to reduce the wear causes itself rather than to look after the consequence.

So in addition to the two previous actions a complementary approach is to study remedies to mitigate the hydraulic forces which induce vibration of the cluster rods. This could be done by direct in loop test but such device is not very flexible and not fully representative of in reactor conditions. An other way is the construction of a full computer model which can simulate all the steps from hydraulic flows all around cluster rod to the cladding wear .The aim of this work is to obtain a comprehensive answer regarding wear problem.

EdF set up this R&D activity since 1990, partially in collaboration with FRAMATOME and CEA, with a large programme which has last for several years. The EDF Research and Development department is in charge of this work.

The programme is divided in three parts:

- characterisation of the hydraulic sources,
- study of the mechanical behaviour of cluster rods subject to the hydraulic forces,
- study of the wear kinetic.

All these parts are bound and are based on mock-up tests to characterise the main parameters and to validate each modelling steps.

The goal of the first part is to identify the stimulating hydraulic sources and to quantify the hydraulics forces which act on the RCCA rods. The identification has been done on FRAMATOME MAGALY from the 900 MWe guide tube tests (figure 7). Two main hydraulic sources has been found:

- the first is located in the lower part of the guide tube, with two components: the main flow going outside the continuous part of the guide tube to the upper plenum and the residual flow going up inside the guide tube,

- the second takes place at the upper part of the guide tube, where the RCCA go through the house keeping.

In the upper part the source is a combination of fluid elastic forces and turbulent forces while in lower part only turbulent forces are considered. To study these sources, two mock-up has been built: GRAPPE 1 for the lower part and GRAPPE 2 for the upper one [figure 8]. Specific analytical methodology call "reverse assessment of turbulent hydraulic forces" are developed in such way that its can be used as input in a computer code.

The second part is devoted to the determination of the RCCA behaviour submitted to the previously calculated hydraulic forces. This requires the development of non linear dynamic models with sliding and impact forces. The MASSIF device has been used to validate the models considering the hydraulic forces in order to assess the displacements, the impact forces and the wear powers at the various level of the cards. The tests are performed in various conditions (air-water, mono-biaxial stimulation, gap size....).

After completion of these two parts it is necessary to validate the hydraulic and mechanical model on a specific mock-up. This mock-up call MOCHE simulates only one rod and only one support with a gap. The comparison between calculated and measured values are very satisfactory.

The third part concerns the wear modelling in order to predict the volume and the depth of the damage zone, starting from the characterised impact conditions obtained in the former part. The wear mechanism is very complex and dependent on the vibration regime. MAGALY have shown that the impact on the cards are combined with a sliding motion.

In a first approach a mono-axial stimulation machine named VIBRATEAU has been used to study wear kinetic. It can run in various conditions and even in PWR conditions. For example the big sensitivity of the wear to the temperature has been investigated and confirmed by the results obtained. Considering the simple ARCHARD's wear rule it is possible to determine a wear factor corresponding to the ratio between the worn volume and the power of wear forces. Bi-axial tests has been also underway on a new AECL machine call ERABLE which produce more representative impact conditions. All the results contribute to develop a new wear model based on Archard's law but with time effect.

The connection of all the models inserted in the code ASTER is in progress but required more time than anticipated. This computer code has been run for wear prediction in sensitivity studies regarding different parameters like gap size between RCCA clad and card, or RCCA axial position, or clad material. The results has come to light the major wear consequence of slight modifications of these parameters. So it is not very surprising that some difficulties appeared in the benchmarking with the in reactor. Additional work is required to improve the knowledge of the input data and particularly the actual flow entering and coming out the guide tube.

This long modeling development should be very useful in the future to assess the benefit provide by any modification in the RCCA design or in the guide tube design or to optimise the RCCA management strategy.

6. Conclusion

The wear and swelling phenomena which could affect the RCCA clad integrity, are now well known and it is now demonstrated that it is possible to manage it and keep the reactor very safe. But the replacement of a RCCA by a new standard one could be very expensive.

Remedies regarding these two problems has been introduced in reactors. After seven years the ion nitride treatment looks very efficient to reduce drastically the RCCA wear. The exact life time of this remedy has yet to be established and the absence of adverse consequence on the mating surface has to be confirmed. Also the benefit provided by the swelling remedies will be appreciated only in several years.

In addition to these engineering approach EdF has carry out a more fundamental work in order to be able to predict the wear and to test design modifications. The computer model is now finished for 900 MWe plant but qualification is not completed. However all the work performed allows a better understanding of the mechanisms involved in the wear phenomena. Studies performed with such code will contribute to improve the design of all the components involved in the wear problem and the RCCA management strategy.

References

[1] Calculation of RCCA wear reject criteria in nuclear plants C. Bernaudat - K. Christodoulou, SMIRT 12 (1993)

[2] Behavior of irradiated B4C EPRI NP-4533-LD,May 1986

MAIN PLANT & RCCA DESIGN FEATURES

	CPY - 900MWe	PQY - 1300MWe
Number of plants	28	20
Plant type	17x17 - 3Loops	17x17 - 4Loops
Hat temperature	T cold	T cold since 1993
Guide tube type	Deep beam 125"	Inverted hat 96"
Number of RCCA	53	65
Number of rods	24	24
Clad thickness	0,47	0,98
Absorber material	SIC	SIC and B4C
Tubing material	316L	316L

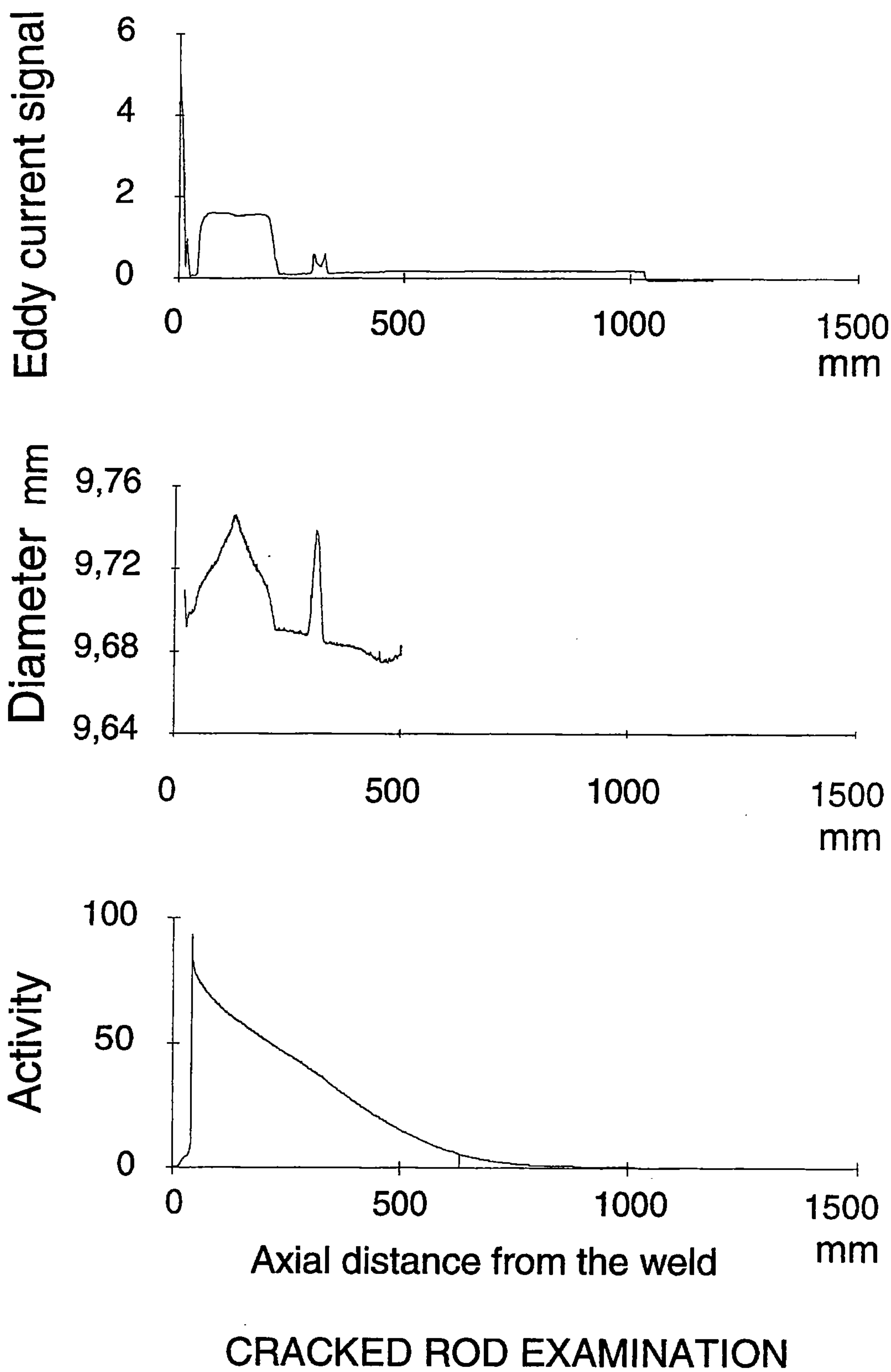
Table 1

EXPERIENCE ABOUT NEW RCCA DESIGN

Year	Type	Number	Reactor	Control 1 cycle	Control 2 cycles	Control 3 cycles	Control 4 cycles	Control 5 cycles	Control 6 cycles	Control 7 cycles
88	Ni	2	CPY	No wear	No wear	No wear	No wear	*	*	No wear
89	Ni	2	PQY	No wear	Tip 50µm	70µm	100µm	160µm	*	*
89	LC1C	2	PQY	Tip 60µm $	360µm	360µm	360µm	380µm	*	*
91	Ni	14	CPY	No wear	*	*	No wear	*	*	*
92	Cr	15	CPY	*	*	1 tip wear $	*	*	*	*

$ wear on uncoated area

Table 2



Eddy current signal
6
4
2
0
0
500
1000
1500
mm
Diameter mm
9,76
9,72
9,68
9,64
0
500
1000
1500
mm
Activity
100
50
0
0
500
1000
1500
mm
Axial distance from the weld
CRACKED ROD EXAMINATION

Figure 1

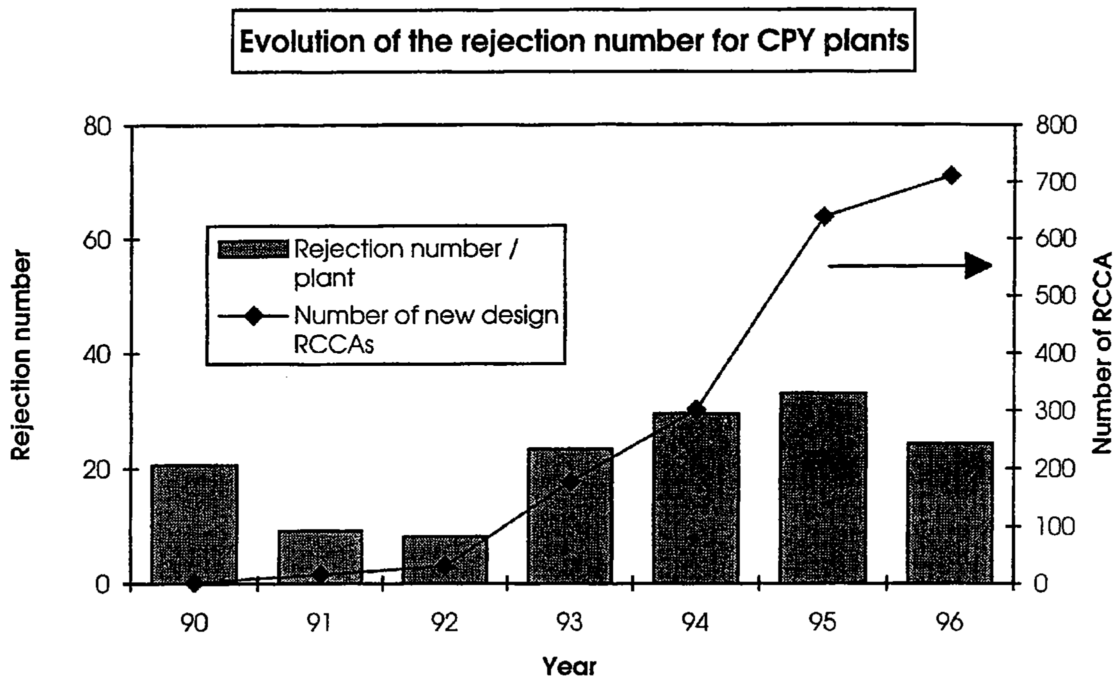


Figure 2

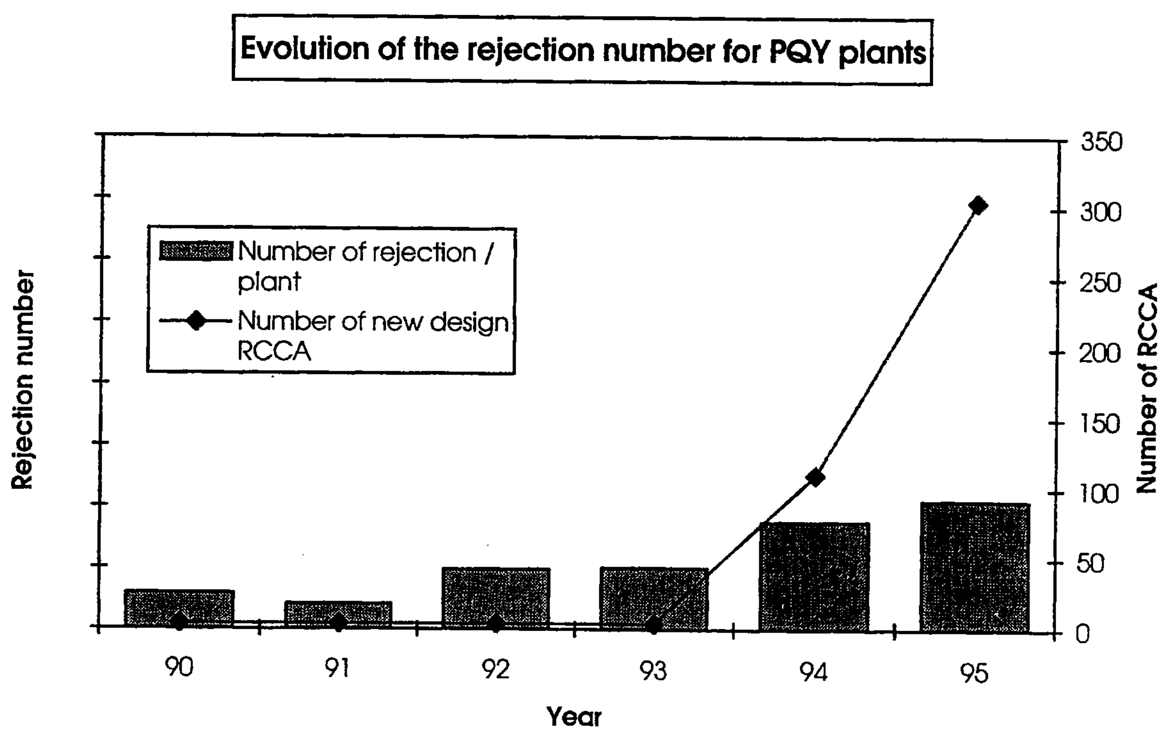


Figure 3

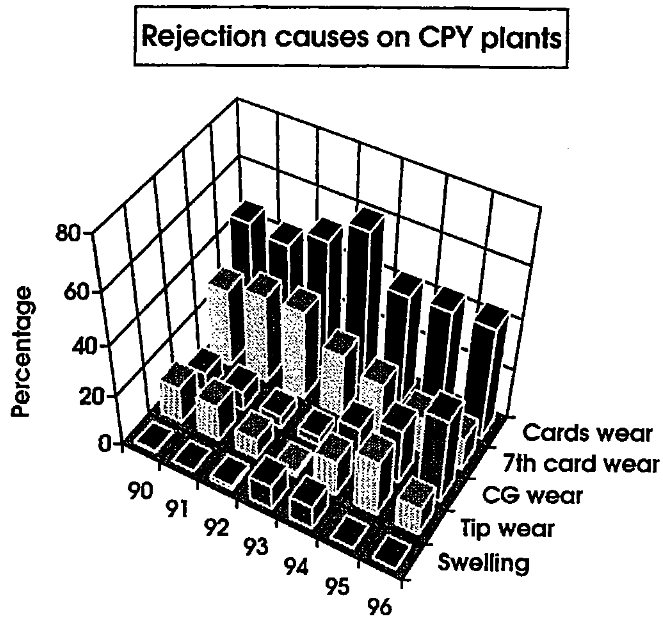


Figure 4

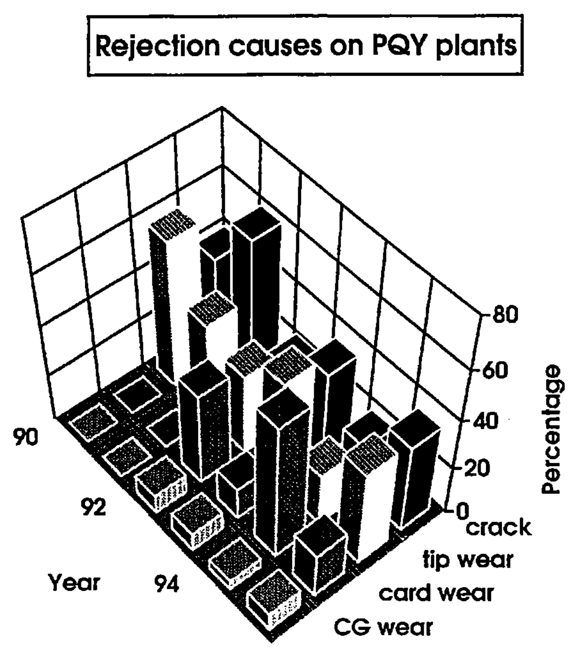


Figure 5

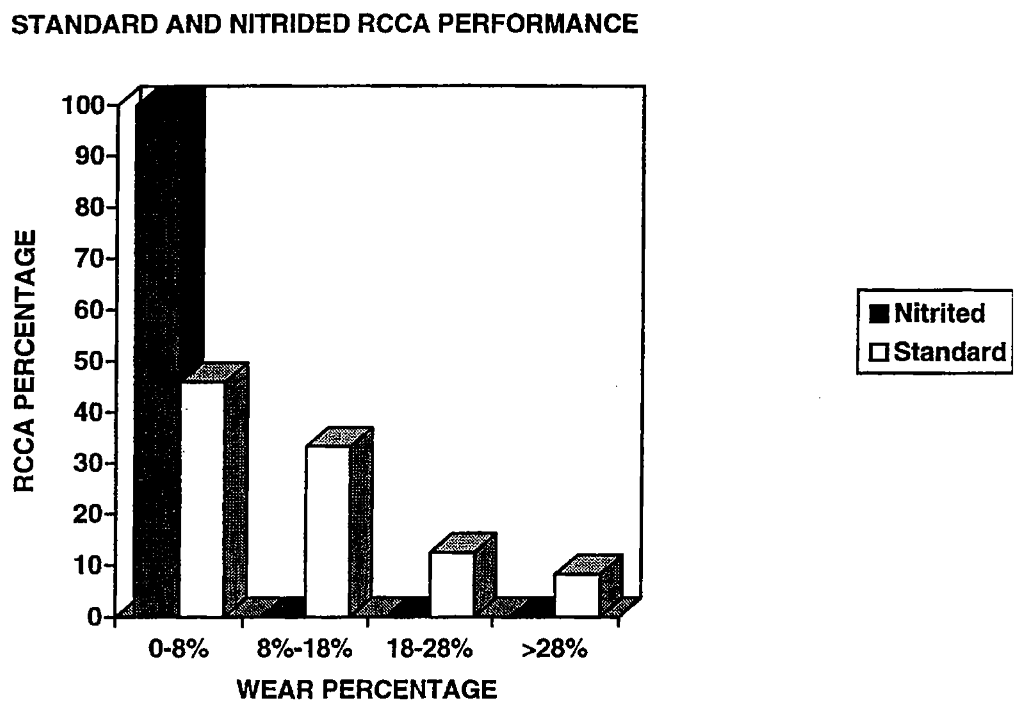
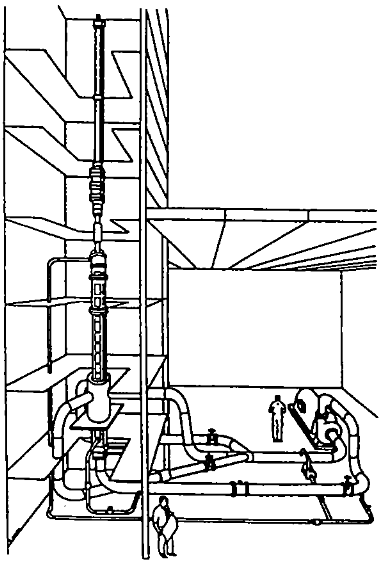
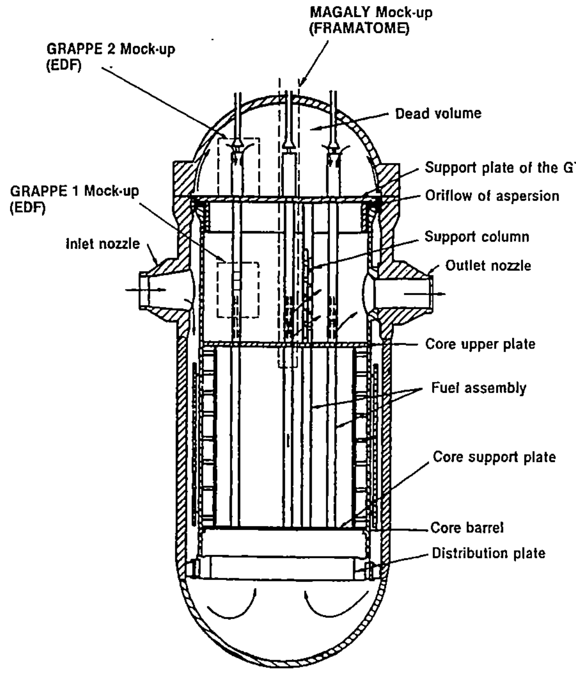

STANDARD AND NITRIDED RCCA PERFORMANCE
RCCA PERCENTAGE
100
90
80
70
60
50
40
30
20
10
0
0-8%
8%-18%
18-28%
>28%
WEAR PERCENTAGE
Nitrited
Standard

Figure 6



MAGALY TEST FACILITY

Figure 7



LOCATION OF HYDRAULIC MOCK-UPS IN THE INTERNALS

Figure 8

Rod Cluster Control Assembly Management for Pressurized Water Reactors

Klaus Knecht, Lothar Heins
(KWU)

Summary

The lifetime of rod cluster control assemblies (RCCA) is of particular interest to operators of reactor plants. Effects such as wear marks caused by RCCA vibration in the reactor's guide structure and the swelling of neutron absorbers which can cause cracks in control rod cladding can limit their lifetime. At the beginning of the 1970s, Siemens/KWU developed inspection methods which permit surveillance of the operating performance of RCCAs.

Absorber swelling and control rod wear can be controlled by an RCCA management program.

1. What are our experiences based on?

At present, four different inspection and measurement methods are used, either individually or in combination:

- Eddy current (EC) examination - a rapid and simple nondestructive testing method used to check the mechanical integrity of control rods.
- Visual inspection for more detailed analysis of general operating performance and particular indications gained by examination
- Profilometry for identification of local clad thinning and wear mark orientation
- Measurement of control rod diameters for long-term evaluation of absorber swelling.

Since 1970, Siemens has performed more than 200 inspection campaigns on RCCAs.

All inspection results have been thoroughly evaluated and channeled back into the design of new RCCAs.

2 How is the operating performance of Siemens-supplied RCCAs?

The life of PWR RCCAs is limited on the one hand by absorber/clad interaction due to absorber swelling and creepdown of the cladding surrounding the absorber and, on the other, by possible wall thinning due to wear. The physical life, i.e. the time up to a reduction in RCCA worth of 10%, is not considered here since it is considerably longer than 30 years [1].

AgInCd is used as absorber material. As a result of neutron capture, different isotopic conversion processes take place, e.g. conversion of In → Sn and Ag → Cd as a result of the resonance capture of epithermal neutrons (0.6 eV < E < 0.8 MeV), the sum of which result in an increasing volume of the absorber as a

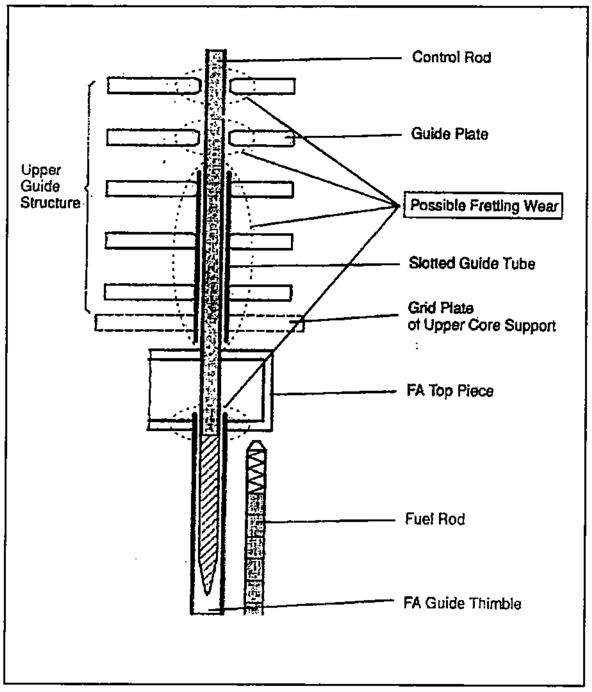


Fig. 1 Possible fretting wear on a fully withdrawn control assembly

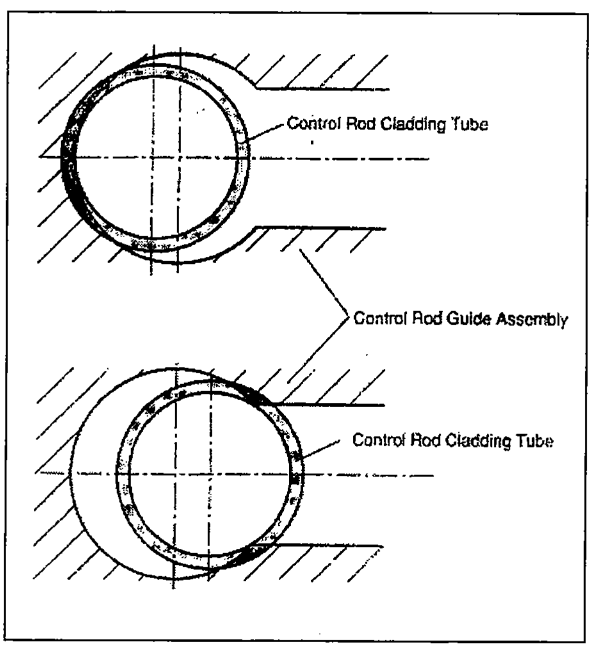


Fig. 2 Possible shapes of wear marks on control rods

function of neutron fluence [2]. The pressure inside the reactor vessel produces an external overpresssure acting on the cladding, resulting - in connection with neutron irradiation - in creepdown of the steel cladding. After gap closure all further absorber swelling leads to tensile stress in the clad. The allowable residence time is limited by the ductility of the cladding material.

In some reactors, RCCAs experienced up to 21 cycles with a fast neutron flux reading of 7 x 10^{22} cm^{-2}. Only in a single case did absorber swelling result in irradiation-assisted stress corrosion cracking on a few control rods.

Wear of control rod cladding may occur as a result of axial movement of the RCCA and vibration of the control rods against the inner walls of the upper core structure tubes or plates, vibrations of the guide assemblies themselves, and/or vibrations against the inner walls of the guide thimbles of the fuel assemblies, see Figure 1.

Wear causes wall thinning, resulting in an increase of the stresses in the cladding. Possible shapes of control rod wear marks can be seen in Figure 2.

Siemens recognized at an early stage that in the long term the coating of control rods is an effective means of improving wear resistance in plants exhibiting accelerated wear. Insertion of Cr_3C_2 coated control rods in specific „wear locations" has been proven as an excellent remedy without affecting other core components or reactor internals. It has been demonstrated that since insertion of the first Cr_3C_2 coated RCCA in 1979 up to today, no wear marks or spalling effects have occurred.

3 How does the RCCA management concept work?

In 1979, Siemens started a material test program to study the swelling behavior of various absorber materials under irradiation.

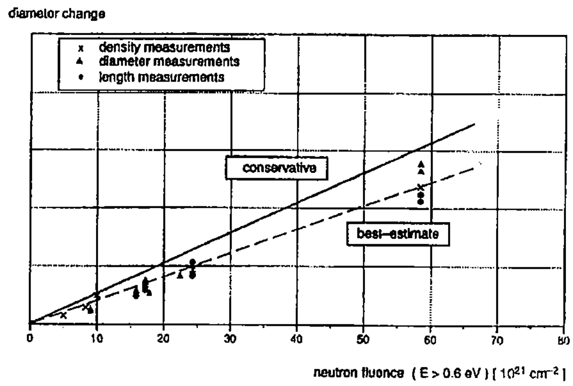


Fig. 3. Swelling of AgInCd

Samples with a length of 100 mm were assembled to produce test rods and then irradiated for up to eight cycles in the guide thimbles of a PWR fuel assembly. To date the specimens have attained a maximum fluence of almost 7 x 10^{22} cm^{-2} (E > 0,6 eV). Further measurements were performed on control rods.

Density, diameter and length measurements were performed. The results are shown in Figure 3. The measured data agree well with results of measurements, which are published in [1, 3,4]. It is apparent from Figure 3 that absorber material swelling is essentially isotropic and is a linear function of fluence.

For design purposes a swelling curve is used which covers the present measuring points with adequate statistical certainty.

Cladding creepdown

The individual segments of the material test rods were inspected and the diameter measured after each irradiation cycle.

These data serve as the basis for a design curve which describes the in-reactor creepdown of steel cladding tubes. Figure 4 shows the measured values as a function of fluence.

The design curve is specified such that it covers the measuring points with adequate statistical certainty

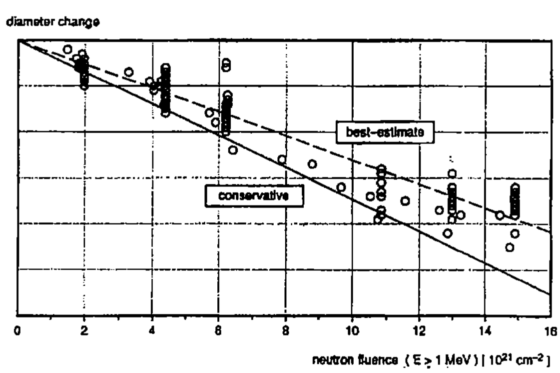


Fig. 4. Diameter changes of stainless steel cladding tubes

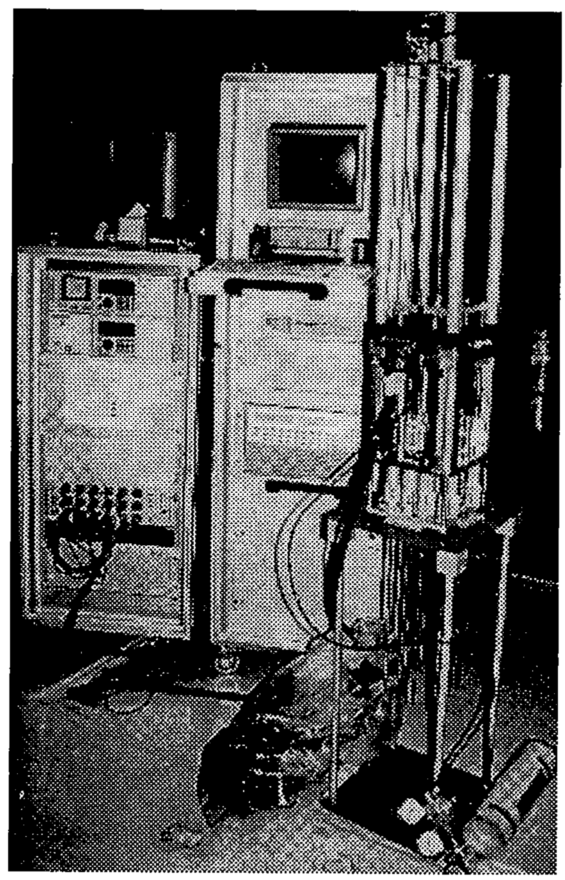


Fig. 5. Eddy current testing gauge together with calibration rig, control cabinet and data acquisition equipment

Wear

The evaluation of inspection results shows that in most cases wear on control rods may limit the residence times of control rods in reactors [6].

Wear results in a reduction of the cross-sectional area of the cladding tube and thus in an increase in cladding stresses. The maximum permissible wall thinning is reached when - in accordance with the design criteria [5] - the required margin to control rod fracture is reached.

a) Types of stress

The following stresses are considered:

- Stresses resulting from forces of inertia during RCCA stepping motion, i.e. compressive/tensile (membrane stresses), distributed uniformly over the cross section. They are described by the equation $\sigma = F / A$, where

σ: Membrane stress

F: Force

A: Cross section

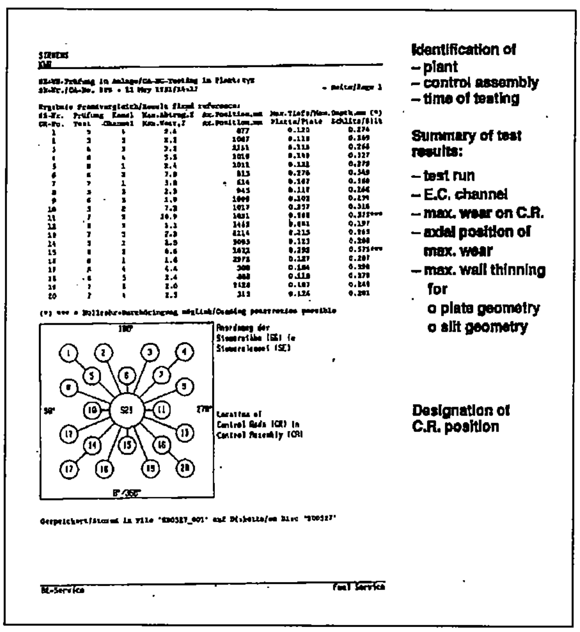


Fig. 6. Eddy current test results for an RCCA

- Stresses resulting from bending moments. Here the maximum rod bending and thus the maximum bending stresses are limited by the gap between cladding tube and guide thimble.

- Cyclic stresses
  Cyclic stresses occur as a result of bending vibration of the control rods.

b) Conservative assumptions

The possible shapes of wear marks, connected with the manner in which the control rods are guided in the guide assemblies, are as shown in Figure 2. Considering the maximum bending stress, the postulated one-sided, crescent-shaped wall thinning is conservative for calculation purposes.

Further conservative assumptions are:

- Minimum cladding cross section

- Maximum bending moment calculated from the maximum guide thimble inside diameter and minimum rod diameter

- Maximum RCCA acceleration during stepping motion

- Attack of force of inertia resulting from the entire control rod mass at the bottom end plug and thus at the location of maximum wall thinning.

Permissible control rod residence times

The possible residence time of a control rod is limited by the time it takes for the swelling-induced cladding strain to reach the permissible equivalent strain.

For calculation purposes, the fluence at gap closure is first determined using the design curves for absorber swelling and cladding creepdown and the fabrication data (e.g. filling gap). The permissible cladding strain is then calculated as follows:

After gap closure, the cladding tube is strained isotropically in the axial and tangential directions ($\varepsilon_t = \varepsilon_a$) as a result of absorber swelling, and from the equation for the permissible equivalent strain

$$\varepsilon_v = \frac{2}{\sqrt{3}}\sqrt{\varepsilon_t^{\;2} + \varepsilon_a^{\;2} + \varepsilon_t\varepsilon_a}$$

one obtains et = 0,5 x en for the permissible tangential cladding strain.

This is used to determine the permissible swelling expansion of the absorber and the associated permissible fluence.

Performing a conservative analysis (absorber swelling, clad creepdown and fabrication data) one typically obtains, for example, a permissible fluence (E > 0.6 eV) of approx. 4 x 10^{22} cm^{-2} for a control rod in a reactor loaded with 16 x 16 fuel assemblies.

For comparison, using best-estimate data results in a permissible fluence of approx. 7 x10^{22} cm^{-2}.

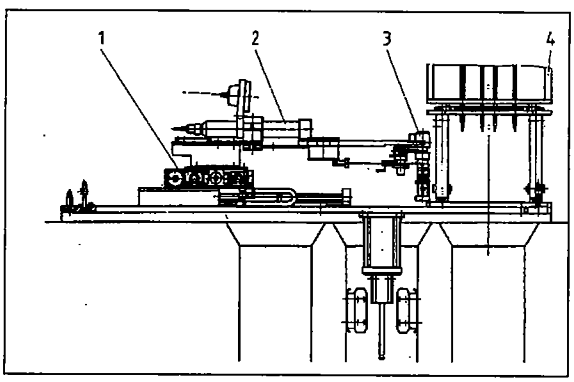


Fig. 7. Equipment for visual inspections of RCCAs. A profilometry gauge is attached onto the inspection equipment. (1) X-Y table, (2) underwater TV camera, (3) profilometry measuring gauge, (4) RCCA handling tool

The resultant permissible time and the permissible number of operating cycles cannot be fixed definitively.

The reason is that a specific fluence may be accumulated over different residence periods as a function of position, operating mode and insertion depth of control rods, which is in turn determined by the bank allocation.

Using the above - mentioned conservative assumptions, the maximum permissible cross section reduction due to wear is 23 to 31%, depending on control rod design (different mass of the control rods due to different control rod lengths). Here the stress category "membrane and bending stress" is limiting; the membrane stress still shows a large margin to the design limit. A permissible residual time can not be inferred directly. Knowing the permissible clad thinning, it is, however, possible to ensure reliable RCCA insertion by continuous RCCA monitoring in the form of eddy current testing/profilometry and/or visual inspection.

4. What equipment is used to check RCCAs?

An eddy-current test is carried out to determine precisely the maximum reduction in clad cross section, cracks and the swelling indications for every single control rod.

The testing gauge used is inserted into the spent fuel pool either onto the fuel storage racks or onto the new fuel elevator. The RCCA handling device is used to manipulate the RCCA by placing it into the testing gauge. The testing gauge is shown in Figure 5 together with the calibration rig, data acquisition equipment and control cabinet during check out, before starting the tests.

Data are processed by the computer and immediately printed out [7].

A typical example of such a printout is shown in Figure 6.

Visual inspection on the RCCAs can be carried out by means of a x, y table which is placed onto the storage rack to position an underwater TV camera relative to the RCCA being inspected.

The x, y table can also accommodate measuring devices for profilometry and diameter measurements on control rods.

Figure 7 shows the inspection equipment together with the profilometry gauge.

The profilometry and diameter gauges can be positioned such that every control rod finger is reached. Both gauges allow rotation in order to reach any azimuthal orientation on every control rod.

With longitudinal movement of the RCCAs any position over the entire length on all control rods can be reached.

Figure 8 shows a picture taken of the profilometry gauge during calibration.

Diameter measuring results of a control rod in combination with the result of a profilometry on the most effected location on the tested control rod are shown in Figure 9. The measured wall thinning at this position is approx. 12% of the cladding cross section.

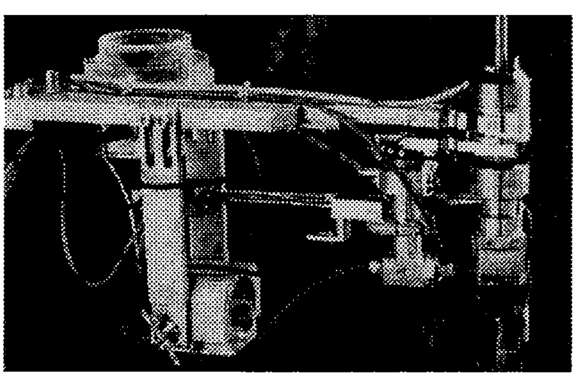


Fig. 8. Profilometry gauge for RCCA fingers during calibration

5. What are our conclusions?

The objective of optimizing RCCA insertion strategy is to minimize the total number of RCCAs used during the entire reactor lifetime.

Absorber clad interaction and wear should not be the reason to replace a RCCA without having attained a considerable fluence.

The extent to which an RCCA is approaching its design limit as a result of absorber swelling, and thus the end of its life, can be easily quantified.

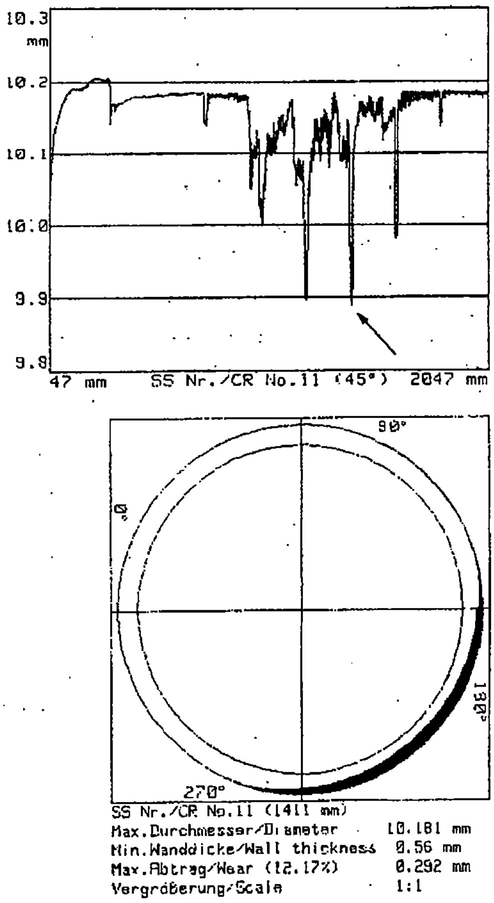


Fig. 9. Diameter measuring results from a wear effected control rod in combination with a profilometry result from the maximum wear mark on the same control rod.

The cumulative fluence is calculated in a RCCA management program as a function of

- insertion depth
- bank allocation (control or shutdown bank),
- in-core position,
- operating mode for the area of the tip of the absorber since this is the most highly stressed part of the control rod.

Variation of the operating modes of RCCAs, which takes fast neutron absorption into consideration minimizes absorber swelling effects.

Exact advance determination of the expected wear per cycle as a function of

- RCCA position in core,
- control rod position in RCCA,
- insertion depth

is not possible as operating experience to date has shown. The reason is that the excitation of vibrations is strongly dependent on the prevailing conditions, e.g., coolant flow, part and location tolerances.

RCCA inspections have therefore been found to be a useful tool for ensuring the integrity of these safety-related items. Wear marks very often show up only on a few control rods in specific reactor positions. Inspections carried out periodically on a certain number of RCCAs allow RCCA lifetime to be extended without reaching the limits of clad cross section reduction.

Based on the practice in Germany to test a quarter of all operated RCCAs per cycle, a RCCA management concept was established. This helps to ensure that RCCAs do not reach the clad thinning limit of up to 31% reduction in cladding cross section, nor limitation by interaction of cladding creepdown and absorber swelling during design lifetime.

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PWR RCCA Rodlet Performance for Cladding Tube Cracking caused by Absorber Swelling

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Abstract

The swelling of the rod tip due to neutron irradiation is one of the important degradation mechanisms of the control rods of a PWR (RCCA; Rod Cluster Control Assembly). In Japan, the diameter measurements of RCCAs have been carried out at several reactor sites in order to investigate the relation between the diameter increase of the rod tip and neutron fluence. During these inspections, an axial crack was found in a rod tip which had a high neutron exposure. Then, the cracked rodlet was examined in a hot-cell to identify the cause of crack. As a result of the examination, it is concluded that the crack in the rodlet tip was caused by a decrease in elongation of the cladding tube due to neutron irradiation and an increase in hoop strain caused by swelling of the absorber. The critical hoop strain for crack initiation is estimated to be approximately 0.7% from the hot-cell studies and this agrees well with the threshold in the diameter increase at which cracking was observed during site inspections. In addition, mechanical tests were performed on the absorber material using unirradiated specimens to investigate the contribution of mechanical deformation of the absorber on the diameter increase of the rod. Test results show that the contribution of mechanical deformation of the absorber is negligible. The crack in the tip of a rodlet is considered to have no impact on the performance and the integrity of the RCCA because the crack is longitudinal, and the absorber material is resistant to corrosion in the primary water environment of a PWR. In Japan, however, a management guideline has been established to prevent cracks based on the above studies, i.e., the RCCA is replaced when the neutron fluence at the rod tip exceeds the critical value, i.e., 0.8 x 10^{22}n/cm^2(E >l M eV) or 3 x 10^{22}n/cm^2(E > 0.625eV). For the purpose of further life extension of the RCCA, an improved RCCA in which the diameter of the absorber is reduced at the tip has been introduced. At this time, more than 700 improved RCCAs are in service and have experienced up to 5 operating cycles. The follow-up program on the improved RCCAs is being conducted for estimating their life time. Recently, diameter measurements were carried out on the improved RCCAs. The results show good performance of the improved RCCA, i.e., no obvious diameter increase was observed and the diameter increase seems smaller than that of a conventional RCCA although the neutron fluence is not so large at this time.

1. Introduction

Although the life of a PWR control rod (RCCA; Rod Cluster Control Assembly) is not limited by worth reduction because the RCCA is almost fully withdrawn from the core in operation, its life time is restricted due to aging degradation. The degradation of the RCCA occurs in two way, i.e., wear of the cladding tube and swelling at the rodlet tip. In Japan, PWR utilities and the manufacturer have made cooperative efforts to clarify these phenomena and establish countermeasures. Concerning the swelling of the rod tip, a management guideline based on neutron fluence and an improved design of RCCA intended for life extention have been introduced as a result of the reactor site and hot-cell examination studies, which were carried out to identify the mechanism of the diameter increase and the crack at the rod tip.[1,2,3]. Current activities and the present status concerning the degradation of the control rod tip due to neutron exposure are presented in this paper.

2. Operating Experience of RCCAs in Japan

The PWR control rods have a rod cluster structure with Ag-In-Cd alloy absorber material enclosed in stainless steel cladding tubes. About 900 RCCAs are in use at 23 PWR plants in Japan. There are three types of RCCA, i.e., 14 x 14, 15 x 15, and 17 x 17 types and they have 16, 20, and 24 rods, respectively. The typical structure of a RCCA is shown in Figure 1. Although RCCAs are kept almost fully withdrawn from the core during reactor operation, a small portion of the rod tip is inserted in the core. Thus, material degradation in the rod tip region due to neutron irradiation is important. As is known well, the absorber material, Ag-In-Cd alloy, swells due to neutron irradiation. To investigate the diameter increase of the rod tip due to absorber swelling, diameter measurements of control rod tips were carried out at several reactor sites. Figure 2 shows the relationship between the diameter change of a rod tip and its neutron fluence, showing that the diameter change increases with fluence. In these investigations, detailed visual inspections using a fiber scope have been performed at plant H to examine the integrity of the rod tips. The fiber scope inspection was performed on the RCCA with the largest neutron fluence and another RCCA with a relatively low neutron fluence as a reference. Table 1 summarizes the inspection results. An axial crack was found at the tips of three control rods of the RCCA with the largest fluence. The cracked rods had relatively large diameter increases of more than 80 μm. No cracks were observed on rods with diameter increases less than 80 μ m.

3. Hot-cell Examination and Crack Mechanism

3.1. Hot-cell Examination

To investigate the crack features and the cracking mechanism, segments were cut from the cracked rodlet and another intact rod at site, and were transferred to the hot laboratoty. The locations of the segments are shown in Figure 3. Two segments, A and B, were taken from the rod tip and the upper portion of the cracked rodlet which had the largest diameter increase. On the other hand, segment C was taken from the rod tip of the another intact rodlet with the smallest diameter increase. A photograph of the cracked rod tip taken in the hot-cell is shown in Figure 4 and the dye penetrant test results are also shown in Figure 5. As shown in these figures, the crack runs longitudinally from just above the weld line of the end plug to the location of the cut. The crack is single and has no branches. Figure 6 shows the typical metallography of the cracked region of the cladding tube. This indicates that the crack is fully intergranular. The profile of the diameter of the absorber of segment C agrees well with the profile of the outer diameter of the rodlet measured at site as shown in Figure 7. This shows that the diameter increase of the rodlet tip was caused by swelling of the absorber. Figure 8 shows the crack surface at the tip of the crack. It can be seen that the length of the crack at the inner surface is slightly larger than that at the outer surface. This suggests that the crack was initiated at the inner surface. Figure 9 is a SEN picture of the same surface, which shows that the original surface is fully intergranular, but both intergranular and transgranular surfaces can be recognized in the fracture surface formed when the crack was forced open in the hot-cell. Then, tensile tests were carried out on specimens of the cladding tubes. The shape of the test specimen is shown in Figure 10. The results are summarized in Table 2. The specimens from segment A have no irradiation effect because these were taken from the upper portion of the rodlet where the neutron fluence is negligibly small. Specimens with various neutron fluences could be taken from segments B and C by taking them at different axial positions because the neutron flux at the rod tip region changes rapidly. The tests were carried out at 320°C, and as a reference, tests at room temperature were also done for the segment A and during certification of an unirradiated specimen. Both the yield strength (0.2% offset) and the tensile strength of segments B and C, which had a high neutron fluence were greater than those of the unirradiated specimen. On the other hand, the uniform and the fracture strain were reduced compared to the unirradiated specimen. At 320°C, the yield strength was 800-910MPa, the tensile strength was 950-1050MPa, the fracture strain was 5.3-6.0%, and the uniform strain was 1.4-1.5%.

3.2 Crack Mechanism

As described above, an axial crack was observed in each of the three control rods with the largest diameter increase, and these belong to the RCCA which has the largest neutron fluence. The diameter increase of the cracked rodlets was more than 80 μ m as shown in Table 1, and 80 μ m corresponds to a 0.7% hoop strain in the cladding tube. This diameter increase, however, was observed under cold conditions. So, the strain under hot conditions during reactor operation is estimated to be approximately 1.0% by adding the differential thermal expansion of the absorber and the cladding, which is about 0.3%. Therefore, the critical strain for crack initiation of the irradiated cladding tube can be calculated by the following equation [2], which is derived from the fracture strain of a cylindrical pressure vessel subjected to internal pressure.

$$\varepsilon_{cr} = \varepsilon_{uc} + 1/2\ \varepsilon_{up} \qquad (1)$$

where, ε_{uc}: the elastic component of uniform strain
ε_{up}: the plastic component of uniform strain

ε_{uc} and ε_{up} of the irradiated cladding material are 0.6% and 0.8%, respectively, which were obtained from the tensile tests in the hot-cell mentioned above. By using these values, the critical strain for crack initiation is estimated to be 1.0% from equation (1). This agrees well with the threshold in the increase of the outer diameter at which cracks were observed during site inspections. The observed crack surface was fully intergranular as shown in Figure 6. It is considered that the fully intergranular crack can be attributed to the very low strain rate caused by absorber swelling.[4] In addition, a hydrogen analysis of the cladding was made, and the results are shown in Figure 11. The hydrogen content of the cladding at a neutron fluence of about 10^{14}n/cm^2 shows little variation and is almost the same as that of unirradiated cladding material. However, the variation in and the maximum value of the hydrogen content increase rapidly above a neutron fluence of 10^{21}n/cm^2. Thus, an increase of the hydrogen content in the cladding due to neutron irradiation is considered to be another cause of the intergranular cracking.

3.3 Mechanism of the Diameter Increase

It is thought that the diameter increase of the control rod tip is caused by the swelling of the absorber, which bulges the cladding from the inside. Figure 12 shows the relation between the absorber diameter change and the neutron fluence, and Figure 13 also shows the relation between the absorber density and the neutron fluence. It can be seen from these figures that both the absorber density and the diameter are strongly correlated with neutron fluence. As mentioned previously, the relationship between the outer diameter of the control rod measured at site and the neutron fluence shows a good correlation, These facts firmlv art the idea that the cause of the diameter increase of the control rod tip is swelling of the absorber due to neutron irradiation. However, mechanisms other than irradiation induced swelling , such as creep deformation of the absorber, were also investigated because the Ag-In-Cd alloy has quite a low mechanical strength. It seems that the temperature of the absorber tip region is higher than the coolant temperature because of heating. The temperature of the absorber tip was calculated by FEM (Finite Element Method). The FEM analysis model is shown in Figure 14. The absorber temperature depends on the gap between the cladding and the absorber. The calculated temperature distribution for the nominal gap condition and the zero gap condition are indicated in Figure 15 and 16, respectively. In the nominal gap analysis, the peak temperature location is about 50 mm above the rod tip and the peak temperature is approximately 400°C. In addition to the temperature analysis, tensile tests and creep tests were carried out using unirradiated absorber material in order to examine the mechanical properties of Ag-In-Cd alloy under hot conditions. The test specimens are shown in Figure 17. Table 3 summarizes the results of the tensile tests. Both the yield and tensile strength decrease as temperature increases, and the yield strength at 400°C is approximately 28MPa. Figure 18 shows the relation between the minimum creep rate and the applied stress. The loads acting on the absorber during reactor operation are the dynamic load induced by the movement of the control rod drive mechanism and the static load consisting of the absorber weight and the force of the hold-down spring. The dynamic load is estimated to be less than 10MPa. It is much smaller than the yield strength of the absorber material as shown in Table 3. The sum of the absorber weight and the hold-down spring force is about 1.5MPa, and it can be seen from Figure 18 that the corresponding creep rate is very low. In summary, it is concluded that the diameter increase of the control rod tip is caused by the swelling of the absorber due to neutron irradiation and the effect of mechanical deformation of the absorber material induced by dynamic loads and creep can be neglected.

4. Safety Evaluation

The fiber scope inspection and the subsequent hot-cell examination revealed that the crack in the rodlet tip was longitudinal, and could not lead to rupture of the rod. The absorber material, Ag-In-Cd alloy, is quite stable in the primary coolant environment of a PWR. Therefore, even if it is exposed to the coolant due to cracking, the integrity of the absorber will not be lost. Figure 19 shows a picture of the absorber taken from the cracked rodlet tip examined in the hot-cell, and it can be seen that there is no obvious corrosion. Thus, the integrity and the performance of a RCCA are not affected by a crack in the rodlet tip caused by an increase in diameter.

5. Countermeasures

5.1. Management Guideline

The diameter of the control rod tip can be predicted to increase linearly with neutron fluence as shown in Figure 20 based on the on-site diameter measurements and the hot-cell examination results. The neutron fluence at the point where the line of the predicted increase in diameter reaches the critical increase in diameter for crack initiation of 0.7% derived earlier is approximately 0.8 x 10^{22}n/cm^2 (E > lMeV) or 3x 10^{22}n/cm^2 (E>0.625eV). This can be regarded as a criterion for crack initiation in terms of neutron fluence, i.e., a critical neutron fluence. In Japan, a RCCA is replaced when its neutron fluence at the rod tip exceeds the above value.

5.2 Design Improvement

The increase in the diameter of the rod tip can be reduced by slightly decreasing the diameter of the absorber at its tip, i.e., increasing the gap between the cladding tube and the absorber, and so delaying the time at which they come in contact. At this time, more than 700 RCCAs of this type are in service and have experienced up to 5 operating cycles. The diameter reduction at the absorber tip is about 0.1 mm, and the life of the improved RCCA is expected to be approximately twice that of a conventional RCCA. The follow-up program on the improved RCCAs is being conducted for estimating their life time. Recently, on-site diameter measurements were performed on the improved RCCAs. The results are shown in Figure 21. No obvious diameter increase can be observed and the diameter increase seeems to be less than that of a conventional RCCA, although the neutron fluence is not so large at this time.

6. Conclusions

One of the life limiting phenomena of a RCCA is the increase in the diameter of the control rod tip due to neutron irradiation and the subsequent initiation of a crack in the cladding tube. be have made much effort to clarify the phenomena concerned with this problem and to establish countermeasures through on-site and hot-cell studies. The conclusions are as follows.

1) The diameters of the RCCA rodlets have been measured at several reactor sites and it is recognized that the diameter of the control rod tip increases with neutron fluence.

2) The RCCA with the largest neutron fluence was visually inspected with a fiberscope. As a result, an axial crack was found at the tips of the three control rods with the largest diameter increase, and the increase in diameter of the cracked rodlets was greater than 80 μ m, which corresponds to a change in diameter of 0.7%.

3) To investigate the crack mechanism, segment samples taken from the cracked rodlet and the another intact rodlet were transferred to the hot laboratory for detailed examination. As a result, it is concluded that the crack in the cladding tube of the control rod was caused by a decrease in elongation due to neutron irradiation and an increase in hoop strain due to the swelling of the absorber.

4) The critical strain of the cladding tube for crack initiation was estimated from the tensile tests to be approximately 0.7% in the cold condition, and this agrees well with the threshold for the increase in diameter at which cracks were observed in rodlets at site. The corresponding critical neutron fluence for crack initiation is also estimated to be approximately 0.8 x 10^{22}n/cm^2 (E>lMeV) or 3x 10^{22}n/cm^2 (E>0.625eV) based on the predicted increase in diameter of the rod tip with neutron fluence derived from the on-site and hot-cell examinations.

5) Tensile tests and creep tests using unirradiated absorber material were also carried out to investigate the effect of the mechanical deformation of Ag-In-Cd alloy on the diameter increase of the rod. The results suggest that the mechanical deformation of the absorber material caused by dynamic loads and the creep deformation due to static loads such as the absorber dead weight is negligibly low.

6) The crack in the control rod tip is longitudinal and so does not affect the mechanical integrity of the control rod. In addition, the absorber material, Ag-In-Cd alloy, is very stable in the primary coolant of a PWR. Therefore, it is thought that the occurrence of a crack will not affect the performance or the integrity of a RCCA.

7) In Japan, a management guideline has been established in order to prevent crack initiation at the control rod tip, i.e., the RCCA is replaced when the neutron fluence at the rod tip exceeds 3 x 10^{22}n/cm^2 (E > 0.625eV).

8) For the purpose of further life extension of the RCCA, improved RCCAs in which the diameter of the tip of the absorber has been slightly reduced have been adopted, and have experienced up to 5 operating cycles.

9) The follow-up program on the improved RCCAs is being conducted, and the diameters of the improved RCCAs have been measured recently. It is recognized that the increase in the diameter of the rod tip is much less than that of a conventional RCCA although the neutron fluence is not so large at present.

7. Acknowledgement

This study was performed as a cooperative research project of the Kansai Electric Power Co., Inc., the Hokkaido Electric Power Co.,Inc., the Shikoku Electric Power Co.,Inc.,the Kyushu Electric Power Co.,Inc., the Japan Atomic Power Co.,Inc.,and Mitsubishi Heavy Industries, LTD.

8. References

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RCAA	Neutron Fluence* at Rodlet Tip $\times 10^{22} n/cm^2$	Rodlet No.	Increase in Outer Diameter (μm)	Crack	Remark
R1	1.2	1	88	Yes	Specimen for PIE
		2	83	Yes	
		3	81	No	
		4	80	Yes	Min. diameter increase with crack
		5	76	No	
		6	70	No	
		7	68	No	
		8	67	No	
		9	62	No	
		10	62	No	
		11	59	No	
		12	57	No	
		13	53	No	
		14	49	No	
		15	48	No	
		16	40	No	Specimen for PIE
R2	0.7	1	46	No	
		2	30	No	
		3	19	No	
		4	16	No	
		5	16	No	
		6	15	No	
		7	10	No	
		8	9	No	
		9	6	No	
		10	3	No	
		11	3	No	
		12	2	No	
		13	-2	No	
		14	-4	No	
		15	-6	No	
		16	-13	No	

※E > 1.0MeV

Table 1. Relationships between increase in outer dismeter and cladding tube cracks

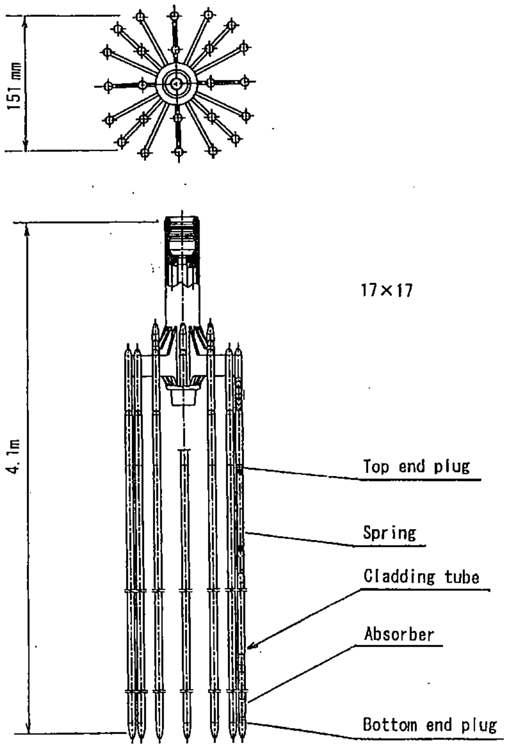
Segment	Test Condition: Test Temp. (℃)	Test Condition: Strain Rate (1/sec)	Yield Strength (MPa)	Tensile Strength (MPa)	Uniform Strain (%)	Fracture Strain (%)	Fluence ('n/cm^2 E>1MeV)
A	Room Temp.	6×10^{-5}	687	829	17	33	5×10^{13}
A	320	6×10^{-6}	613	734	6.3	11	2×10^{14}
B	320	6×10^{-6}	908	1032	1.4	5.9	9.0×10^{21}
B	320	6×10^{-6}	881	1046	1.5	6.0	1.1×10^{22}
C	320	6×10^{-6}	799	953	1.4	5.4	8.3×10^{21}
C	320	6×10^{-5}	868	1010	1.4	5.3	8.3×10^{21}
C	320	6×10^{-6}	868	981	1.4	5.4	1.1×10^{22}
Inspection Certificate	Room Temp.	–	616	745	–	38	–

Table 2. Results of tensile tests for irridiated cladding tubes

Table 3 Results of tensile tests for absorber material

Test Temp. (℃)	Yield Strength (MPa)	Tensile Strength (MPa)	Elongation (%)	Reduction of Area (%)	Young's Modulus (GPa)
Room Temp.	123	292	59.0	80.1	67.9
320	70	105	62.5	79.1	34.2
400	28	57	90.0	90.2	28.1
480	12	32	134.0	95.2	27.7

Table 3. Results of tensile tests for absorber material



		14×14	15×15	17×17
Total Length (m)		10ft 3.4 12ft 4.0	4.0	4.1
Number of Rodlets		16	20	24
Absorber Material		Ag-In-Cd		
Cladding Tube	Material	SUS 304		
	Outer Diameter (mm)	11	11	10
	Thickness (μm)	475	490	470

Figure 1. RCCA specifications

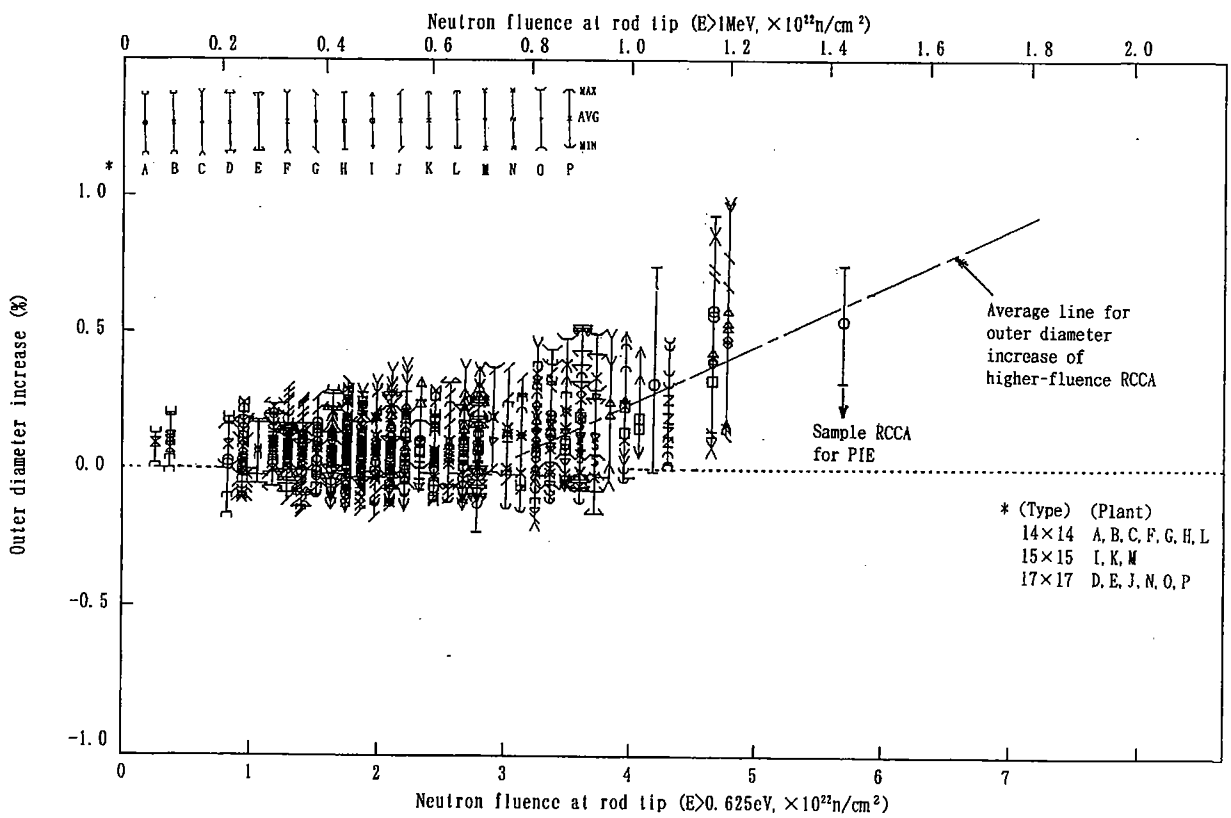


Figure 2. Outer diameter increase vs neutron fluence at rod tip

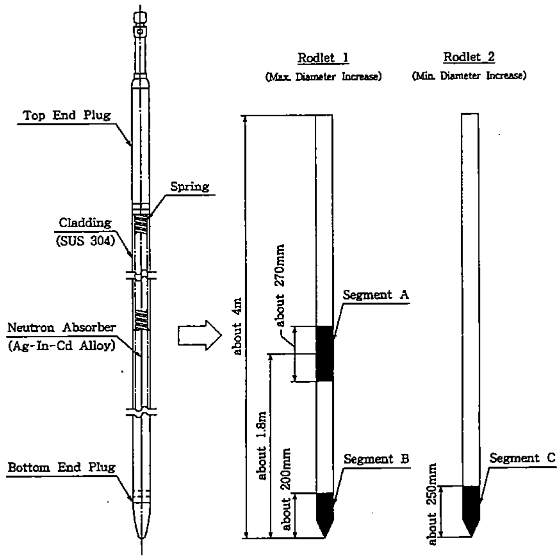


Figure 3. RCCA rodlet configuration and location of segments taken from rodlet

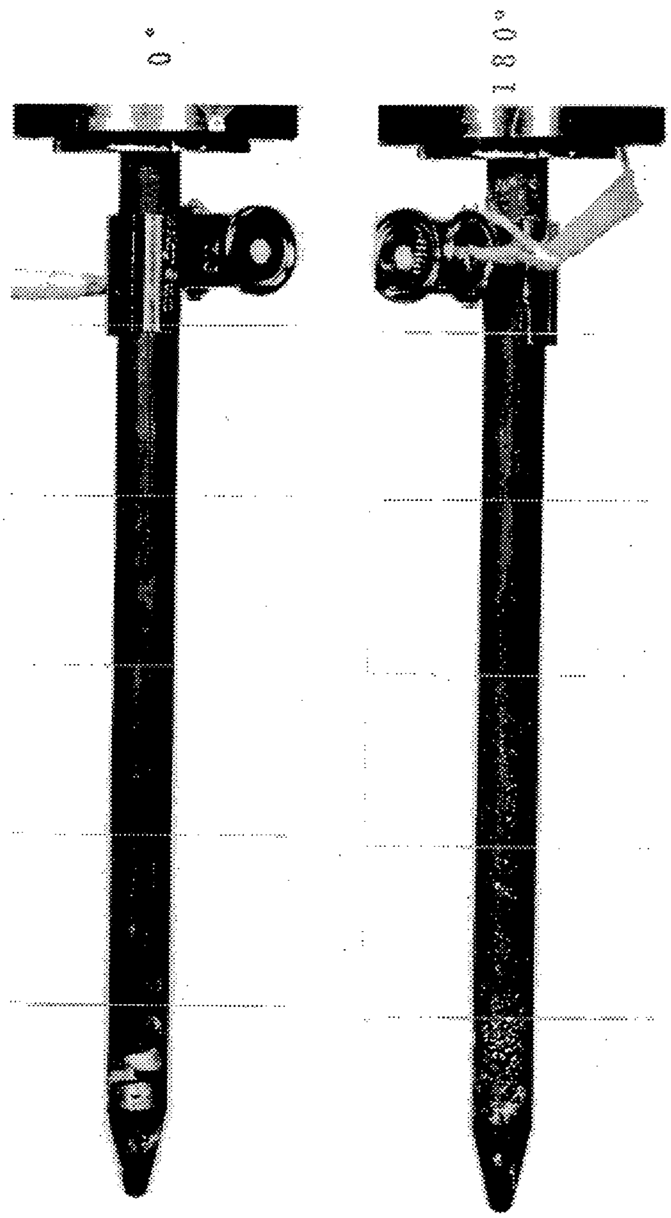
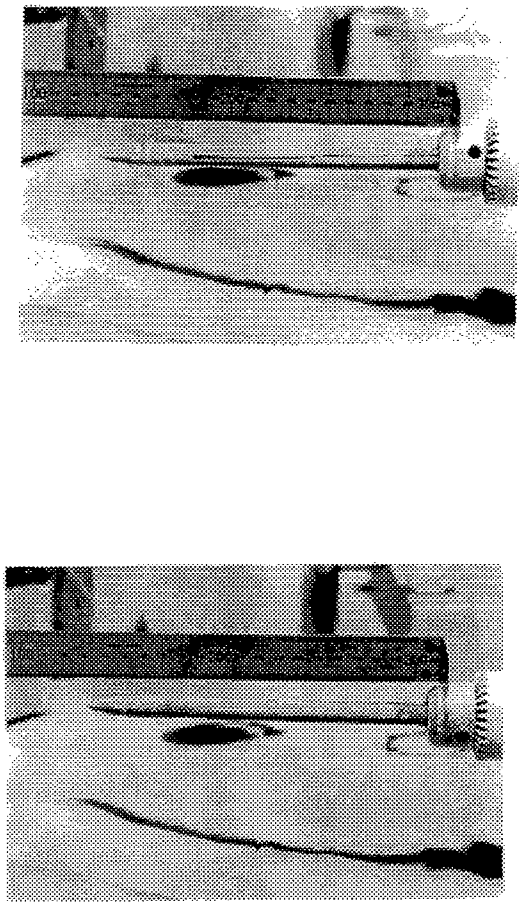


Figure 4. Cracked rodlet tip in the hot-cell



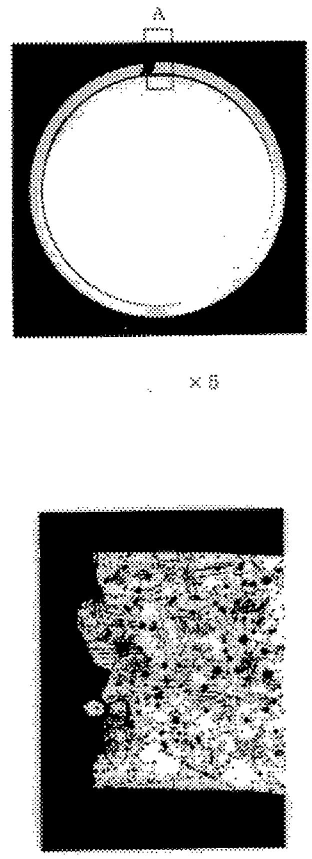


Figure 5. Dye penetrant test examination of cracked rodlet tip

Figure 6. Typical metallography of a cladding crack

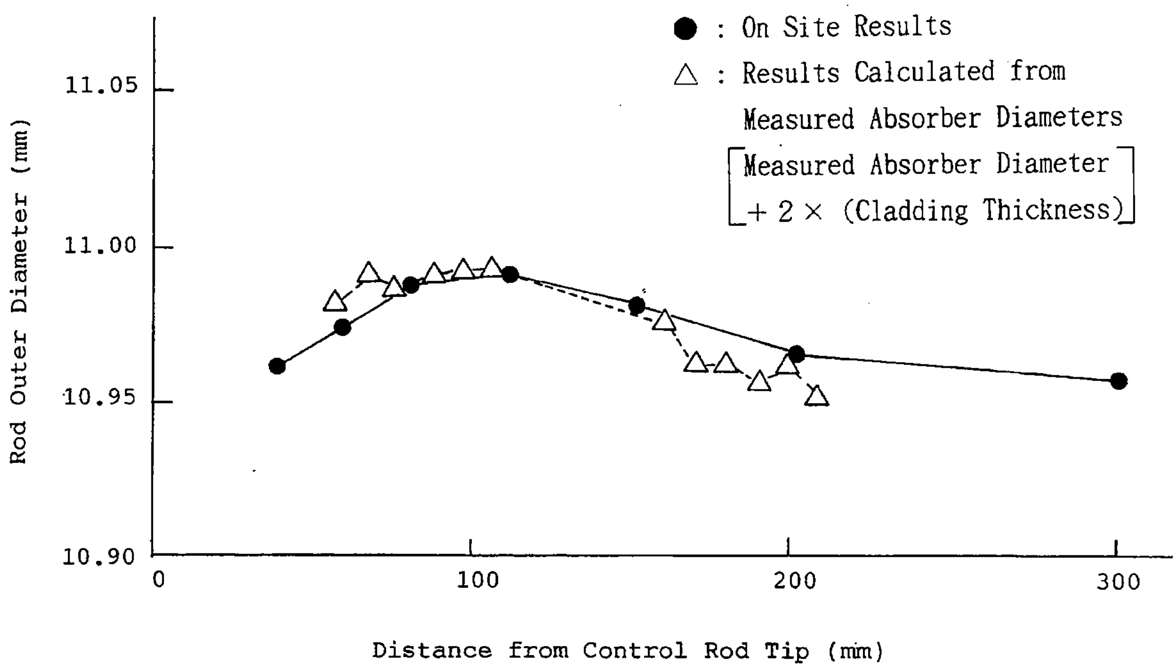


Figure 7. Comparison of rodlet outer diameters obtained by measurement of absorber diameters with those measured on-site (segment C)

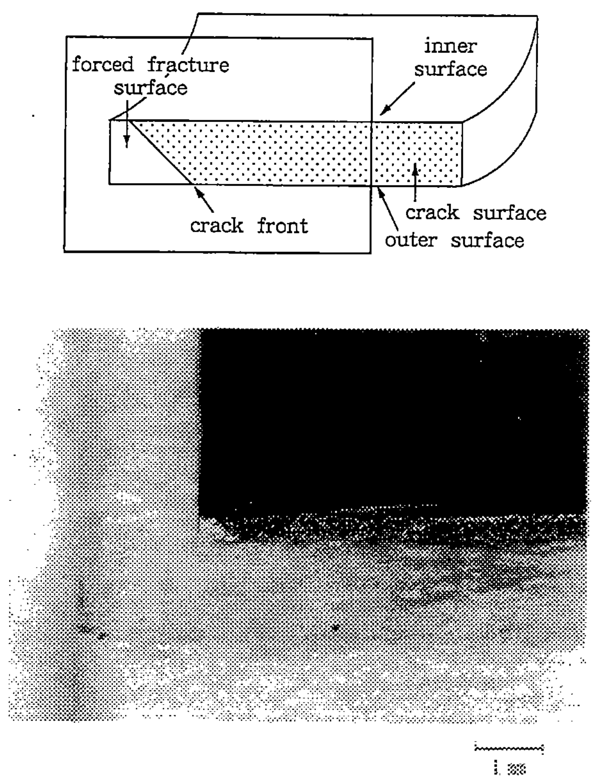


Figure 8. Crack surface of rodlet cladding tube (segment B)

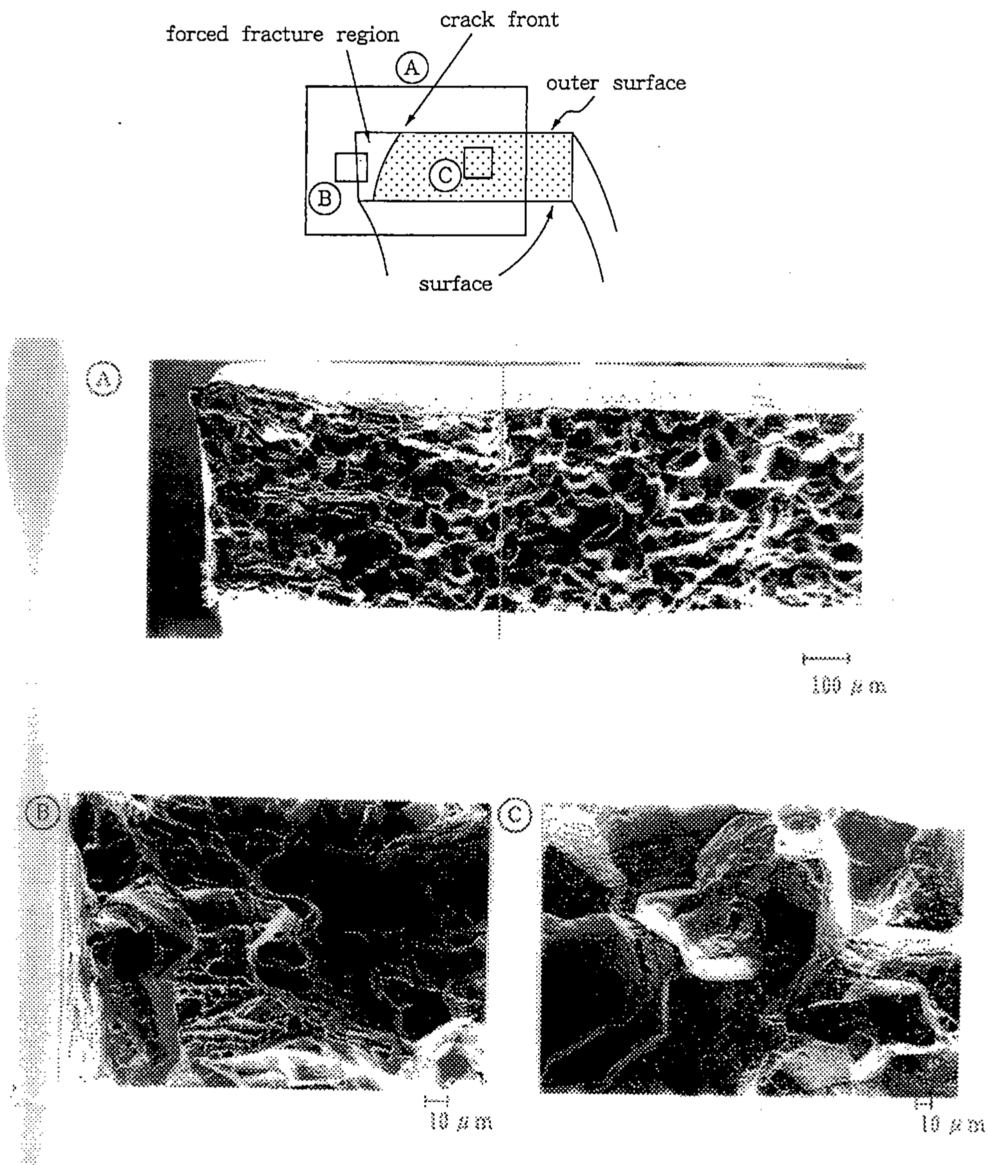
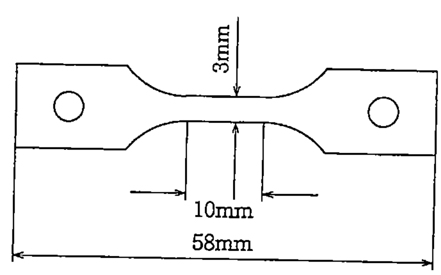


Figure 9. SEM investigation of the fracture surface of rodlet No. 1



Thickness : about 0.5mm

Figure 10. Tensile test specimen taken from segments, A, B, and C

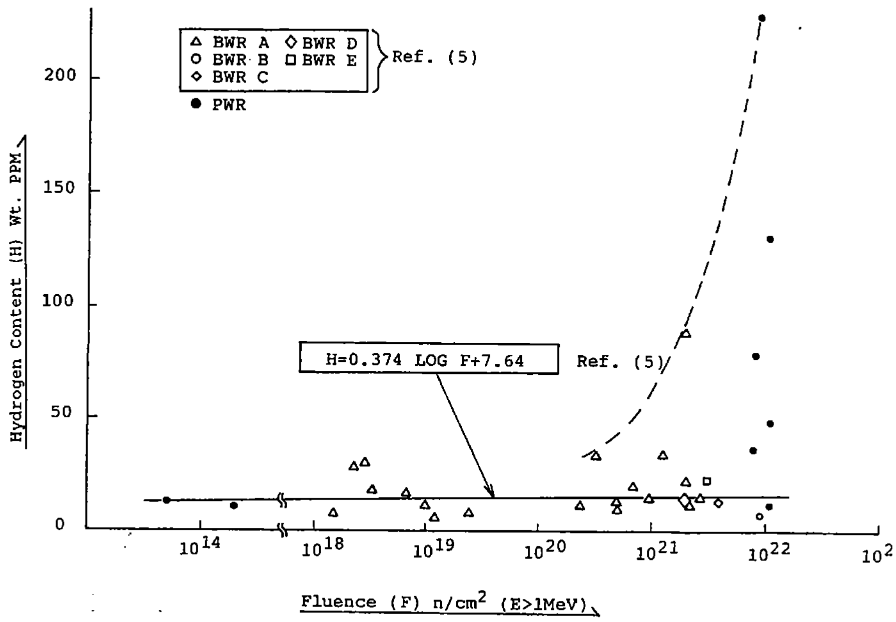


Figure 11. Hydrogen content of cladding tubes as a function of neutron fluence

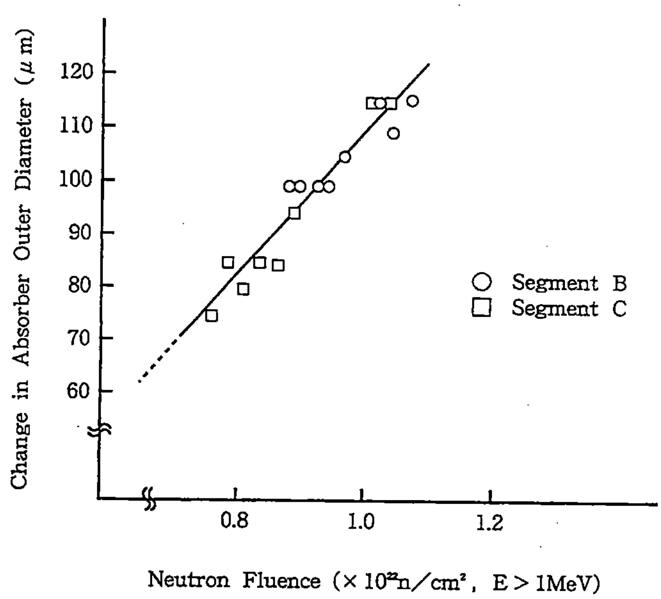


Figure 12. Relationship between increase in outer diameter of absorber and neutron fluence

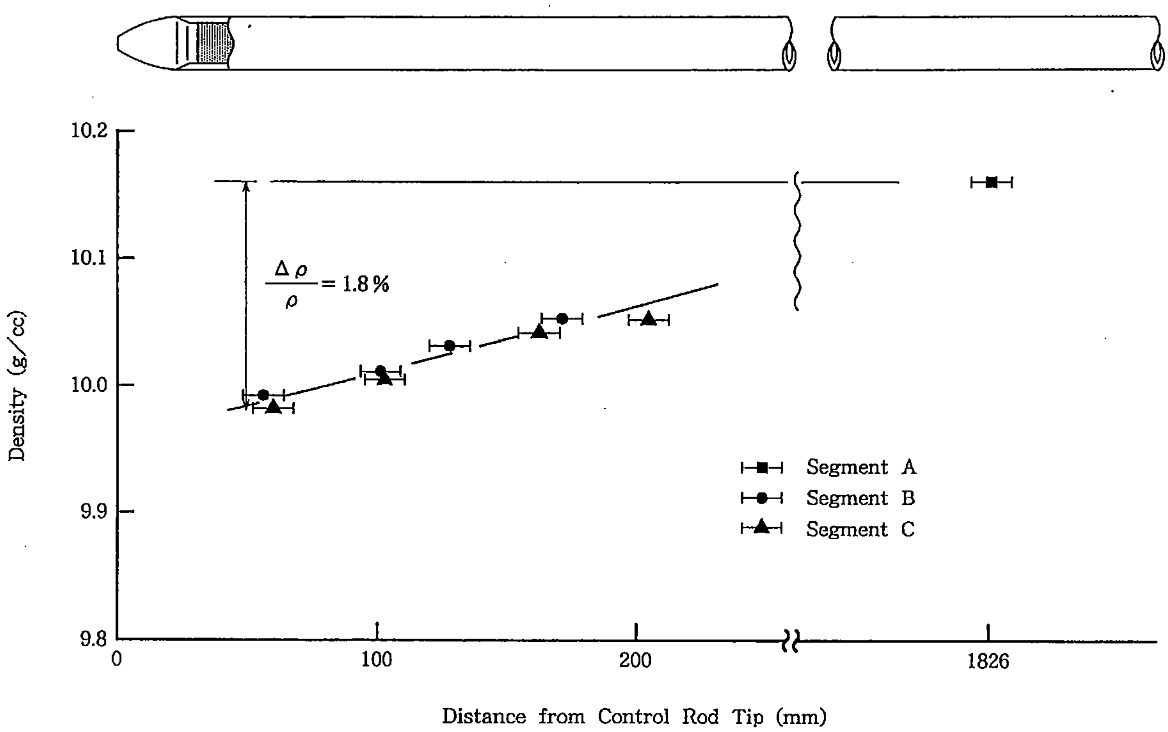


Figure 13. Density deviation of absorber tips

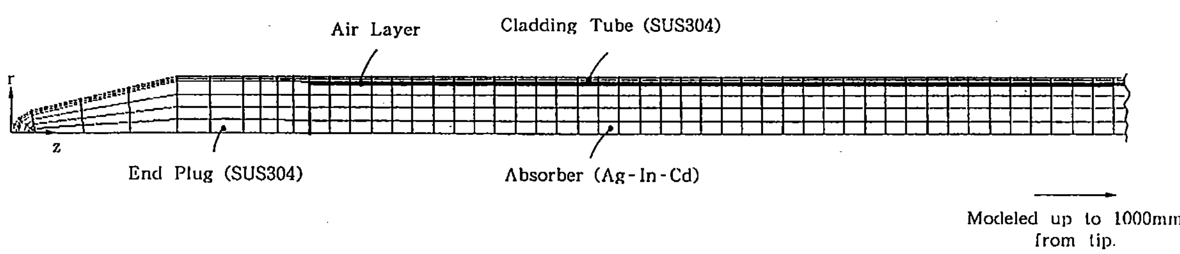


Figure 14. FEM analysis model for absorber temperature distribution

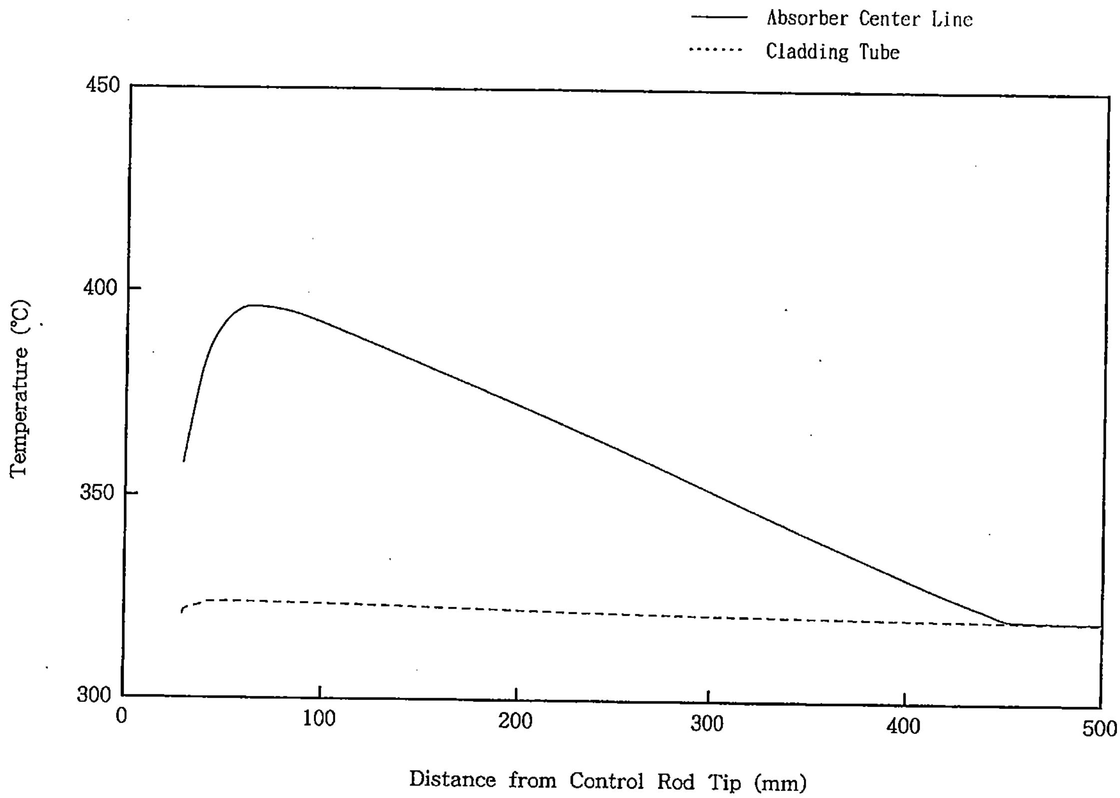


Figure 15. Analytical results of temperature distribution (Nominal Gap)

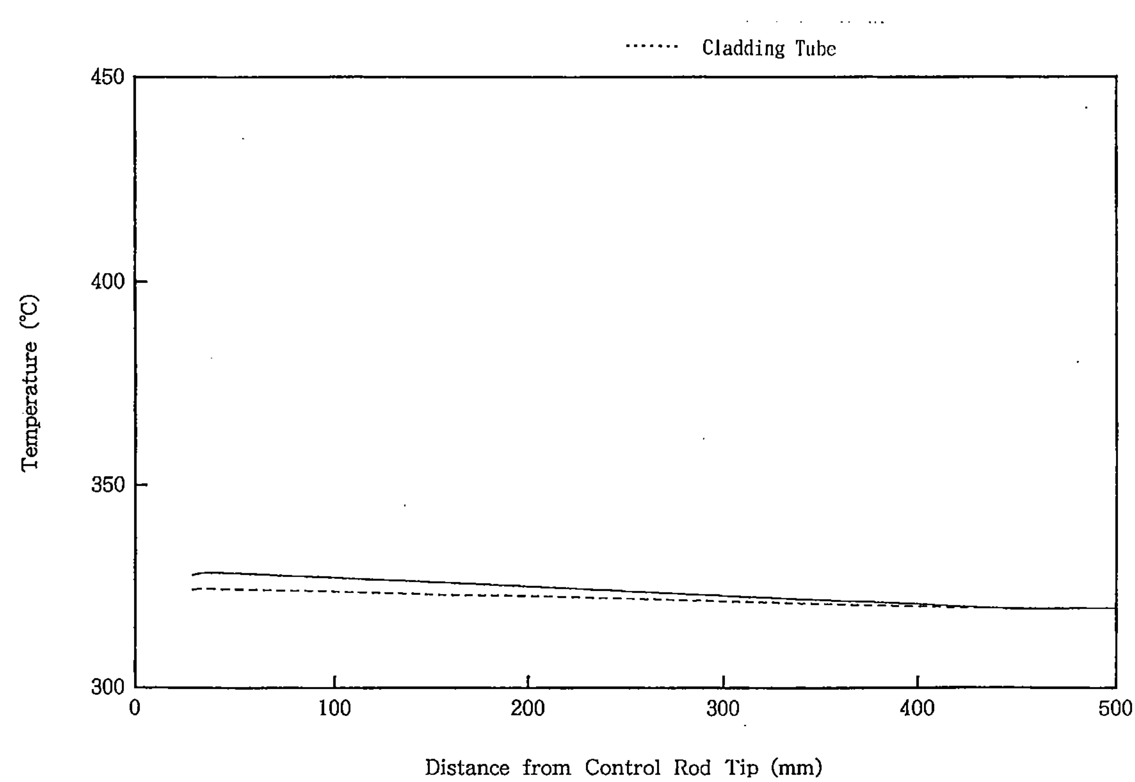
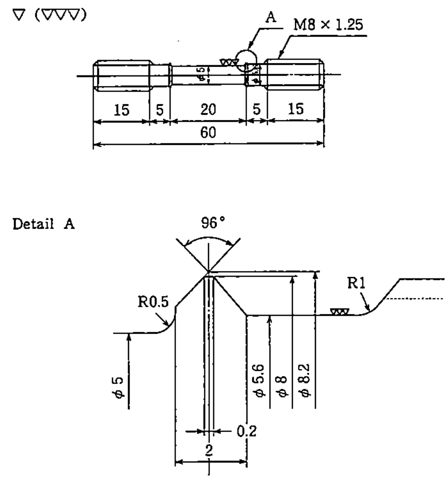
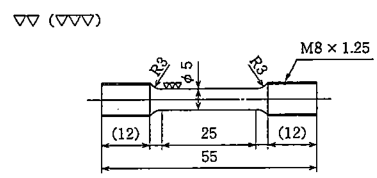


Figure 16. Analytical results of temperature distribution (0 Gap)



(1) Tensile Test Specimen



(2) Creep Test Specimen

Figure 17. Mechanical test specimens for absorber material

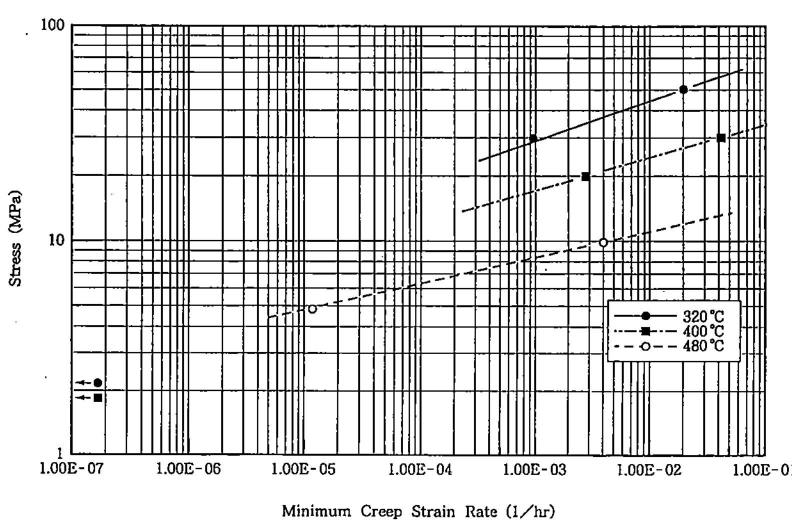
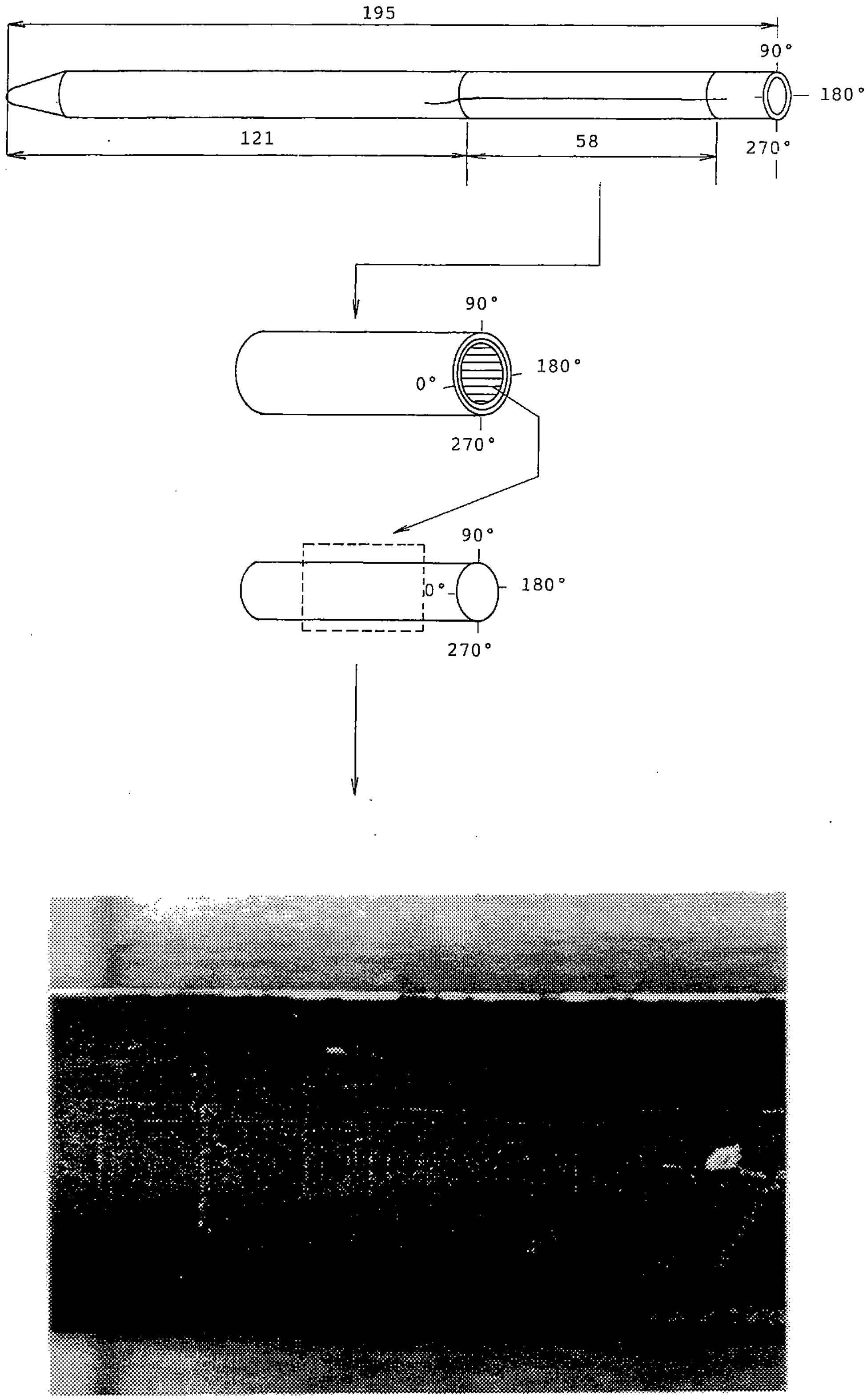


Figure 18. Creep test results for absorber material



×5

Figure 19. Absorber of cracked rodlet tip

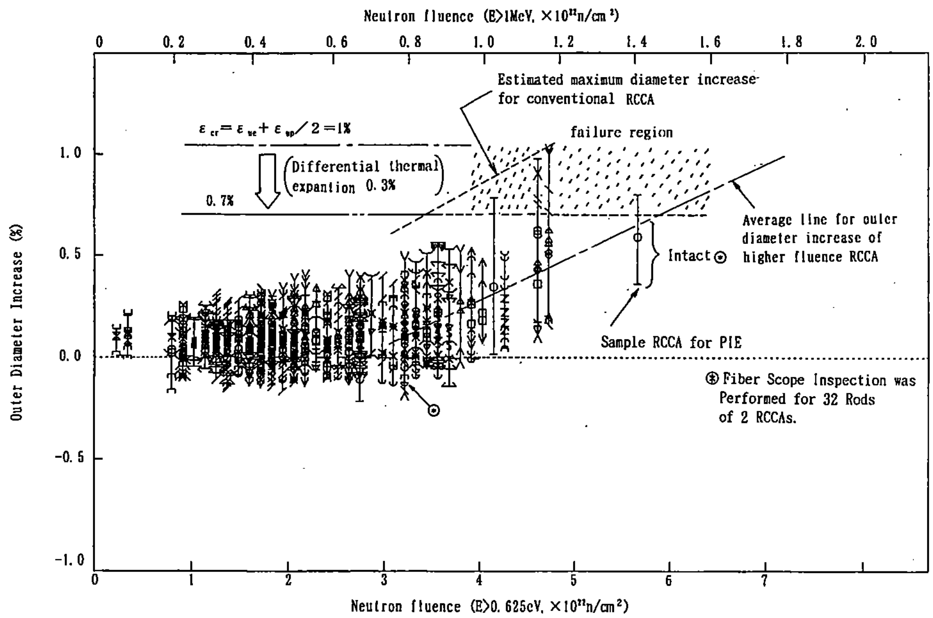


Figure 20. The mechanism of cracking and critical outer diameter increase of RCCA rodlet

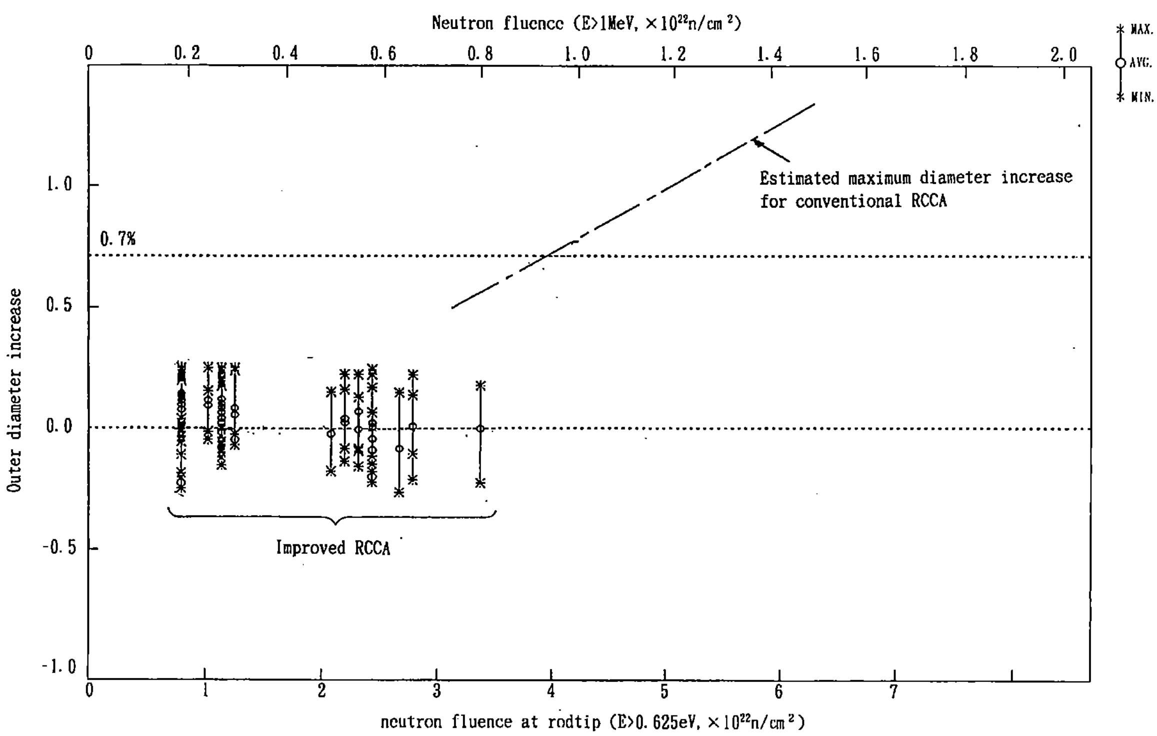


Figure 21. On-site diameter measurement results for the improved RCCA

Lesson Learned form Control Rods Irradiation Experience, Development of Advanced Absorbers and their Refractory Properties under Accident Conditions

V. Chernisov
V. Troyanov

Introduction

There are two generations of LPWR in Russia: WVVER-440 and WWER-1000 types. These reactors have different design of control members: 37 absorber assemblies in WWER-440 and multirod cluster system in WWER1000.

All Russian WWERs of the new generation are using the cluster absorber systems.

Therefore, design and operational experience of absorbing components (ACs) and control members of modern and advanced WWER-1000 designs are presented in this paper. Some results of ACs high temperature behavior for safety analysis under accident condition are presented too.

1. WWER-1000 - design and operational experience of absorbing components (ACs) and control members.

1.1. During the initial stage of design work, experiments were performed to determine the relative nuclear physics efficiency of different absorbing materials in the WWER-1000 power reactor environment. The measurements were carriedout using cladding specimens produced from X18HIOT steel, 8 mm OD, 0.3 mm wall thickness, with neutron absorber L = 50.0 ± 0.1 mm, OD = 7.0 ± 0.01 mm. Table 1 lists the most important results of this experiment.

1.2. The first reactor tests of short AC dummies (L = 100 mm, cladding OD 8.0 mm, wall thickness 0.5 mm, material - X20H40 alloy) were carried out in the SM-2 reactor (water pressure 190 kg/cm2, temperature = 350°C). The maximum thermal neutron fluence accumulated by AC dummies was $1.7x10^{21}$ n/cm^2).

The principle characteristics of the AC dummies are presented in Table 2.

1.3. Based upon the results of fi~rther reactors tests, performed in the WWER-2 reactor (NPP "Rheinsberg", Germany), the following AC types were tested in special clusters (10 ACs per cluster) - (see Table 3).

In addition to the tests run for "normal" sealed ACs, during the last stage of the experiments at NPP "Rheinsberg", the working capacity of dummy unsealed Acs was tested(L=100mm, hole dia. in cladding= 1 mm). Such specimens were tested for 5000 h autoclaves in a by-pass loop and directly in the reactor core.

Relative physical efficiency (ρ) of different absorbing materials (AM) under the conditions found in WWER-1000 reactors		
Type	Absorbing Material	ρ%
n, α	B_4C (pellets; natural; 1.8 g/cm^3) - standard	100
n, α	B_4C (extruded rod; natural; 1.93 g/cm^3)	$\approx$100
n, α	B_4C (pellets; natural; 2.4 g/cm^3)	104
n, α	B_4C (pellets; 40%B10;2.3 g/cm^3)	117
n, α	B_4C (pellets; 80% B10; 2.3 g/cm^3)	122.5
n, α	TiB_2; CrB_2 (pellets; natural; 4.5 g/cm^3)	96-98
n, α	B-alloy SBJ-2 (rods; 2%B_{nat} ; 8 g/cm^3)	71.5
n, γ	Al_2O_3 + Eu_2O_3 (1 g/cm^3) - extruded rod	72.5
n, γ	Al_2O_3 + Eu_2O_3 (1.5 g/cm^3) - extruded rod	78
n, γ	Al_2O_3 + Eu_2O_3 (2 g/cm^3) - extruded rod	85
n, γ	Eu_2O_3 (pellets 7.4 g/cm^3)	111.5
n, γ	Hf-Zr (5%) - rod	79
n, γ	Ni-In (10%) - Sm (10%) - Hf (10%) - rod	76
n, γ	Sm_2O_3 (pellets; 6.8 g/cm^3)	81
n, γ	$Dy_2O_3.TiO_2$ (pellets; 5.7-6.5 g/cm^3)	76-80
n, γ	Dy-metall (rod)	82
n, α n, γ	EuB_6 (pellets; 4.7 g/cm^3)	112.5

Table 1

In all cases water parameters were similar - P = 70 kg/cm2, temperature = 270°C, pH = 8.0-8.5. Irradiation does not increase the corrosion of absorbing materials significantly. The results obtained have demonstrated that the leaking ACs deliberately produced with the above mentioned absorbing materials (AMs) retain their working capacity in typical WWER operating environments.

Based upon the analysis of the results of investigation of irradiated ACs two types were chosen for WWER-1000 control members:

1. as-drawn ACs of an Al + Eu_2O_3 composite (2 g/cm^3) in X18HIOT steel cladding (initial OD 15 mm x 0.8 mm, as-drawn dimensions OD 8.2 mm x 0.6 mm). For Novo-Voronezh NPP to meet the AM efficiency requirement, the AM must be the equivalent of > 80% natural B_4C.

Basic characteristics of mock-up Neutron Absorbers (NA) elements irradiated in reactor SM-2 to the fluence $1.7x10^{21}$ n/cm^2 (thermal)

Neutron Absorber Type	Extruded Rod Type		Pelleted Type		
Absorber Material	B_4C	$Al+Eu_2O_3$	B_4C	$Dy_2O_3+.2TiO_2$	Ni-In-Sm-Hf
Overall density of AM (g/cm^3)	1.93	3.63	1.72	4.85	9.28
Density of B_{nat}, Eu,Dy etc. (g/cm^3)	B~1.53	Eu~1.85	B~1.35	Dy~2.93	Sm~0.4 In~ 0.44 Hf~0.9
Cladding diameter increase (sealed design)	~0.5	0	~ 1.5	0	0

Table 2

2. as-drawn ACs based B4C (1.9-2.0 g/cm^3) in X18HlOT steel cladding (initial OD 11.0 mm x 0.7 mm, as-drawn dimensions OD 8.2 mm x 0.6 mm) for the control of commercial WWER-1000s.

To avoid NPP unit contamination by radioactive isotopes Eu^{152} (half-life 13y) and Eu^{154} (half-life 16y) as the result of accidental rupture of at least one AC during repositioning of absorbing rods (ARs) of control and safety systems between refueling, the decision was taken to replace all Al+Eu2O3 ARs of control and safety systems with B_4C ARs though they have considerably less radiation stability (B_4C swelling and free helium formation due to $B^{10} + n \rightarrow He, Li$)

Based on the generalization of test results in reactors on the modified ACs and further investigation of irradiated specimens, the service life was defined as follows:

- 1 year in the automatic control regime,
- 5 year in the emergency protection regime.

1.4. Fig. 1 shows the design, dimensions and materials of a vibrocompacted AC.

1.5. Table 4 lists the basic results of ACs investigated after service life tests in the WWER-1000 reactor, Unit No 5 of Novo-Voronezh NPP.

As a result, it can be stated that in the process of AC modification, the following measures are expedient:

a) utilization of a cladding material with less radiation embrittlement under prolonged neutron irradiation;

b) utilization of an n,γ - neutron absorber (e.g. D_{y2} $0_3.TiO_2$ or Hf instead of B_4C in the bottom part of AC due to its considerably higher radiation stability (no out-gassing, high structural and volumetric stability).

The basic characteristics of NAs irradiated in reactor WWER-2, NPP "Rheinsberg"

Clad of NA: ∅ 8.2 x 0.6 1) steel X20H40 2) steel X18H10T Absorbing material: 1) B_4C_{nat} - pellets (~1.75 g/cm^3) 2) B_4C_{nat} - extruded rod (~1.9 g/cm^3) 3) $Dy_2O_3.2TiO_2$ - pellets (~5.4 g/cm^3) 4) Σ $R_2O_3.TiO_2$ - pellets (~6.0 g/cm^3) - Base: $Dy_2O_3.TiO_2$ 5) alloy Ni-In-Sm-Hf (In,Sm,Hf -10% of each) 6) composition Al+Eu_2O_3 (2 g/cm^3)
Irradiation conditions: 1) Time = 1-3 years; 2) The devices worked in water (Pressure = 10 Mpa ; Temperature 270°C), Φ^{max} = (2.2-9.1)x10^{20} n/cm^2 (thermal)
Main results of the investigations of irradiated NAs: 1) All the investigated NAs maintained their sealing; 2) None of the investigated NAs swelled, i.e. + Δd_{NA} = 0 3) Swelling of ($Dy_2O_3.2TiO_2$) pellets was not over 0.5% 4) Swelling (due to microcracks) of Σ $R_2O_3.TiO_2$ ($Dy_2O_3.2TiO_2$) pellets ≤ 2.3% 5) Burnup of B^{10} Maximal in pellets - 8.8% in extruded rod - 11.8% 6) Swelling of B_4C pellets: Δd_{am} = 1.35-3.5% 7) Reduction in plasticity of metal claddings of NAs: X20H40 : δ_o^{min} = 6.7% X18H10T: δ_o^{min} = 4.0%

Table 3

Table 4

The basic results of investigations of NAs in the hot cells of Novo-Voronezh NPP and Research Institute. NAs used in the Automatic power control subsystems of Unit 5 of Novo-Voronezh NPP (NAs- vibropack type, based on B_4C and $Dy_2O_3.TiO_2$)

Characteristics	NA based on B_4C		NA,based on $Dy_2O_3{\cdot}TiO_2$
	τ_{PC} = 2 years	τ_{PC} = 3 years	τ_{PC} = 3 years
Common condition of NA	Δd = 0; All rods are intact; there are no traces of scale or deposits of corrosion products; thickness of black film is 5-7 μm		
Max. Burnup, B^{10} (%)	~32,5	~42,6	Isotopic composition of Dy will be determined at 0-750mm distance from the bottom of the NA
Production of free He within NA (cm^3 NTP)	70-130	202-248	
Pressure of gas under clad of NA (atm) 20°C 350°C	 5,0-9,3 10,6-19,8	 14,4-17,7 30,5-37,6	
Condition of absorbing material	For mechanical properties of irradiated clad, NAs were cut into circular specimens. The powder of B_4C was cleaned from all specimens	At ~300 mm from the bottom of NA the B_4C was sintered; it was in close contact with the clad, but the stress in the metal was low and no creep was observed (Δd = 0).	see B_4C with τ_{PC} = 2 years

Table 4 (Continue)

Mechanical props.of clad most irradiated (bottom of NA): ultimate strength: σ_B (MPa) at 20°C σ_B (MPa) at 350°C	 802-840 446-640	 ~1020 ~680	
yield stress: at 20°C $\sigma_{0,2}$ (MPa) at 350°C	472-519 284-293		
uniform elongation, % at 20°C at 350°C	10,0-17,3 4,9-9,2	- 1.4-1.8	
Characteristics	NA based on B_4C		NA, based on $Dy_2O_3.TiO_2$ τ_{PC} = 3 years
Conclusions about NA condition	The clusters are far from their limits	The clusters are close to their limiting conditions	Clusters may be exploited in the PC-regime for an extra 1-2 years

1.6. During operational experience of B_4C-based ACs in commercial WWER1000 reactors at NPPs in Russia and the Ukraine no failure of any of these components during 95 reactor-years at 17 NPP units has yet been observed.

In 1993, on the basis of the above-mentioned data on B_4C-vibrocompacted ACs, the absorber assembly service life in the control and safety system (crossmember and 18 ACs) was increased to 2 years in the automatic control regime.

Work is in progress to validate the safe extension of the service life of these items to 6-8 years in the emergency protection regime.

2. Further improvement of WWER-1000 power reactor control members.

It is evident that it is necessary to use an n,γ - absorbing material instead of B_4C in. the bottom part of the AC which bears the highest radiation load. Thus, it becomes necessary to increase the AC diameter, otherwise it is impossible to provide for the required physical efficiency, because only Eu-based absorbing materials can compete with B4C in this aspect, but they are highly radioactive. Along with the requirements for high physical efficiency and radiation stability, n, - absorbers at the bottom of ACs should not produce ~ -active daughter isotopes whell interacting with neutrons. For AC cladding a structural material with higher radiation stability able to retain a plasticity (~0) of not less than 34% in the environment of high neutron irradiation doses should be used.

Fig.2 shows two variants of AC designs with n,γ -absorbers.

Compared to currently produced components, the new AC designs possess the following advantages:

- efficiency is 14-18% higher;
- reliability in operation, because the area under the heaviest irradiation contains structural and absorbing materials with a higher radiation stability;
- service life is equal to reactor service-life (about 30 years), including 2-3 years in automatic regime and the rest of the time in the emergency protection regime.

The basic factors which support the decisions taken are the following:

1) dysprosium titanate ($D_{y2}O_3.TiO_2$) compared to the In-Cd-Ag alloy commonly used in absorber assembly clusters has the following advantages:

   a) "burn-out" of dysprosium isotopes (see Fig.3) is slower than the "burn-out" of indium, cadmium and silver isotopes;

   b) it has no long-lived γ -active isotopes such as Ag-110;

   c) with equal physical efficiency, it has a rather high radiation stability and corrosion resistance. In addition, because of its lower density, heat generation is lower;

2) metallic hafnium has the advantage of being both an absorbing and structural material. Consequently, because of a greater effective diameter, one can provide for the efficiency of the

bottom part of the AC to be within 90-100% of the efficiency of the top part (boron carbide in cladding), despite the fact that:

$P_{Hf}/P_{B+C} = 0,8$

In addition the utilization of hafnium without cladding results in its temperature being practically equal to the temperature of the reactor primary circuit coolant and the strong oxide film formed on AC surface (5 -8 μm) makes a good barrier against hydrogenation of the metal proper;

3) advanced cladding material high-chromium nickel-based alloy XHMI has a high mechanical strength which allows for a smaller AC cladding thickness and makes it possible to increase the absorber diameter, i.e. to increase the efficiency compared to X18HIOT or 304 steels.

The high resistivity ofthis alloy to radiation embrittlement increases the life expectancy of an AC and absorbing assembly to between 15 (minimum) and 30 (maximum) years.

3. Basic results of investigations of the properties of Hf, $D_{y2}O_3.TiO_3$ and XHM1 alloy.

Below, the main results of the work which supports the utilization of these materials in reactor are presented.

3.1. Hafnium.

3.1.1. The relative physical efficiency of hafnium is equal to 80-85% of B4C (1.8 g/cm^3) with the natural isotopic content of B^{10}.

3.1.2. Radiation stability of hafnium was investigated using specimens irradiated in SM-2 and BOR-60 reactors up to fluences of:

- thermal neutrons: 7.2×10^{22} n/cm^2,
- with $E \geq 0.1$ MeV: 2.4×10^{22} n/cm^2.

These investigations of irradiated specimens and AC dummies have demonstrated:

- Hf has high structural and volumetric stabilities ($+\Delta V \leq 0.6\%$);
- an oxide film (5-8 llm) protects hafnium against hydrogenation by atomic hydrogen formed by the radiolysis of primary circuit water:

  $HfH_2 < 3 \times 10^{-2}\%$
- mechanical characteristics of hafnium are quite good:

σ_u and σ_y increase by factors of 1.5-2 while plasticity retains a reasonably high level (δ_0 = 2-4%).

3.1.3. The corrosion resistance of pure metallic hafnium in conditions corresponding to WWER-1000 primary circuit water after 15000 h testing is found to be very high approaching the "absolute resistance" class. Under irradiation hafnium retains its high corrosion resistance: after 26800h a dense oxide film is formed (5-20 μm). The corrosion rate after the first 4000 h equals zero, and + Δm does not exceed 0.8 %.

3.2. D_y2O_3. TiO_2 (dysprosium titanate).

3.2.1. The relative physical efficiency (compared to B_4C_{nat},t; 1.8 g/cm^3) in a WWER-1000 reactor environment is equal to 76-83% (for Dy within the range 2.5-5 g/cm).

3.2.2. Dysprosium titanate has been commercially produced for several years already (it is used as an absorbing material in the "stripe" ACs in the latest modification of the rods in the automatic control and emergency protection systems of RBMK-1000 and RBMK-1500 reactors).

3.2.3. The high radiation stability of dysprosium titanate is supported by:

– tests and further investigation of spent ACs (after 3 years of operation in the automatic control regime at Unit No.5 of Novo-Voronezh NPP);

– test and investigations of irradiated pellets in the SM-2 and BOR-60 test reactors ($\phi E \geq o$ 2.4×10^{22} n/cm^2): + $\Delta d_{AM} < 1\%$ (due to thermocyclic cracking);

- - ΔV of the grid ~ 2%; structure and phase composition are practically unchanged.

3.2.4. The corrosion resistance of AC dummies in a WWER-1000 primary circuit water environment during 10000 h testing is rather high.

$+\Delta m < 1\%$;
$+\Delta d_{AC}$ - 0

3.3. HNM1 alloy.

3.3.1. Chemical composition:

Chromium 42% (by weight);
Molybdenum 1 %;
Nicke the remainder.

3.3.2. Mechanical properties of pre-irradiated and irradiated material.

Test temperature - 350°C
Neutron fluence - 3×10^{22} n/cm^2 (E > 0.1 MeV)

	Initial cladding material	Irradiated claddings
Ultimate strength, MPa	702 ± 57	660
Yield stress, MPa	388 ± 42	540
Uniform elongation, %	41.5 ± 5	20

4. AC behaviour under ab-normal and accident conditions

4.1. Introduction

To substantiate the safety of nuclear power plant under ab-normal and accident conditions the number of high temperature tests has been carried out. The goal of these experiments is to study a high-temperature behaviour of control rods under transient conditions. Not only single AC dummies are tested, but dummies jointly to cluster channels and spacer grid are tested too. Experiments are carried out both in high temperature water steam loop and in vacuum installations. Thus, water steam presence influence is estimated. Besides, helium generation was simulated by additional internal gas pressure in the some boron carbide dummies.

4.2. Task order and test description

Loss of Cooling Accidents (LOCA) is accompanied by rise to temperature for the core structures. The design of reactor units and safety systems prevent to severe damage of the core, the limit of temperature increasing is not more than 1150°C and only few part of fuel rods may becomes unsealed.

In concern to AC's behaviour the question is to understand their ballooning, component's chemical interaction and ability to move AC insideclaster channel during accident and after it will be happen.

At the stage of the Severe Accident and Core Severe Damage we would like to understand the ultimate reserves of AC structures and to study the conditions of AM-cladding interaction, liquification, interaction between AC melts and another core structures. Different AMs under accident conditions should be compared.

The AC dummies have been made in Moscow Polimetal Plant and in IPPE. HNM-l a Hoy is used for dummy claddings, dia 8.6 x 0.6 mm. AM are boron carbide (1.6856 g/cm^3 density), disprosium titanate (5.1680 g/cm^3 density), metallic ganium. The numbers of the each AM kinds are 31. The lengths of dllmmies are 100 mm. Both sealed and unsealed dummies have been tested.

The long-term tests (about 15.000h) under operating temperature are the arststage of the study. Some structure changes and few chemical interaction between AM and cladding are observed.

The second stage is the high temperature tests within the steam loop "Gamma". Steam pressure is 0.1...0.5 Mpa. Maximum test temperatures for the comparative estimations are 900, 1050, 1150, 1200°C in different experiments. Specimens were examined and weighted. Diameters were measured, metallographic analysis was conducted, mechanical tests were carried out, X-ray phase analysis was conducted to study phase composition in the interaction areas.

4.3. Test results

4.3.1. B_4C AC

Interaction between B_4C powder and HNMI cladding is showed as cladding carbizing at 700°C and higher. If test temperature is 900°C and higher and duration of the test is more than 15 minutes, the complete cladding depth is carbized. In this case an embrittlement of cladding material take place.

At the 1200°C temperature test the total AC destroying is happen, liquid eutectics (Fe, Ni)-[(Fe, Ni)$_2$B, (Fe, Ni) $_2$B, Ni_3B_2] is appeared. Steam presence stimulates this process because of B_4C oxidation. The example of tested dummy is shown on fig.4. No liquification is observed under the temperature levels below 1200°C. At the same time the ballooning of pressurised B_4C dummies take place at the temperature 900°C and higher. It lead to the AC sticking inside cluster channel. So, the relocation of control rods will be impossible after LOCA will happen. The opportunity to put AC into operation is depends on the duration of process and temperature rise. Normally under LOCA reactor must shut down during 4 seconds. This duration is not dangerous for AC operating ability.

4.3.2. Dy_2O_3 TiO_2 AC

The ab-normal behaviour of $D_{y2}O_3$ TiO_2 AC is more optimistical. No liquification is observed under test temperature below 1400°C. Experiments under higher temperature didn't conducted. No significant chemical interaction including water-AM interaction are observed. Only some cladding oxidation take place.

From this point of view the $D_{y2}O_3$ TiO_2 AM are most preferable.

In the fig.5 the $D_{y2}0_3TiO_2$ dummies in compare with Hf dummy after 1250°C 0.5 hour test are shown. There were no superfluous intemal pressure in these dummies.

4.3.3. Hf AC

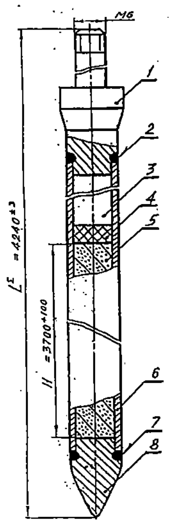
The liquification of Hf AC is happen under test temperature about 1250°C. The intermetallic phase $HfNi_2$ is observed after X-ray phase analysis. The extemal view of Hf dummy is shown on the fig.5.

4.4. Conclusion

The high temperature tests of different AC are conducted. From the point of view of the high temperature behaviour the $D_{y2}O_3$ TiO_2 AC are most preferable ones. The liquification of these types of AC starts from 1450°C or higher.

The liquification of B_4C AC is observed at 1150°C. The damages of B_4C ACS are catastrofical ones at this temperature.

The ballooning of each type of AC is take place at the test temperature 900°C and higher, but the ballooning of B_4C AC after long-term irradiation will be much more becouse of He generation. The ballooning will make difficult the control rod relocations, but not at once after LOCA will happen but only after some high-temperature exposure.



1. End cap
2. Upper weld
3. Compensation volume
4. Plug
5. B4C absorbing material
6. Vibropacked =1.75 g.cm^{-3} Φ=8.2 (mm)
7. Lower weld
8. Tip

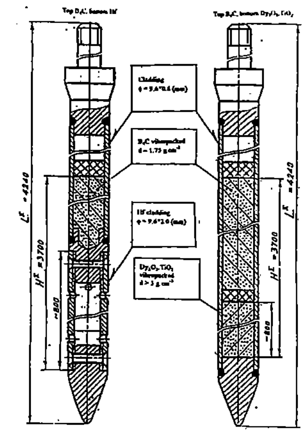


Figure 1. Two advanced variants of AC designs with n,y-absorbes

Figure 1. Design of vibrocompacted AC

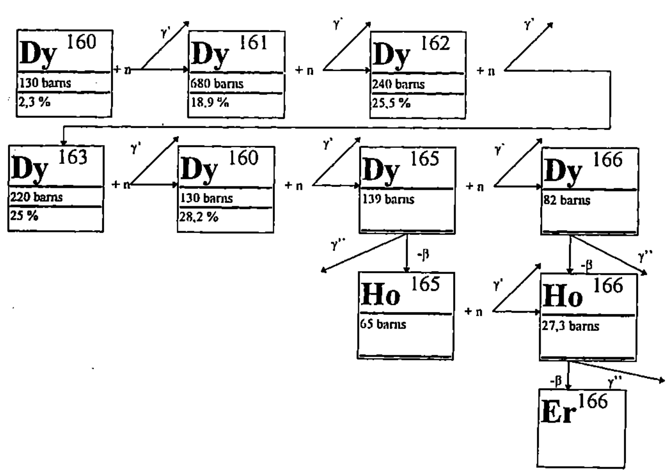


Figure 3. Isotopic conversions of Dy 16 absorption resonances of Dy can be found in the energy interval from 1 to 100 e V.

N B $T_{1/2}$ (Dy^{165}, Dy^{166})<<$T_{1/2}$ (Eu^{152}, Eu^{154})

Fig. 4. Tested dummy of B_4C AC.
Cladding material - X18H10T steel
Test temperature - 1200°C.

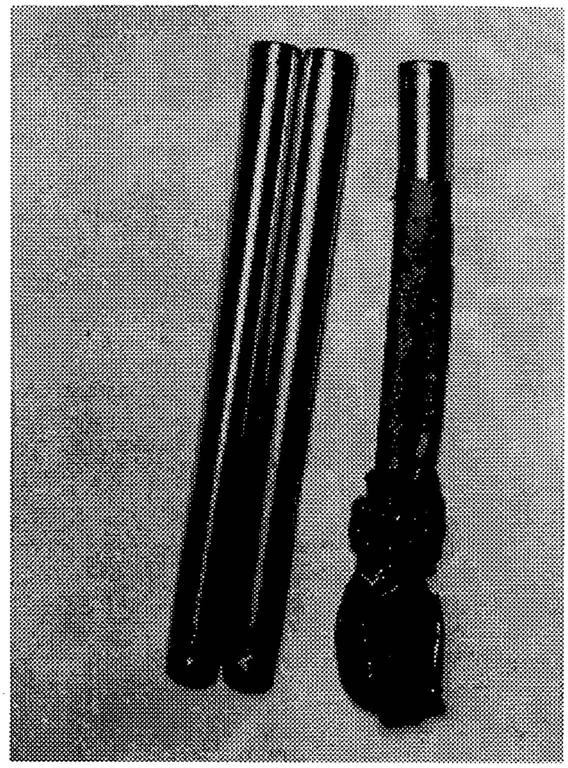


Fig. 5. Tested dummies of $Dy_2O_3 \cdot TiO_2$ (a) and Hf (b) Acs.
Test temperature - 1250°C
Cladding material - HNM1 alloy.

Incomplete Control Rod Insertion due to Extreme Fuel Element Bow

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E. Francillon
Framatome

The last two years incidents with control rods (RCCA) sticking in the lower part of the fuel assemblies (F.A.) have been reported of several reactor operators and fuel vendors throughout the world. The first event of this type occurred in Ringhals Nuclear Power Plant, Sweden, where the utility, VATTENFALL, and the fuel vendor, FRAGEMA, had to face a completely new problem.

The aim of this paper is to explain the way VATTENFALL and FRAMATOME/ FRAGEMA have managed this complex problem. It also outlines both partners´ present understanding of the phenomenon.

Introduction

During a reactor trip in Ringhals unit 4, the 22nd of August 1994, one RCCA stuck at 18 steps, i.e. in the dash-pot region of the guide thimble tubes. At a subsequent hot rod drop test four RCCAs stuck at the elevations 6, 12, 12, and 24 steps. As these occurred just three days ahead of the scheduled shutdown for the refuelling outage it was decided to cool down immediately and start an investigation of the sticking problems.

Ringhals 3 and 4 - General Information

Ringhals unit 3 and 4 are 3-loops, 960 MW(e), Westinghouse PWR. The units have been in commercial operation since 1980 and 1983 respectively. They are both operated in 12 month fuel cycle. The maximum burn-up usually reached is lower than 45000 MWd/MTU.

The cores of the two units consist of 157 F.A.s each with a rod array of 17x17. The active fuel length is 12 feet (= 228 RCCA steps) and the extension of the dash pot is 0.5 m (= 32 RCCA steps). The cladding and guide thimble tubes are made of improved Zircaloy-4. In 1994 more than 90% of F.A.s in the cores were supplied by FRAGEMA and the assemblies were of the AFA 2G design.

Preliminary Analysis

An extensive investigation program was initiated to cover all conceivable causes to the sticking problem.

Based on the behaviour of the rods during the rod drop test, visual inspections of the F.A.s, and previous industry experience the following causes were judged to be feasible:

- Restrictions in the guide thimble tubes due to debris, hydride concentrations and/or corrosion products
- Distortion/damage to the upper and/or lower internals
- Excessive swelling of the control rod tips
- Fuel element bow

Crud samples were taken from the inside of some tubes and measurements of the oxide thickness were accomplished with an EC-inspection of the guide tubes of several F.A.s. Furthermore, laboratory examination was made on two guide tubes, taken from the F.A. in which the RCCA stuck during the reactor trip. All investigations showed that the inner surfaces and the metallography of the tubes were normal. Thus, it was concluded that the problem could not be explained by the presence of debris, abnormal amounts of corrosion products, or excessive hydriding.

A TV-inspection confirmed that the upper and lower internals were in their normal positions and did not appear to have been subject to any damage or distortion.

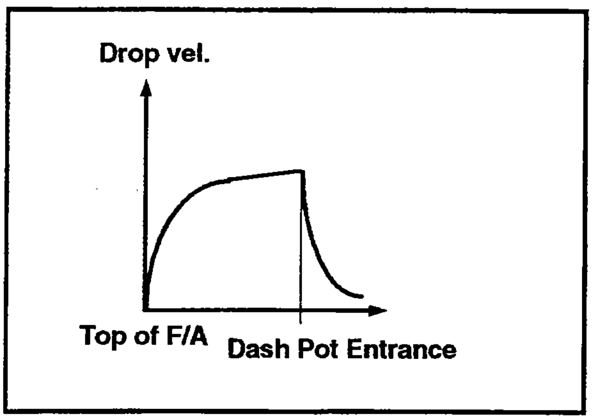


Figure 1a: Normal rod drop behaviour

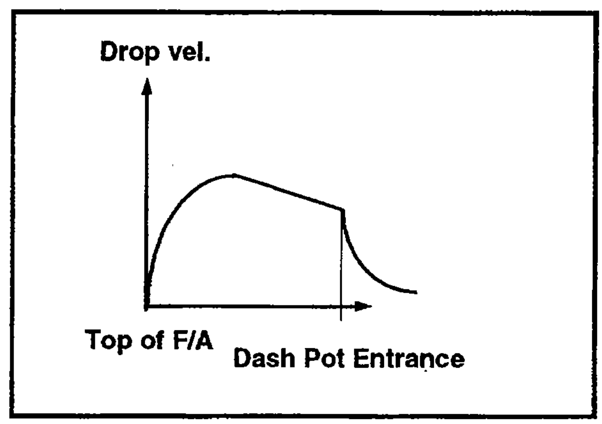


Figure 1b: Disturbed rod drop behaviour

Measurement of the drag force in the affected F.A.s using first the RCCAs in operation during the reactor trip and then new RCCAs gave identical results, i.e. the problem could not be attributed to swollen control rod tips.

An investigation of the results from the drop time testing showed that some rods deviated from a normal drop behaviour. Usually the rod accelerates quickly until the hydraulic forces start to out balance the force from gravity. The rod will then only slowly increase its velocity (figure 1 a). The rods, experiencing sticking problems or prolonged drop times had a different velocity characteristic. At the top of the F.A. the velocity increase was similar to what is found in a normal assembly but when coming about 1 meter into the guide thimble the velocity instead started to decrease indicating there was an extra force starting to act on the rod from this point (figure 1b).

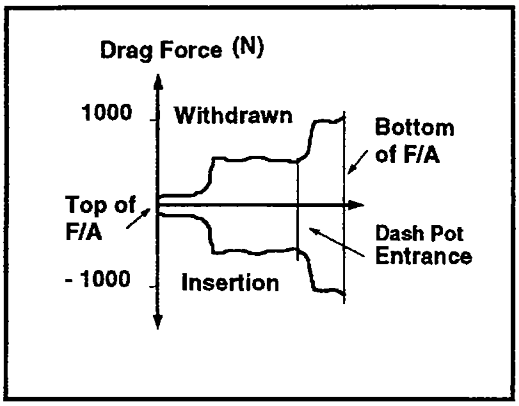


Figure 2: Typical drag force characteristics for a S-bowed F.A.

The evaluation of the drag force measurements showed that the friction at the top of the assembly was low, but about 1 meter into

the F.A. there was almost a step increase of the drag force. The force then remained more less constant until the rod reached the dash

pot where the force increased to even higher values (figure 2).

Measurements of the axial shape of the F.A.s, experiencing sticking, showed that they were bent in S-shape and with significant bending amplitudes. The location of the maximum amplitude of the upper hump of the S was coincident with the point where the drop velocity started to decrease and the step increase of the drag force was found.

The *apparent cause* to the sticking problem was concluded to be the S-bowing of the F.A.s in combination with large bow amplitudes. The extra friction force was generated when the rod was formed to the shape of the assembly. It decelerated the rod and resulted in a relatively low rod velocity when entering the dashpot. The low entrance velocity together with a further increase of friction force in the dashpot could, in more severe cases, make the rod stick in the lower regions of the F.A.

These preliminary conclusions were later confirmed by modelling of the phenomenon and experimental verification. Tests were done on a skeleton bent in S-shape having different amplitudes. The drag force were measured and it was found that gross S-shaped bows introduce friction forces with sufficient magnitude to impede the rod movement in the lower part of the F.A.

Justification for Re-start

Knowing only the apparent cause of sticking the following strategy was applied:

1. Quantify the safety significance of the sticking problem,
2. Implement a loading strategy which could be expected to improve the situation relative to the previous fuel cycle.
3. Implement of program for surveillance of the core conditions during operation.

The safety evaluation considered two aspects of the RCCA safety functions during a reactor trip:

The first function analysed was the requirement that the RCCAs, shall promptly stop the fission process. The assumptions in the safety analysis are met as long as the RCCAs reach the entrance to the dashpot in less than 2.2 seconds. The results from the drop time testing and the drag force measurements showed that the bowing problem could not create a conflict with the 2.2 second criterion.

The second safety function of the RCCAs is to provide sufficient shutdown margin for those transients which could add positive reactivity to the core. To understand how the shutdown margin could be affected by the sticking problem several calculations were made assuming that all RCCAs, except one assumed to remain fully withdrawn, would stick at different elevations in the dash pot. The studies showed that 47 RCCAs could stick at 24 steps, at BOC, without violating the minimum required shutdown margin. The corresponding value at EOC was 18 steps. It was therefore concluded that the sticking of some RCCAs in the dashpot did not affect the reactor safety.

All the assemblies experiencing sticking control rods had a relatively high burn-up, ranging from 40000 to 44500 MWd/MTU. To establish if the bowing was burn-up dependent the drag forces for a large number of F.A.s were plotted as a function of burn-up. It was found that the drag force was strongly influenced by the burn-up for certain types of F.A.s, while others seemed to be more or less unaffected.

Based on this finding it was decided not to allow higher burn-ups than 30000 MWd/MTU for F.A.s located under rodded positions as the impact below this value seemed to be limited for all types of fuel. This restriction meant that only fresh or one cycle old F.A.s could be used in rodded positions.

It was concluded that the only practical method to survey the core conditions during operation was rod drop times testing. It was however not obvious that the method was accurate enough to detect a core degradation before it could lead to a situation giving a sticking problem.

In a normal core configuration, ie with only straight or C-shaped F.A.s, the friction between the RCCAs and the guide thimble tubes will have an in-significant impact on the drop time and could at the most increase it by 0.01-0.02 seconds. The assemblies having sticking problems, ie F.A.s having S-shape, typically had increases of the drop time to dashpot in the range of 0.2 to 0.4 seconds.

Until the problem with sticking RCCAs surfaced, the cores of Ringhals have always, from a bowing point of view, been normal. Thus, it was concluded that the small variations seen in drop times between different test occasions could solely be attributed to the inaccuracy of the method. A statistical analysis of all test data for unit 3 and 4 showed that the inaccuracy was small with standard deviation not larger than 0.03 seconds. It was therefore believed that the drop time testing was accurate enough to detect a core degradation leading to S-shaped bowing of the fuel assemblies.

However as one of the rods, which stuck during the drop time testing, only had an increase of the drop time of 0.12 seconds the surveillance criterion had to be set very strict and it was decided that an increase of each position specific drop time of 0.06 seconds could at the most be accepted. The tests had to be conducted after 50, 75 and 100% completion of the fuel cycle. It was also decided that even more frequent tests had to be performed in Ringhals 3. Unit 3, which was in operation when the event occurred in unit 4, had not been able to implement the restricted loading pattern, and thus was judged to be in a more critical situation.

The results of the safety evaluation and the strategy to justify the re-start of unit 4 and the continued operation of unit 3 was presented for the Swedish Authorities. They decided that the precaution implemented were sufficient to allow the re-start of unit 4 and to continue the operation in unit 3.

Experience from Continued Operation

At a drop test in Ringhals unit 3, February 1995, the surveillance criterion was exceeded in two positions. According to the agreement with the authorities it was decided to shutdown, refuel reactor and introduce the same loading pattern as in unit 4. It was also noted that several RCCAs showed significant increases of the drop time through dashpot. During cool down the reactor tripped, at a RCS temperature = 175 °C. At the trip the two rods which exceeded the criterion and one with long dashpot penetration time stuck in the lower parts of the F.A.s. Later measurements in the fuel pool confirmed that these assemblies were bent in S-shape.

The chosen core loading pattern in unit 4 proved adequate to successfully pass the drop tests until the end of cycle 12. During the last test, in conjunction with the shutdown for the refuelling outage 1995, the surveillance criterion were slightly exceeded in a couple of rod positions. Investigations in the fuel pool revealed that even some one year old fuel had S-bows with relatively large amplitudes. The bowing was due to the impact from the surrounding F.A.s having high burn-ups and in some cases large S-bows.

Due to this it was decided that unit 3 and 4 had to face similar restriction as the previous cycle, i.e. burn-up < 30000 MWd/MTU in rodded position and surveillance of the core condition during operation by drop time testing also for the cycle 1995/1996.

During spring 1996 an analysis was completed with the aim of relaxing the very strict surveillance criterion. The new criterion were developed to better fit the kinetics of the rod during the entire drop and the core conditions now prevailing in Ringhals unit 3 and 4.

The new criteria take into account both the drop time to the dash pot (t5) and drop time through dash pot (t6) and the expected evolution of the bows as a function of time. By comparing the expected time to a critical state with the time to the next rod drop test it is possible to calculate the remaining margin against sticking at this time. The remaining margin is, in the new criteria, expressed as a safety factor.

a. t5 normal drop time + 0.06 and t6 0.55 seconds. This represents a situation with normal guide thimble tube / RCCA friction. The safety factor is expected to be > 3 at the next test occasion. No actions required. This situation is re- ferred to as normal core mechanical conditions.

b t5 normal drop time + 0.12 and t6 0.62. This represents a situation with slightly degraded core conditions. The operation can however continue without changing the frequency of the drop time testing. The safety factor is expected to be = 2-3 at the next test occasion. No actions required. This situation is referred to as "disturbed" core mechanical conditions.

c. t5 > normal drop time + 0.12 or/and t6 > 0.62. This represents a situation with more degraded core conditions. The safety factor is expected to be < 2 at the next test occasion. In this situation the test frequency must be revised and adjusted so the safety factor will be >3 at the next test occasion. This situation is referred to as "degraded" core mechanical conditions.

The new criteria were accepted by the Swedish authorities in April 1996.

Root Cause Analysis

During winter 1994/1995, the problem was further analysed with the intention to find the root cause. The F.A.s, not re-loaded in the core and therefore available in the fuel pools in both unit 3 and 4 were subject to extensive investigations. The program covered measurements and inspections of a large number of F.A.s with a varying degree of burn-ups. The forces of the holddown springs were measured and several RCCAs were also investigated.

The results from all investigations, analyses and operating experience demonstrated that the large bowing amplitudes of some of the F.A.s can be explained by irradiation induced creep deformation.

Irradiation creep is a normal phenomenon and affects all structures made of Zircaloy subject to neutron radiation. The amount of creep deformation a F.A. will exhibit after some time of operation is a function of several parameters and variables: The axial force applied on the F.A., the guide thimble tube growth, the initial deformation, the burn-up, the F.A. lateral stiffness, and the creep properties of the material.

The axial compression depends on the top nozzle spring forces. The purpose of the spring is to counteract the lift force from the flow, preventing the F.A. from losing contact with the lower core support plate. In the specific case of Ringhals unit 3 and 4 the balance of the axial force proved unfavourable, due to a moderate primary flow rate, translating into relatively high holddown spring forces.

The irradiation induced growth of the guide thimble tubes will increase the compression of the holddown springs and the axial force on the F.A.s. This effect will, however, to some extent be balanced out by the irradiation induced relaxation of holddown spring stiffness.

The initial deformation is of special importance in this case as the core is a mixture of new and older F.A.s. The older and more bowed assemblies will impact the adjacent F.A. and exert an initial deformation on them reducing the time to reach a state when the newer F.A.s undergo too much deformation.

The F.A. lateral stiffness depends on the guide thimbles and fuel rods design and on the clamping conditions of both to the grids.

Analytical models were developed to take into account the design parameters and operating conditions to quantify their influence on bow evolution. The influence of F.A. bow on drag forces and drop times were also analysed. Comparison to drag force measurements and drop time testing confirmed reasonably well the models. The models were then used for parametric and comparative studies.

The analyses revealed that the fuel in Ringhals 4 had been subject to unusually high hold down forces. It was also found that the changes from AFA to AFA2G had decreased the lateral stiffness. It explained why the older F.A. types were almost unaffected by the bowing problem while the later models of the fuel experienced S-bowing with large amplitudes in the specific Ringhals conditions.

It was concluded that the bowing in Ringhals unit 3 and 4 had been caused by large creep deformation driven by excessive compressive forces on the F.A.s, the decrease of the lateral stiffness being an aggravating factor.

Remedial Measures

It was decided to equip the top nozzles with three-leaf springs instead of the four-leaf ones used on the previous fuel types. This modification reduced the axial force by 25%. Further, to increase the lateral stiffness of F.A., the inner diameter and the wall thickness of the guide thimble tubes were increased.

The combined effect of the two remedies was expected to give an F.A. stiffer than the old type of fuel not showing any severe bowing. The modified version of the F.A.s was introduced in the cores of unit 3 and 4 for the refuelling outage 1995.

For irradiated F.A.s it was agreed to perform yielding of the hold down springs, prior to the re-start after the refuelling outage 1995, to lower the axial force to a level equivalent to 3-leaf assemblies.

The design changes of new F.A.s and the yielding of the springs were also believed to contribute to a general reduction of the F.A. deformations. The reduction rate was predicted to normalise the core conditions in Ringhals unit 3 and 4 in about 4 fuel cycles. The prediction was made with a complete mechanical model of the 157 F.A.s taking into account the behaviour of the deformed F.A.s and the impact on their neighbours.

Recent Measurements

During the cycle 1995/1996 a total of 5 drop tests have been performed in Ringhals unit 3 and 4. The tests were performed at 75 and 100% of the fuel cycle in unit 3 and at 25, 70 and 100% in unit 4. All tests during operation (ie < 100% of the fuel cycle) met the normal core criterion. The tests in both Ringhals 3 and 4 at shutdown showed in some positions increases in the dash pot penetration time (t6). The affected rods met the disturbed core criterion. These results indicated improved core condition because re-evaluation of the data from drop time testing at shutdown for the previous refuelling outage 1995 showed that several rods were in the degraded category.

Also the geometrical measurements of the F.A.s indicated improving core conditions. In unit 3 the average bow amplitude of the core had decreased with an amount being close to the one predicted. The improvement in unit 4 was even more pronounced and was significantly better than predicted. The reason for this difference in behaviour can be attributed to the amount of measures taken at the 1995 outage. In unit 4 all old, reloaded, F.A.s had the holddown springs yielded, while in unit 3 only one and two year old F.A.s were treated.

Conclusions

The remedial measures implemented on the fuel and the surveillance actions revealed an effective way to manage the fuel problems of Ringhals 3 and 4 without, significantly, impacting the availability of the two units. Recent measurements indicate that the core mechanical conditions are improving at least at the expected rate.

The bowing in Ringhals unit 3 and 4 had been caused by large creep deformation driven by excessive compressive forces on the F.A.s, the decrease in the lateral stiffness and the relatively moderate primary flow rate being aggravating factors. Some measures taken in the past years, with the aim to reduce operating costs and improve fuel utilisation, had made the fuel more sensitive to creep deformation.

The close teamwork between VATTENFALL and FRAMATOME has allowed both companies to face successfully this entirely new problem. The in-depth analysis promoted the understanding of the very complex behaviour of F.A.s under irradiation. It also enabled the fuel designer to implement adequate changes in the design of its fuel assemblies delivered throughout the world.

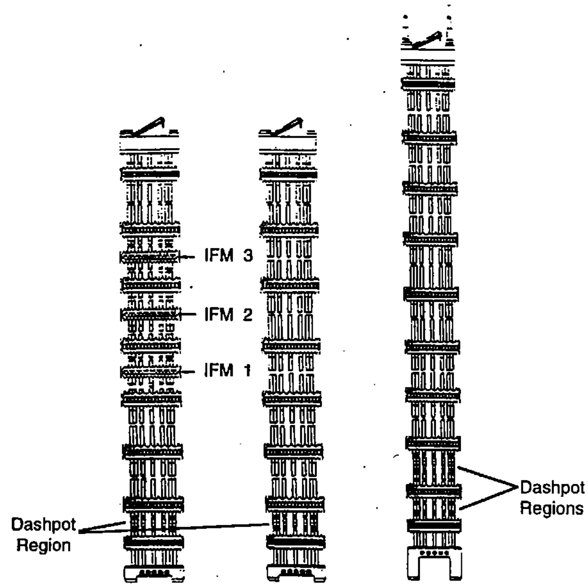
Incomplete RCCA Insertion Observations in Westinghouse-Fueled Plants

H.W. Wilson
Westinghouse Electric Corporation
OECD Specialist's Meeting on
Nuclear Fuel and Control Rods
Madrid, Spain
November 5-7, 1996

Incomplete RCCA Insertion Events

- Where
  - South Texas Unit 1: December, 1995
  - Wolf Creek: January, 1996
    - Only observed in one region
- What
  - Some rods do not fully insert (6-18 steps out)
  - Rods insert to dashpot (except 1 case)

Fuel Assembly Skeleton Comparison



Actions Taken

- Westinghouse/WOG/NRC meeting - February, 1996
  - Addressed safety significance
  - Demonstrated adequate shutdown margin
  - Discussed operating experience
- NRC issued Bulletin 96-01
- Westinghouse/WOG/NRC meetings - March and May, 1996
- Identified population of susceptible assemblies - June, 1996
- Hot cell exam of Wolf Creek assemblies - started July, 1996
- Root cause identified - September, 1996
- Final Report - December, 1996

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- Required testing
  - RCCA drop testing at end-of-cycle (or during any shutdown of significant duration)
  - Measure recoil behavior if possible
  - Drag testing of assemblies to be placed under RCCAs (refueling outage)
- Operator training

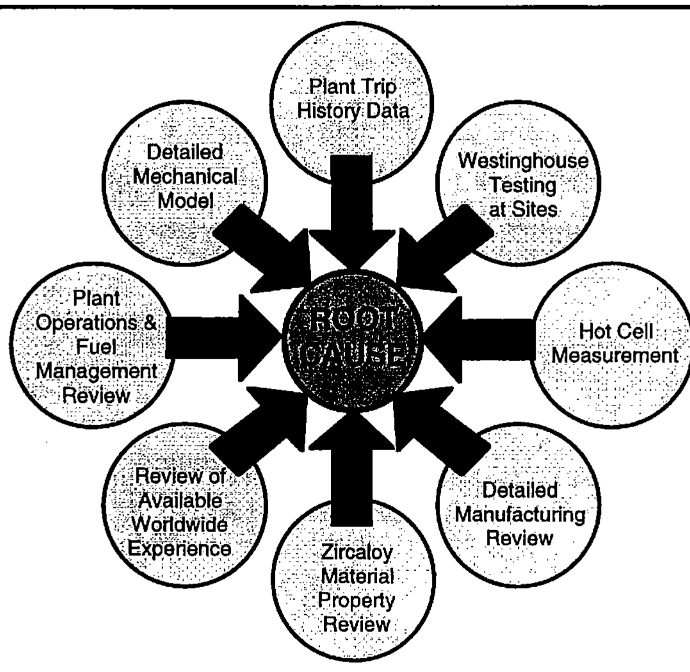
Safety Significance Evaluation

- Incomplete RCCA insertion could impact SDM, a Tech Spec requirement
- SDM analytical technique is conservative
  - Design basis: worst single stuck rod (all the way out)
  - Analytical procedure very conservative
  - Conservatisms approximately equal to stuck rod worth
- Operating plant assessment
  - Partially inserted RCCAs assumed in addition to worst stuck rod
  - All RCCAs in assemblies >42 GWD/WTU assumed to stick in dashpot region
  - Normal conservatisms assumed
  - Conclusion: SDM tech spec requirement met

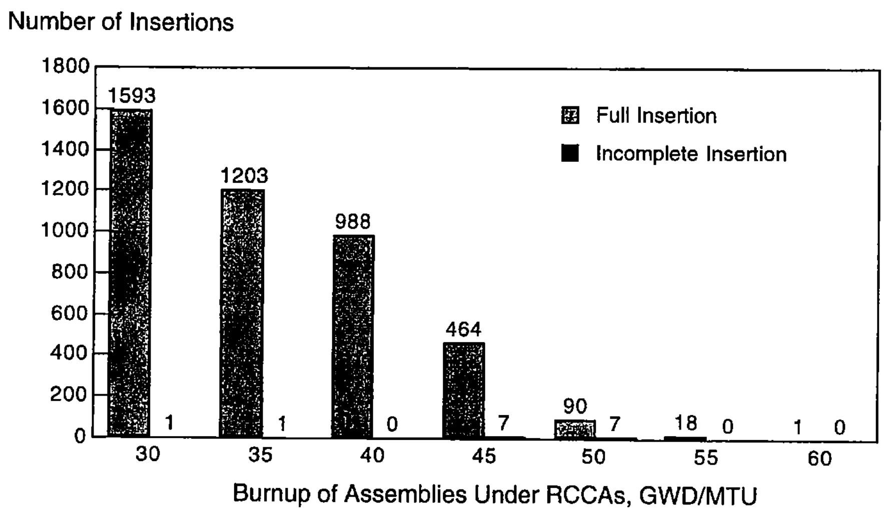
Incomplete RCCA Insertion Safety Assessment Conclusions

- Actual Wolf Creek Trip Scenario resulted in negligible uninserted rod worth
- Postulated uninserted worth is small relative to design basis assumption
- Design conservatisms bound postulated scenarios
- Current safety analyses valid

Determination of Susceptible Plants and Root Cause



Many RCCAs Have Fully Inserted in High-Burnup Fuel



Site Testing Program Has Been Completed

Plant	Fuel Type	Visuals	Drag Testing	Growth	Probe	Boroscope
Wolf Creek	17x17 V5H	✓	✓	✓	✓	✓
	17x17 V5H w/IFM	✓	✓	✓	✓	
	17x17 STD	✓	✓	✓		
Millstone 3	17x17 V5H w/IFM	✓	✓	✓	✓	
South Texas	17x17 XL	✓	✓	✓	✓	✓
Point Beach	14x14 OFA	✓	✓	✓	✓	
Surry	15x15 OFA	✓	✓	✓	✓	
VC Summer	17x17 OFA w/IFM	✓	✓	✓	✓	
Sequoyah	17x17 V5H	✓	✓		✓	
Diablo Canyon	17x17 OFA w/IFM	✓		✓	✓	
North Anna	17x17 V5H		✓	✓	✓	
Vogtle	17x17 OFA w/IFM	✓	✓	✓	✓	

Site Testing Results

- Incomplete insertion at Wolf Creek related to high fuel assembly growth (Zircaloy-4 components)
- High assembly growth only seen at a few high temperature plants
- High temperature a necessary, but not sufficient, condition for high fuel assembly growth
- Growth sensitive to power histories at high temperature plants
  - Long residence time
  - High power in later cycles
- IFM assemblies showed insignificant drag in upper guide thimble area

Hot Cell Testing

- Purpose
  - To perform in-depth testing on skeleton of two Wolf Creek assemblies which experienced incomplete insertion
- Program
  - Visual examination
  - Thimble tube dimension measurements
  - Metallography
  - Oxide measurements
  - Hydrogen measurements
  - Tensile tests

Hot Cell Results

- Growth higher than normal saturation growth
- Contributors to Wolf Creek high fuel assembly growth
  - Oxide accumulation (high temperature related)
  - Accelerated growth (high temperature related)
  - Oxide and accelerated growth components dependent on temperature and power history

Manufacturing and Material Property Review

- Detailed Manufacturing Review
  - Not related to a manufacturing anomaly
  - Thimble tubes from same lot as used at Wolf Creek did not grow or distort at other plants
- Zircaloy Material Property Review
  - Included examination data, literature review, and discussions with industry experts
  - Accelerated growth possible depending on temperature and fluence
    - Occurs after an incubation period
    - Initiation and rate depends on temperature

Industry and Operational Experience

- Worldwide Experience
  - Incomplete insertions observed elsewhere
  - Related to excessive compressive loads on fuel assembly guide thimble tubes
- Plant Operations
  - Wolf Creek assemblies with incomplete insertions experienced unusually demanding operation
    - High operating temperatures
    - Three cycle operation
    - High power in second and third cycles

Mechanical Model Conclusions

- Mechanical model developed based on available data
- Predicted growth differences between:
  - Wolf Creek assemblies with incomplete insertion
  - Wolf Creek assemblies with complete insertion
  - South Texas assemblies
- Reasonably reproduced span-dependent bow measurements from hot cell

Susceptible Design Evaluation

- Fuel assemblies with IFM grids are not susceptible
- Twelve-foot Westinghouse fuel assemblies without IFMs are not susceptible below 40 GWD/MTU
- It appears 14x14 and 15x15 fuel are less susceptible than 17x17, but no definitive conclusions have yet been drawn
- Manufacturing has not affected susceptibility

Incomplete RCCA Insertion Root Cause

- Cause of incomplete insertion
  - Excessive thimble tube distortion
  - Distortion caused by excessive fuel assembly compressive load
  - Compressive load caused by high fuel assembly growth
- High growth due to a combination of:
  - Oxide accumulation
  - Accelerated growth
  - Both are temperature sensitive
- High growth only observed in high temperature plants with certain types of power histories

Control Rod Cluster Drop Time Anomaly. Guangdong and Electricité de France Power Station

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IPSN

1. Introduction

Since 1978, as the proportion of electricity provided by nuclear power stations increased steadily, Electricité de France decided to change the operation mode for pressurised water reactors (PWRs) to regulate the French grid. One consequence was an increase in the number of movements of the control rods in relation to compared the initial technical specification of the Westinghouse licence.

To avoid the risk of premature wear of the neutron absorber rod cladding due to friction in the continuous guides of the control rod cluster guide tubes, a modification to reduce the pressure of the contact between the neutron absorber rods and the continuous guides was designed.

This modification consists in balancing the pressure of the fluid on both sides of the neutron absorber rod using holes and grooves in the rod travel housings of the continuous guides, and in adapting the geometry of the fluid outlet ports in the guide tube fairing (see Figures 1, 2, 3, 4, 5, 6 and 7). Its endurance was tested on the CEA's SUPERBEC loop at Cadarache. The results obtained were satisfactory. Besides the reduction in wear by friction of the control rods in the continuous guides, the tests showed that the drop time did not depend on the variations in flowrate. As a result, this modification was adopted for the guide tubes of the twenty PWRs of Electricité de France's 1300 MWe series.

In 1986, verifications by non-destructive testing (eddy currents and ultrasonic testing) of the cladding of the absorber rods of the 900 MWe PWR control rod clusters showed significant wear at the discontinuous guide cards, for control rod clusters which moved little, particularly the shutdown control rod clusters. The same phenomenon was found on the 1300 MWe series PWRs. It shortened the life of the control rod clusters more than the wear by friction on the control rods in the continuous guides. To understand the cause of this phenomenon, a programme of tests was carried out on Framatome's Magaly loop in Le Creusot, in 1988.

The conclusion was that the neutron absorber rods which are most worn in front of the discontinuous guide cards are those which vibrate the most under the effect of hydraulic excitation and which have the lowest contact pressure on the continuous guides.

This result led to a new design of guide tube, called M1, which reduced the vibratory range of the rods, cause of their premature wear at the discontinuous guide cards, by increasing their contact pressure on the continuous guides. The modification consists in both reducing the number of holes and grooves which balance the fluid pressure on both sides of the rod, and in modifying the geometry of the primary coolant outlet ports in the fairing of the guide tube (see Figures 5, 6 and 7). The solution adopted for the M1 guide tube results from a vast testing programme which varied several parameters on the Magaly loop. In particular, the contact pressure in the continuous guides shall not be increased beyond that of the initial design of the 900 MWe PWR guide tubes.

Drop time tests were conducted on the CEA's Hermes loop at Cadarache. Comparison of the results obtained with the M1 guide tube and with the Westinghouse guide tube of the 1300 MWe series PWRs did not reveal any significant differences. The drop time between the effective beginning of the drop of the mobile assembly (cluster and control rod drive shaft) and the entry of the neutron absorber rods into the dash-pot located in the lower part of the fuel assemblies (see figure 8), was slightly higher by 0.1 second with the M1 guide tube, but with a value of the order of 1.7 seconds, well below the corresponding safety criterion of 2.15 seconds.

In 1989, the decision was taken to adopt this new M1 guide tube for the 1450 MWe PWRs of Electricité de France's N4 series, and for the two 1000 MWe PWRs of the Guandong Power Station at Daya Bay. This design modification, both for the French 1450 MWe reactors and the Guandong reactors, was not commented upon by the Institute for Nuclear Safety and Protection because the tests did not reveal any unfavourable effect on the safety of the installation.

Due to the delay before start up of the N4 series pilot unit, Chooz B1, operation of the new M1 guide tubes for the control rod clusters began at the Chinese power stations.

2. Guandong Nuclear Power Station (DAYA BAY)

The initial start-up tests produced satisfactory results, in particular for the control rod cluster drop times (see Figures 9 and 10).

2.1. Anomaly

The first refueling outage of Unit 1 at Guandong Power Station took place at the end of 1994-beginning of 1995.

The periodic control rod cluster drop tests carried out during this outage detected an anomaly. The drop time had increased for all the control rod clusters in comparison with the results of the initial start-up tests and, in addition, for seven of the fifty three control rod clusters, it was above the safety criterion of 2.15 seconds (see Figure 9).

Three months later, additional tests carried out at the beginning of the first refueling outage of Unit 2 gave similar results, although its state was slightly less deteriorated than on Unit 1. Only one control rod cluster had a drop time higher than the safety criterion of 2.15 seconds (see Figure 10).

2.2. Safety Consequence

The consequence of this anomaly was that the safety criterion relating to control rod cluster drop time in the safety analysis report accident studies was no more met. The safety studies include the most anti-reactive control rod cluster stuck outside the core during a reactor scram.

2.3. Searching the causes of the anomaly

The investigations carried out by the utility, GNPJVC (Guandong Nuclear Power Plant Joint Venture Company) and the vendor, Framatome, considered around ten possible causes (see table in Figure 11).

The results of the expert appraisal of a guide tube extracted from the H12 position, the non-destructive tests on the control rod clusters and the supplementary endurance tests (control rod cluster drops and translations) carried out on the Hermes loop which made it possible to reproduce the drop time increase phenomenon (see Figure 12), led to the following conclusions : the increase in the neutron absorber rods contact pressure along the continuous guides produced a progressive wear of the continuous guides on contact with the control rods (see Figure 13). This wear resulted in increased hydraulic force holding control rods on the continuous guides. Therefore, there is an increase of the resisting force opposing the weight of the mobile assembly, and as a result, of the drop time. This increase is progressive in accordance with the number of drops and the number of translations made. Thus, the difference in behaviour between the control rod clusters of the two units of Guandong Power Station at Daya Bay was due both to the difference in the number of reactor scrams they had undergone (38 for Unit 1 and 18 for Unit 2), and to the difference in the number of control rod cluster movements made for regulation (greater for Unit 1 than for Unit 2).

2.4. Dealing with the anomaly

The National Nuclear Safety Administration of the People's Republic of China requested technical support from the Institute for Nuclear Safety and Protection and the advice of the French safety authority, the Nuclear Installations Safety Directorate to assess the safety of the power station :

- for its provisional operation, taking into account the palliative measures implemented by the operating organisation,
- for the final solution of the anomaly.

2.4.1. Provisional measures for one cycle

In a first time, the operating organisation, GNPJVC, requested authorisation to resume provisional operation (for one cycle before final repair) and presented a dossier to the Chinese safety authority, which described measures to compensate for this deterioration :

- addition of eight control rod clusters initially forcasted for mixed uranium and plutonium oxide fuel (MOX), so as to increase the antireactivity margin (see Figure 14),
- replacement of the guide tubes of Unit 1, corresponding to the clusters which have the longest drop times with the new guide tubes available (1300 MWe guide tube type),
- base load operation so as to minimise the control rod clusters movements,
- reduction of the number of operation manoeuvres which could trigger a reactor scram, so as to minimise the number of control rod clusters drops,
- application of a surveillance programme supplementing the normal periodic tests to verify the control rod clusters' ability to perform their function of stopping the nuclear chain reac-

tion. This surveillance consisted on the one hand of measuring the travel time on 10 of the 225 steps of the total travel of each cluster every fortnight with the unit on power, and on the other of measuring the control rod clusters drop time every 45 days. In the event of significant change in the travel time, the drop time should be immediately measured.

Furthermore, as the development kinetics of the phenomenon was poorly understood, the operating organisation and the vendor presented a dossier which aimed to justify the safety of these units for a control rod cluster drop time of 2.62 seconds instead of 2.15 seconds. The aim of this dossier was to justify that a sufficient level of safety could be maintained with a drop time slightly higher than the safety analysis report criterion by using the specific characteristics of the cycle underway. This made it possible to widen the margins in the accident studies, particularly for control rod cluster drop accidents and control rod cluster ejection accidents.

Finally, the operating organisation submitted a dossier justifying safety in the event of three control rod clusters being unavailable, so that operation could continue if such an eventuality arise.

Following analysis of these dossiers, the Institute for Nuclear Safety and Protection submitted a technical advice to the Nuclear Installations Safety Directorate which sent it to its Chinese counterpart, the National Nuclear Safety Administration.

This advice pointed out that :

- rather than to try to justify safety in the event of three control rod clusters being considered unavailable, a solution which enabled to garantee the emergency shutdown function availability, should be looked at first.
- a provisional operating licence for Guandong Power Station could be granted on the following conditions :
  - to specify the action to be carried out by the operator if the instrumentation which measures the travel time of a control rod cluster become unavailable,
  - to set a drop time recording for the control rod clusters with the longest drop time, in the event of a reactor scram,
  - to set a hold point at around 45 days following the beginning of the second cycle to examine the results (travel time, drop time) of the supplementary surveillance programme, and to subordinate continuing operation to the authorisation of the National Nuclear Safety Administration.
  - to maintain a vigilance stepped up regarding any change in the other parameters which could be associated with the anomaly (primary flow rate, flux azimutal imbalance etc.),
  - to adapt if necessary, the supplementary surveillance programme during provisional operation of the two reactors in accordance with experience feedback and the changes in knowledge on the anomaly,
  - to specify the conditions for which a control rod cluster drop time criterion higher than the criterion of 2.15 seconds from the final safety analysis report could be considered acceptable :

1) taking into account the limited duration of provisional operation before final treatment of the anomaly and the conservatism of the calculations carried out, a drop time criterion, excluding earthquakes, of 2.45 seconds could be accepted,

2) keeping in mind the evolutive nature of the anomaly, and the fact that the new criterion set at 2.45 seconds makes it possible to consider the control rod clusters damaged by the anomaly as being available, closer to the stuck threshold (weight of the mobile system equal to the result of the resisting forces) than with the former criterion of 2.15 seconds. The criterion of unavailability of a control rod cluster should be set, in the specific surveillance programme at a value slightly lower than 2.45 seconds, for example 2.40 seconds.

2.4.2. Provisional operation

After the replacements of the guide tubes carried out on Unit 1, the anomaly was reduced to just one single control rod cluster whose drop time was slightly higher than 2.15 seconds for each of the two units. The National Nuclear Safety Administration gave its authorisation to resume operation of the power station under the provisional conditions, taking into account the advice of the Institute for Nuclear Safety and Protection :

- on 20.05.95 for Unit 2,
- on 30.06.95 for Unit 1.

The application of the specific surveillance programme gave the following results :

1) the measurement of the travel times for ten of the 225 steps of the total travel of each control rod cluster, carried out periodically with the unit on power, did not reveal changes other than those due to the slight inaccuracies in the measurements,

2) the operating organisation took advantage of a reactor scram (at the end of August 1995), caused by disturbances in the electricity grid due to a typhoon, to measure the control rod cluster drop time. The results did not show any particular change other than those due to the slight inaccuracies in the measurements.

In conclusion, these results showed that temporary operation of these units with measures to compensate for the anomaly could be continued until final repair which was planned for the next refueling outages.

2.4.3. Final solution

The operating organisation and the vendor decided to replace the M1 type control rod cluster guide tubes with the type used in Electricité de France's 1300 MWe series PWRs, adapted to the geometry of the Guandong reactors. Indeed, this type of guide tube was designed to reduce the risk of premature wear of the neutron absorber rod cladding in the continuous guides and thus enable a high number of movements to be executed by the control rod clusters. Experience feedback from the twenty PWRs of Electricité de France's 1300 MWe series was satisfactory as regards the drop times measured during each refueling outage for these reactors.

The replacement was carried out during the following refueling outages :

- in January-February 1996 for Unit 2,
- in April-May 1996 for Unit 1.

Figure 15 shows the drop times measured on Unit 2 with the 1300 MWe type guide tubes, compared with those obtained during the initial start-up tests with the M1 type guide tubes. These results show that :

- the drop times with the 1300 MWe type guide tubes are shorter than with the M1 type guide tubes (on average 1.36 seconds for black control rod clusters weighing 126 kg and 1.41 seconds for grey control rod clusters weighing 115 kg),

- the scatter of drop times with the 1300 MWe type guide tubes is lower than with the M1 type guide tubes.

This solution corrects the anomaly relating to the increase in control rod cluster drop times.

However, the wear of the cladding of the control rods in front of the discontinuous guide cards must be monitored by periodic non-destructive tests, such as in the PWR power stations of the 900 and 1300 MWe series of Electricité de France.

3. Electricite de France 1450 MWe PWRs

Analogies exist between the Guandong reactors and those of the N4 series of 1450 MWe relating in particular to the presence of control rod cluster guide tubes of the same design and to the existence of a primary flow rate which is higher than those of the 900 and 1300 MWe series. The Nuclear Installations Safety Directorate has asked Electricité de France, in May 1995, to assess the impact of the Guandong incident on the safety of the French power reactors and to develop a remedy for this anomaly. This request applied, in particular, for the commissioning of the lead unit of the N4 series (Chooz B1).

3.1. Implementation of a provisional solution

In order to confirm the existence of the anomaly, the control rod drop times were checked during the cold and hot pre-critical tests on Chooz B1.

However, the investigations on the root causes of the incident which occurred at Guandong plants had shown that the increase in hydraulic force holding the control rod clusters against the M1 control rod cluster guide tubes had induced the anomaly. Electricité de France planned to proceed with a modification. This modification had to be validated on the reactor during the pre-critical tests on Chooz B1. These tests were carried out with M1 control rod cluster guide tubes, except for the two positions equipped with modified M1 control rod cluster guide tubes.

The modification consisted of welding obturation plates of the two upper levels of the ports in the control rod cluster guide tube cover (eight ports per guide tube cover were sealed). This led a cover geometry approaching that of the 1300 MWe series control rod cluster guide tubes, in which no control rod cluster drop time anomaly had been observed to date (Figure 3). This solution was intended to reduce significantly the forces induced by the plates in the upper parts of the continuous guides.

The operational validation of the modification of the M1 control rod cluster guide tubes was carried out on an experimental test loop and by additional endurance tests on the actual reactor configuration (cold and hot-precritical tests). The loop tests consisted of assessing the increase in the friction forces between the control rod cluster and the control rod cluster guide tube, depending on the hydraulic conditions (Magaly tests), and of quantifying the control rod cluster drop time evolution versus of the number of insertion steps carried out by the rod cluster control assembly (Hermes tests).

The endurance test programme on the reactor was performed with two modified M1 control rod cluster guide tubes during the first pre-critical tests of Chooz B1. It included an initial sequence of 36,000 steps for the rod cluster control assemblies (representative case of a base load cycle), followed by a sequence of 360,000 steps (1 base load cycle and one load following cycle). During these tests, the control rod clusters drop times were measured.

The test programme analysed by the Institute for Nuclear Safety and Protection showed that the results obtained from the Magaly tests presented very variable gains depending on the hydraulic conditions in the loop. Since observations made on the Guandong units had shown considerable disparity between different control rod clusters drop times, the Institute for Nuclear Safety and Protection concluded that it could not be excluded that the hydraulics of the upper plenum might influence the observed phenomena. However, the Hermes tests did not make it possible to reproduce these flows, and only two modified M1 control rod cluster guide tubes were implemented on Chooz B1 for the pre-critical test campaign. The Institute for Nuclear Safety and Protection considered that it would be difficult to be sure, using only these tests, that the by-pass flow in the upper plenum was without consequences for the control rod cluster drop times for the whole core. This analysis concluded that, despite the encouraging results from the experimental phase qualification tests, the solution proposed by Electricité de France should nevertheless be validated through additional reactor tests. Due to this, Electricité de France defined a supplementary programme of periodic surveillance tests to be performed during the reactor operating cycle. The frequency of these tests should be defined in order to assure the absence of evolution of the control rod drop time on the reactor (with respect to with that observed during the pre-critical hot tests).

3.2. Confirmation of the anomaly

The reactor remains sub-critical (boron concentration = 2000 ppm) throughout the pre-critical tests carried out with the implementation of an antidilution procedure. The Institute for Nuclear Safety and Protection thus considered that the Chooz B1 core could be loaded and the cold and hot pre-critical tests could be carried out with the M1 control rod cluster guide tubes (i.e those likely to be affected by the anomaly revealed on the Guandong units with he exception of the two modified M1 control rod cluster guide tubes, introduced into the reactor for validation).

The Hermes tests carried out with a modified M1 control rod cluster guide tube had shown an evolution of drop time with the number of drops which was clearly slower than that observed on the Hermes tests with the Guandong M1 control rod cluster guide tubes. The results obtained made it possible to expect an acceptable control rod drop time during the first cycle while waiting for the new 1300 MWe type control rod cluster guides to be manufactured and to be installed at the first outage for refueling. The results of the cold and hot pre-critical tests of Chooz B1 confirmed the anomaly found at Guandong. They revealed that :

- the two modified M1 control rod cluster guide tubes presented drop times which were equal to or greater than their non-modified counterparts (Figure 16),
- the drop times obtained were greater than the previsional values resulting from tests carried out on the Hermes loop. For 16 positions, of which one had a modified M1 control rod cluster guide, the drop time criterion was not met.

These results revealed the preponderant role of the geometry of the sleeves and of the split tubes (which ensure the continuous guides for the neutron absorber rods (Figure 6)) on the hydraulic forces in the continuous guide tubes. Indeed, the modification of the M1 control rod cluster guide tubes only affected the two upper ports of the cover. The geometry of the sleeves and the split tubes, which was different from that of the 1300 MWe series PWRs, was retained.

As this situation was not compatible with the conditions to let the reactor go critical, Electricité de France finally decided to replace all the M1 control rod cluster guide tubes with 1300 MWe type control rod cluster guide tubes (Figure 17) for which satisfactory experience feedback had been obtained from the 1300 MWe reactors.

3.3. Installation of a final solution

Prior to reactor criticality, the Nuclear Installations Safety Directorate asked Electricité de France to both justify in advance that the 1300 MWe type control rod cluster guide tubes would behave properly in an N4 type reactor, and to present a programme of supplementary tests at the time of the pre-critical tests to validate the final solution (1300 MWe type control rod cluster guide tubes), as well as the definition of a supplementary test programme during the cycle.

3.3.1. Acceptability, from the safety point of view, of the 1300 MWe type control rod cluster guide tubes for the N4 series

In order to prove the safety of the 1300 MWe type control rod cluster guide tubes for the N4 series, Electricité de France undertook :

- to compare the characteristics of the 1300 MWe with the N4 series to show their similarities (hydraulic, geometrical and mechanical),
- to demonstrate that the type of control rod cluster guide tube (1300 MWe or M1) had no significant effect on the hydraulics of the upper plenum or on the mechanical behaviour of the N4 train control rod cluster guide tubes and rod cluster control assemblies.

Figures 18 and 19 show the characteristics of the elements, mentioned below, which are likely to participate directly or indirectly in the control rod cluster drop times :

- the upper internals,
- the control rod cluster guide tubes,
- the mobile equipment,
- the drive mechanisms,
- the fuel assemblies,
- the hydraulic conditions.

As the demonstrations of Electricité de France were based mainly on a comparison of the design characteristics of the N4 and 1300 MWe series, the utility considered that the drop times, which would be observed on the N4 series, should be close to those observed on the 1300 MWe series, or even slightly lower (as the weights of the mobile equipment and the control rod clusters are greater on the N4 series).

The design similarities between the 1300 MWe and N4 series tended to confirm the conclusions relating to obtaining a comparable drop time between the two series. However, the flowrates in the N4 control rod cluster guide tubes had been estimated on the basis of a three-dimensional model of the hydraulics of the upper plenum for which experience feedback had revealed the complexity of the phenomena. This applied, in particular, to the effects of flowrate redistribution in the control rod cluster guide tubes, influencing the plating. As a result, the Institute for Nuclear Safety and Protection considered that the validation of the conclusions drawn on the design and operation similarities between the two reactor series could only be provided by tests realised on the actual N4 reactor configuration, in particular during the first cycle.

To complete the demonstration of the suitability of the 1300 MWe series control rod cluster guide tubes for the N4 series, Electricité de France also used experience feedback from the 1300 MWe trains as well as from Units 1 and 2 of Guandong, following the installation of 1300 MWe type control rod cluster guide tubes on these units. Thus Electricité de France was able to justify the acceptability of the proposed modification, without having to carry out futher tests on experimental loops (Magaly and Hermes). The representativeness of these experimentals loops was questioned by the Institute for Nuclear Safety and Protection. This doubt was supported by the high drop times observed on the modified M1 control rod cluster guide tubes in the actual reactor tests, contradicting the Hermes test results.

3.3.2. Surveillance programme

In order to complete the demonstration of the applicability of the 1300 MWe type control rod cluster guide tubes to the N4 reactor from a safety point of view, and to validate the implementation of this solution, Electricité de France has completed the standard surveillance programme of the 1300 MWe series with the following tests on Chooz B1 before the reactor went critical :

- under cold shutdown conditions, 10 consecutive drops, recording the drop time on the banks which in the previous tests consisted of the modified M1 control rod cluster guide tubes,
- under hot shutdown conditions, 3 consecutive drops on the two banks which consisted of the slowest control rod cluster and the fastest control rod cluster respectively.

The Institute for Nuclear Safety and Protection took into account the design differences between the N4 and 1300 MWe trains, and the differences between the results obtained from tests with the modified M1 control rod cluster guide tubes and those calculated in the studies. The Institute for Nuclear Safety and Protection deemed that Électricité de France should extend the test programme for Chooz B1 in order to ensure there would be no shift in the control rod cluster drop time during the first cycle. A control rod cluster drop time test will have to be performed in the hot shutdown state, in the middle of the cycle and at the end of the cycle.

3.3.3. Results of cold and hot pre-critical tests

Figure 20 represents the map of the core with the positions of the black and grey control rod clusters and the direction of the upper plenum water outlet nozzles.

Test of 10 successive drops under cold shutdown conditions :

The results obtained with the 1300 MWe type control rod cluster guide tubes show constant drop times. They fall into a scatter range of 0.05 second. This scatter is totally acceptable, taking into account the accuracy of the control rod drop time measurement (several hundredths of a second) and the slight variations in the value of the physical parameters which condition the control rod drop time, from one drop to another.

For reference, figure 21 gives a comparison of the control rod drop times obtained during the initial control rod drop and after 10 successive drops. It shows the results for the M1 control rod cluster guide tubes, the modified M1 and the 1300 MWe type for positions F4, D12, P6 and M14 which correspond to a shutdown sub-bank. All the drop times obtained for the different control rod cluster positions in the core are given in Figure 22. In the case of the M1 and modified M1 control rod cluster guide tubes, a systematic increase in the control rod drop time between the first and the last measurement of around 0.16 and 0.76 seconds (depending on the positions) was noted, whereas no shift in drop time can be observed for the 1300 MWe type control rod cluster guide tubes.

Pre-critical tests in hot shutdown conditions :

- Control rod drop times:

Figure 23 gives the comparisons of the control rod drop times obtained at Chooz B1 with the M1 control rod cluster guide tubes and the 1300 MWe control rod cluster guide tubes depending on the position of the control rod cluster in the core.

It can be noticed that replacing the M1 control rod cluster guides with the 1300 MWe type control rod cluster guide tubes leads to a significant decrease in both the drop times and the scatter. With regards to the spatial distribution, the M1 control rod cluster guide tubes presented high drop times both on the 0°-180° axis (0.95 second with regard to the average) and next to the outlet nozzles (0.15 seconds with regard to the average). With the final solution, less scatter is observed (Figure 24), as on the 1300 MWe series, in the drop time (0.05 second with regard to the average) around the outlet nozzles.

- Results of 3 successive drops :

After measuring the drop times for all the rod cluster control assemblies, three successive drops under hot shutdown conditions were carried out on the eight control rod clusters of the two sub-banks (SA1 and X1) including the black control rod cluster N3 (the slowest) and the grey control rod cluster H12 (the fastest).

Figure 25 gives the results of these three drops. Figure 26 indicates the change in drop times obtained for all the drops measured. As with the results obtained during the cold pre-critical tests, a constant drop time is observed with a totally acceptable scatter with regard to the accuracy of the measurement and slight variations (thermal-hydraulic flow rate for example) which can affect the measurement of one drop time or another.

In conclusion, the hot and cold pre-critical tests carried out on Chooz B1 following the installation of the 1300 MWe type control rod cluster guide tubes gave satisfactory results.

3.3.4. Representativeness of the experimental test loops

From the first pre-critical tests carried out at Chooz B1, the drop times obtained were, for the M1 and modified M1 type control rod cluster guide tubes, considerably higher than those shown during the tests in the Hermes test loop. Furthermore, one of the modified M1 control rod cluster guide tubes, for which a reduction in drop time was expected in view of the Hermes results, showed a higher control rod cluster drop time than that of its non-modified symmetrical counterpart during the tests in the reactor (Figure 16).

Following this anomaly, Electricité de France examined the three M1 and modified M1 control rod cluster guide tubes. This examination revealed the beginning of wear of the continuous guides, caused by the nitride coated control rod cluster, which would explain the change in drop time. However, the discrepancies observed between the results of the tests on loops and the tests in the reactor, cannot yet be clearly explained.

Bearing in mind the complexity of the flows in the reactor, the Institute for Nuclear Safety and Protection requested that the representativeness of the experimental test loops (in particular the Hermes loop) with regard to the reactor case, be analysed. The conclusions drawn should be applied to the experimental validation of new designs or improvements for control rod cluster guides.

4. Conclusion

The anomaly of control rod cluster drop time revealed at Guandong Nuclear Power Station in Daya Bay and in the Chooz B1 pilot unit for the N4 series, led to the replacement of the M1 type control rod cluster guide tubes with 1300 MWe PWR type guide tubes, adapted to the geometry of the Guandong reactors and the 1450 MWe reactors of the N4 series.

The comparison of the drop times obtained with the 1300 MWe type control rod cluster guide tubes in Guandong 2 with those obtained during the initial start-up tests with the M1 control rod cluster guide tubes, showed that this solution was applicable for the Guandong Power Station.

The hot and cold pre-critical tests carried out in Chooz B1 following the installation of the 1300 MWe type control rod cluster guide tubes gave satisfactory results. These met the safety criterion for N4 series control rod cluster drop times (2.15 s under hot shutdown conditions).

The drop time tests which will be carried out in the middle of and at the end of cycle 1 of Chooz B1 should make it possible to finally validate the solution already successfully implemented at Guandong.

However, this anomaly has revealed the limits of representativeness of the experimental test loops with regard to the real reactor configuration. In view of this, it has been deemed necessary to ask Electricité de France to pursue its analysis both on the understanding of the phenomena which led to this anomaly and on the limits of the representativeness of the experimental test loops.

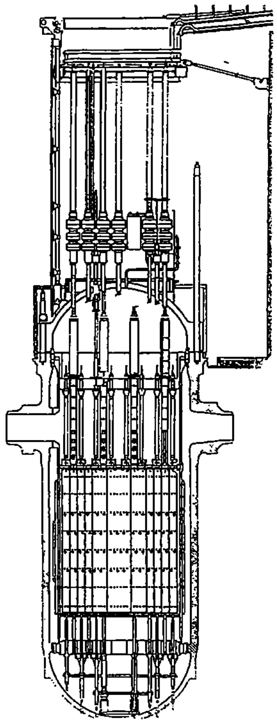


Figure 1. Guangdong reactor block assembly

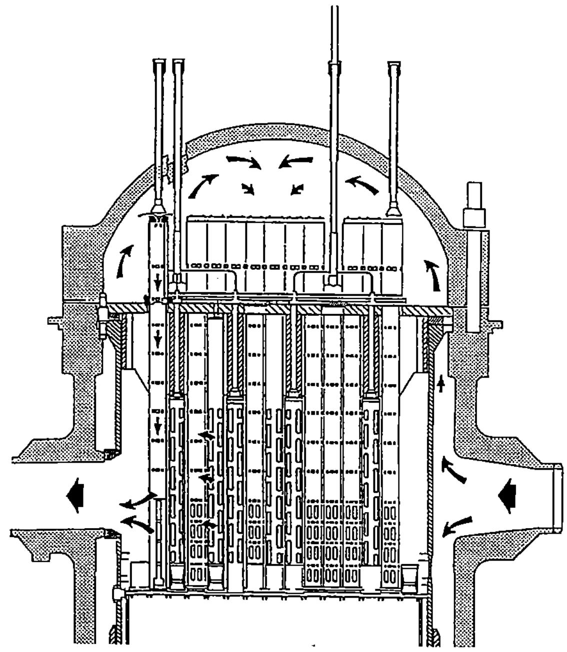


Figure 2. Diagram of flow primary coolant through reactor vessel dome and upper internal

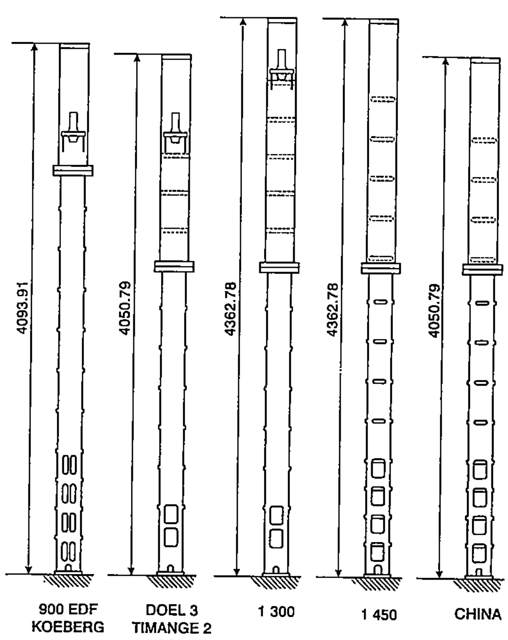


Figure 3. Development of 17 x 17 control rud cluster guides

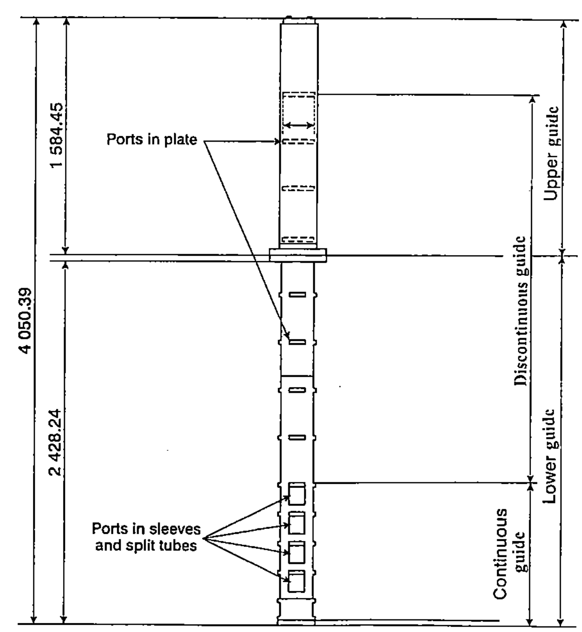


Figure 4. Diagram of Guangdong 1 and 2 guide tube

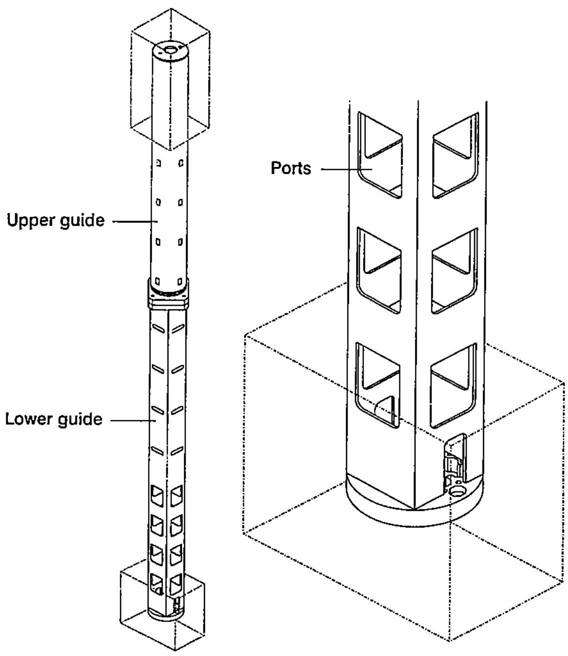


Figure 5. Guangdong control rod cluster guide

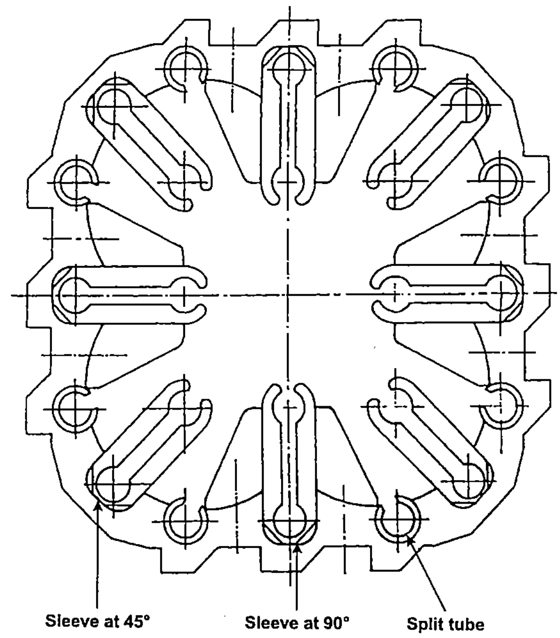


Figure 6. Guangdong. Continuous guides cross section

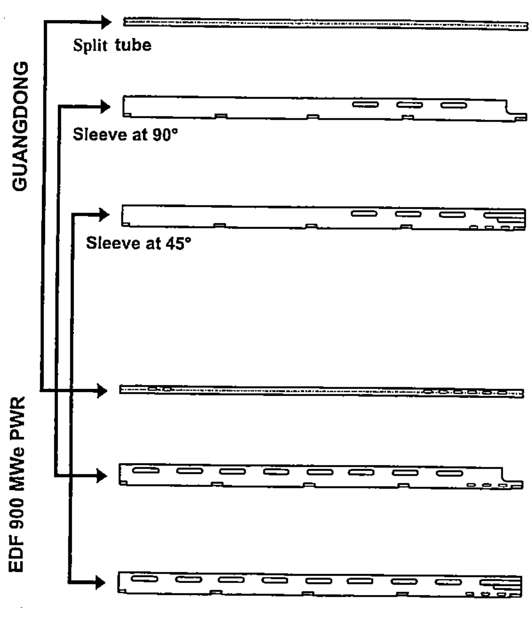
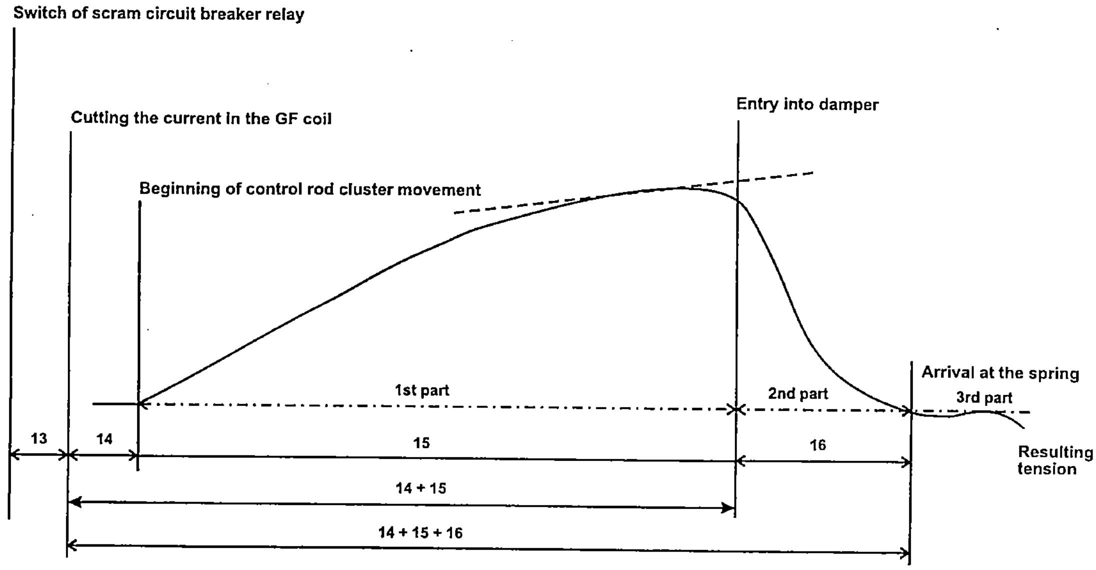


Figure 7. Diagram of continuous guide tubes and split tube of the continuous guides of the CPY series and Guangdong



- Technological criteria: T4 + T5 < -1.75 s
- Test criteria not to be exceeded: T4 + T5 < 1.90 s
- Check: T4 + T5 + T6 < 2.75 s
- Check the shape of the curve

Figure 8. Examination of recording

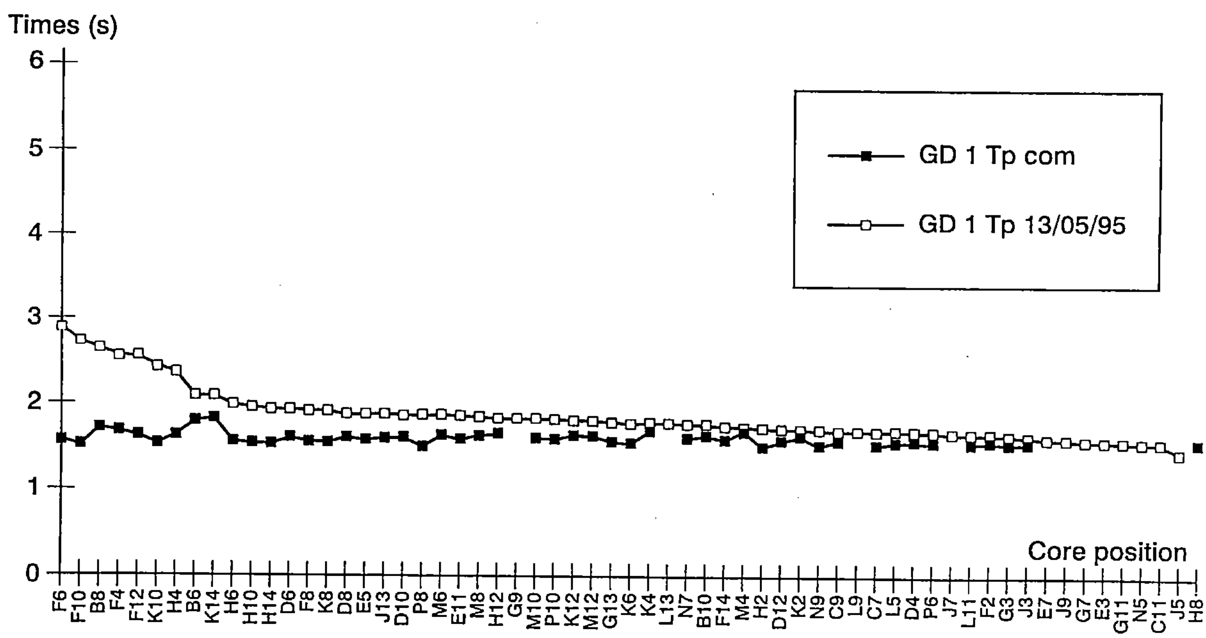


Figure 9
Guangdong 1 - Drop times classified according to measurements (decreasing values) of 13/05/1995 (all groups)

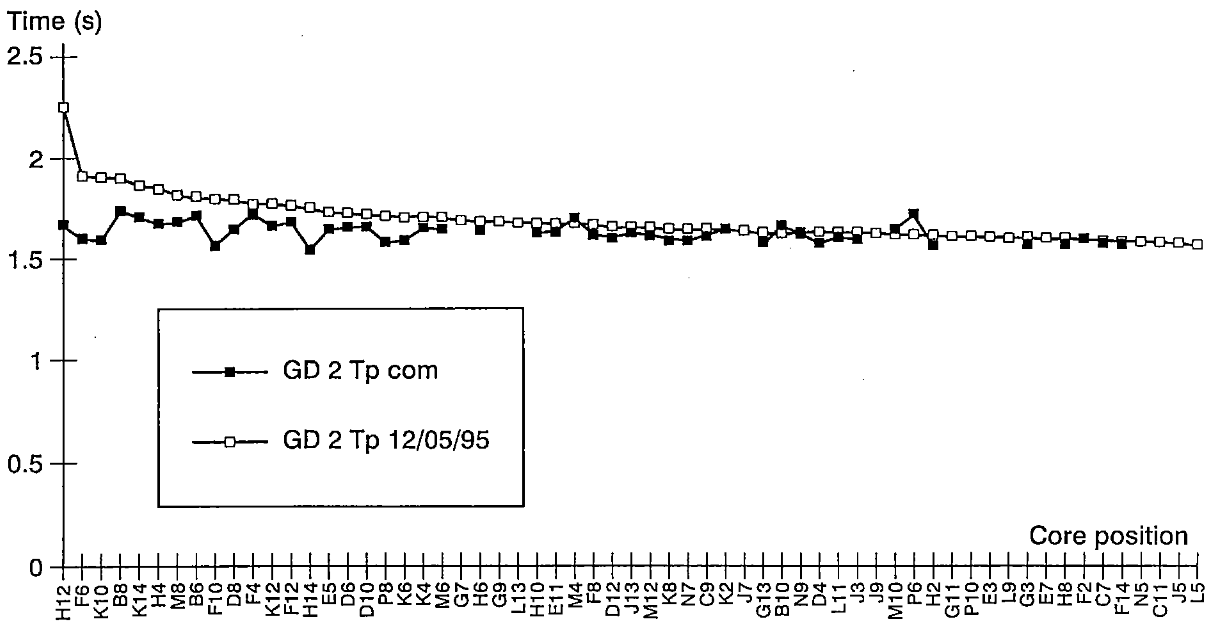


Figure 10
Guangdong 2 - Drop times classified according to measurements
(deecreasing values) of 12/05/1995

Figure 11
Causes of High Drop Times in Guangdong 1 hypotheses

Code	Cause	Investigation results
A	High hydraulic pressing force between the rod and the guide	- Dismantling of a guide tube - Impressions take - Various inspections
B	Imperfect alignment in the drop channel	- Inspections performed ⇨ NORMAL - Inspections of locations
C	Incorrect positioning after operation due to internals degradation	- TV Inspections internals "centering" ⇨ NORMAL
D	Deformation of the guide tubes and/or assembly due to temperature or flow rate	- DRAG TESTS ⇨ NORMAL - Inspections with compression fuel
E	Dashpot effect on the mechanism, assemblies or internals	- TV Inspections of possible problem regions ⇨ NORMAL
F	Non raising of the thermal sleeves - Flow rate effect - Mechanical effect	- Inspections of possible raising of 12 sleeves ⇨ NORMAL
G	Misposition of the reactor vessel head and top of CRDM (seismic tie rod)	- Considerable play ⇨ no possible effect on rod drop time
H	Deformation of fuel assembly pins (top end bending)	- Visual inspection ⇨ NORMAL
I	Vibrations, impact of components ⇨ Deformation (other than internals mentioned in C)	- C + J ⇨ NORMAL
J	Degradation of the hold-down spring ⇨ internal and fuel assembly deterioration	- Inspections of the dimensions of the hold-down spring ⇨ NORMAL
K	Presence of loose parts or cruds in the guide tubes	- TV inspection ⇨ NORMAL

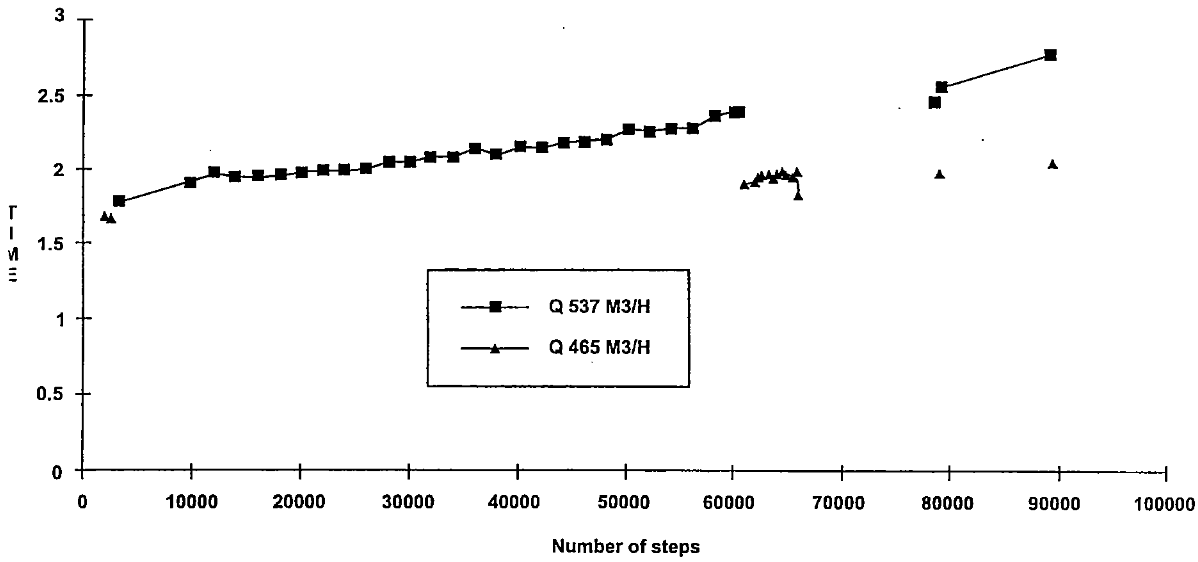


Figure 12
Guangdong Hermes drop time

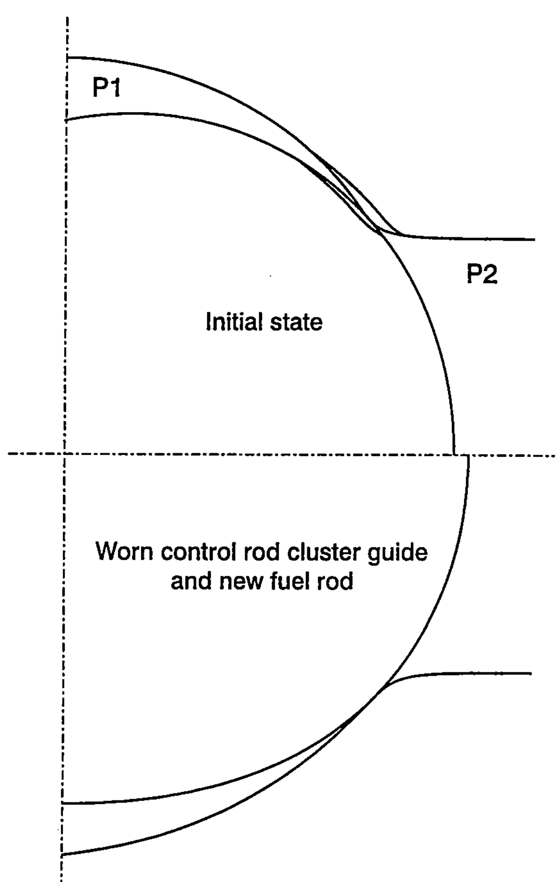


Figure 13
Fuel rod contact in continuous guides

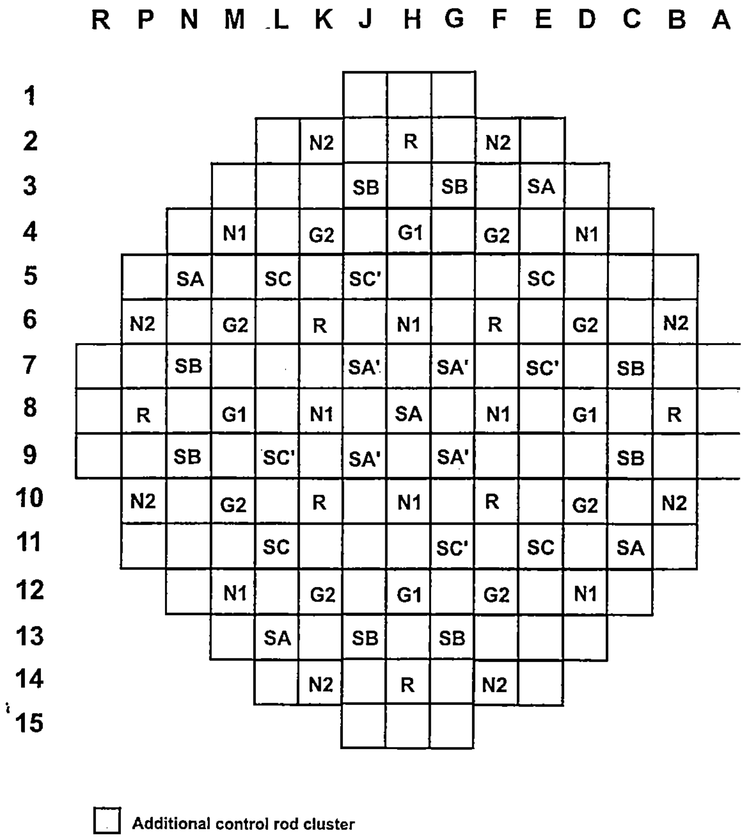


Figure 14
GUANDONG
Core with 61 control rods

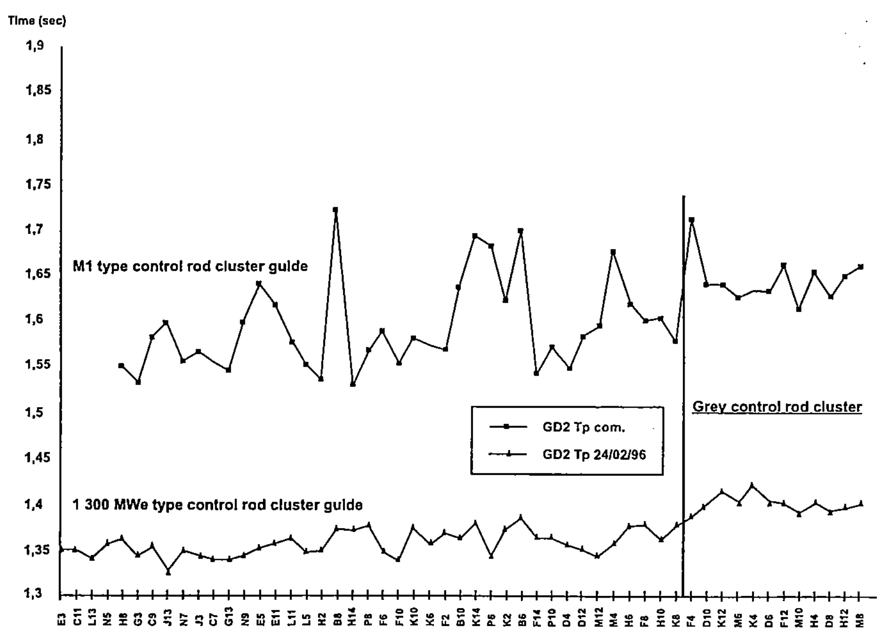


Figure 15
Guangdong 2 - Drop time at start up and after installation of 1300 MWE type control rod cluster guides

Modified M1	Non modified M1
2.31 s	2.32 s
2.47 s	2.05 s

Figure 16
Drop time in Hot State modified M1 Control Rod Cluster Guides and non-modified M1 Control Rod Cluster Guides

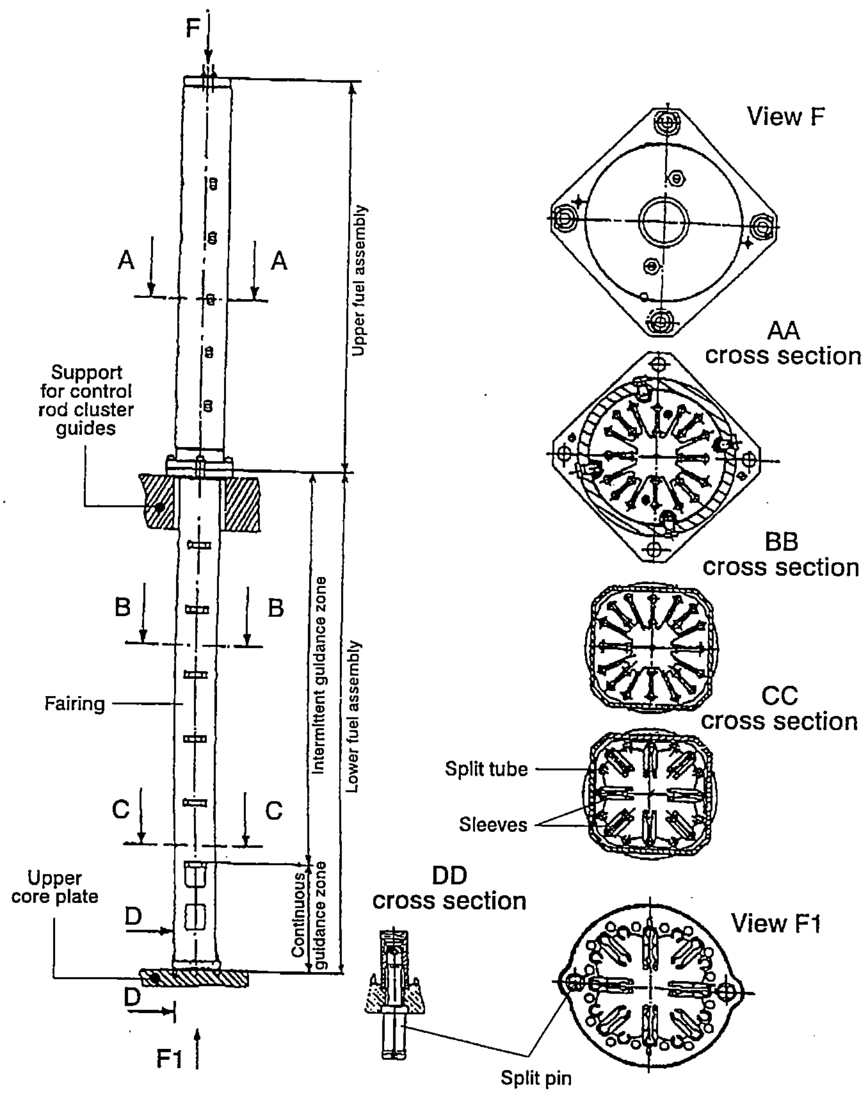


Figure 17
N4 serie 1300 MWE type control rod cluster guide

	1300 MWe		N4
Upper internal equipment			
Control rod cluster support	Identical to the nearest diameters		
Upper plates			
Structural columns	50		56
Number of control rod cluster guides	73		73
Control rod drive shaft – Weight (kg)	74	72.5	73
Control rod drive shaft – Length (mm)	7500	7368	7402
Grey control rod cluster weight (kg)	54		59
Black control rod cluster weight (kg)	45		49
Control rod cluster weight dropping (kg) – Black	119	117.5	122
Control rod cluster weight dropping (kg) – Grey	128	126.5	132
Length of drop (steps)	260		260
Types of fuel assemblies	AFAXL and AFA2GL		AFAXL
Number of fuel assemblies	193		205

Figure 18
Comparison of parameters between the N4 and 1300 MWE series

		1300 MWe SERIES		N4 SERIES
FLOWRATE (m^3/h)*		Mecha. flowrate	NO1	Mechanical flowrate
VESSEL FLOWRATE		95240	96859	103100
FLOWRATE through PSC	D.C.	94383	95987	
	D.F.	92478	94069	100625
Average FLOWRATE per fuel assembly	D.C.	489.0	497.4	
	D.F.	479.2	487.4	490.8
Min. FLOWRATE/GUIDE (central coeff.=1)	D.C.	489.0	497.4	
	D.F.	479.2	487.4	490.8
Max FLOWRATE/GUIDE (Peripheral coeff.=1.17)	D.C.	572.2	581.9	
	D.F.	560.6	570.1	574.3

Figure 19
Comparison of N4 and 1300 MWE flowrates

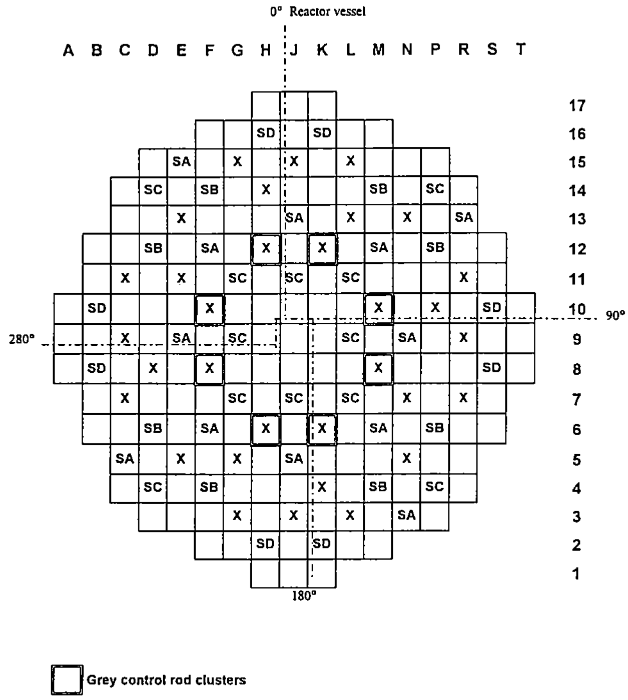


Figure 20
Position of Control Rod Cluster in the core

	F4		D12		P6		M14	
	M'1	1300	M'1	1300	M1	1300	M1	1300
$T_{INITIAL}$	3.28	2.02	3.62	2.00	3.28	2.06	2.68	2.01
T_{FINAL}	3.74	1.99	4.38	1.99	3.67	2.03	2.84	2.00
$T_{FINAL} - T_{INITIAL}$	+0.46	-0.03	+0.76	-0.01	+0.39	-0.03	+0.16	-0.01
T_{MIN}	3.28	1.99	3.62	1.99	3.28	2.01	2.68	1.97
T_{MAX}	3.74	2.04	4.38	2.05	3.67	2.06	2.86	2.01
$T_{MAX} - T_{MIN}$	+0.46	0.05	0.76	0.06	0.39	0.05	0.18	0.04

Figure 21
Bank SE Comparison of 11 Drops Under Cold Leg conditions

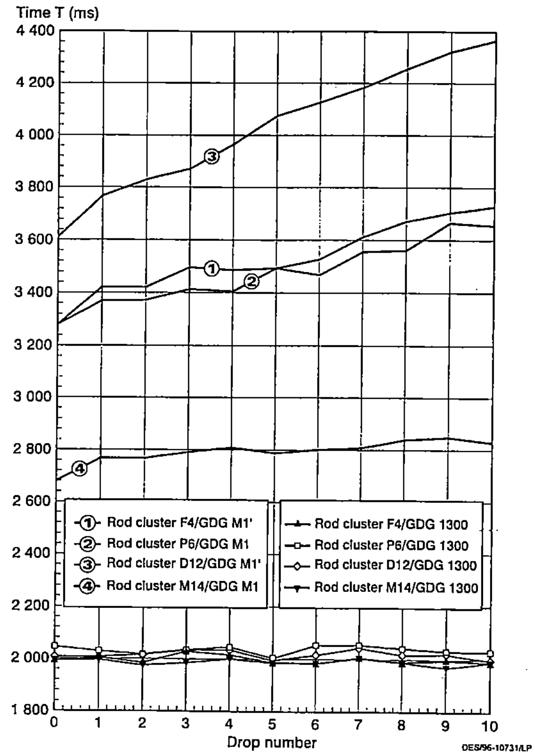


Figure 22
Chooz 1 - Drop time in cold leg
Test of 10 drops

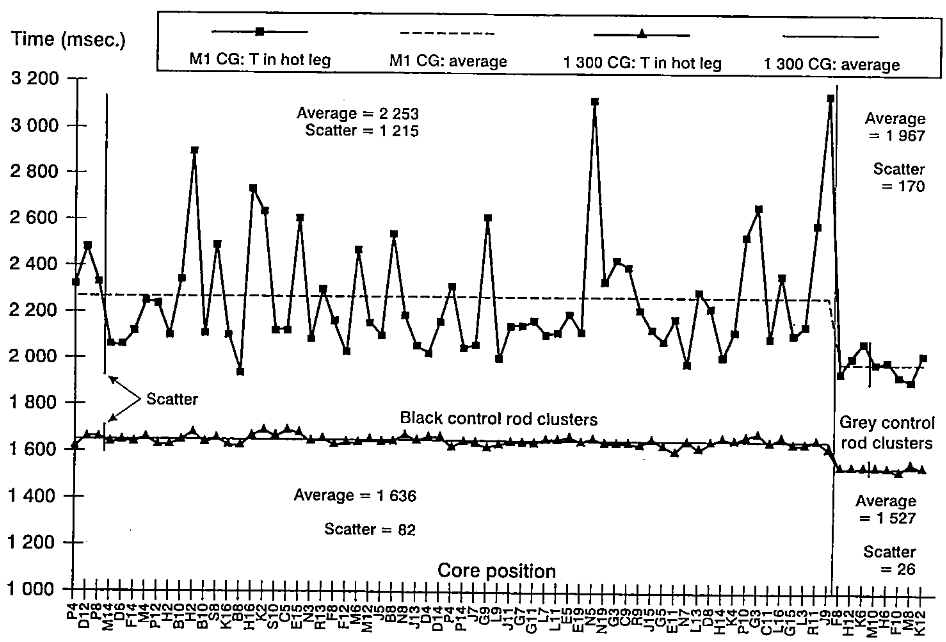


Figure 23
Chooz 1 - Drop time at start up and after installation
of 1300 MWE type control rod cluster guides

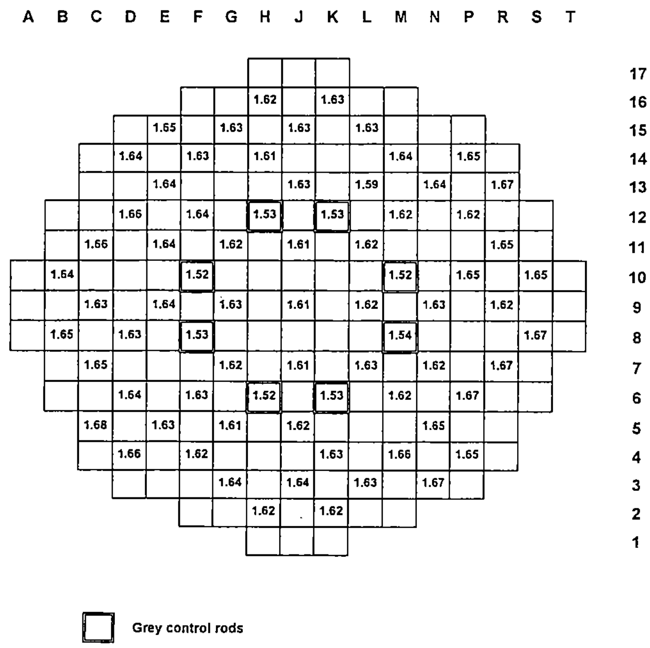


Figure 24
Map of Drop Times in Hot Leg

BANKS			T (s)
SA	C5	1	1.678
		2	1.688
		3	1.678
	E15	1	1.664
		2	1.662
		3	1.658
	N3	1	1.666
		2	1.662
		3	1.674
	R13	1	1.668
		2	1.664
		3	1.670
X	F8	1	1.524
		2	1.518
		3	1.506
	H12	1	1.522
		2	1.510
		3	1.502
	K6	1	1.538
		2	1.520
		3	1.510
	M10	1	1.538
		2	1.520
		3	1.506

Figure 25
Results of 3 Drops in Hot Leg under Hot Shutdown conditions

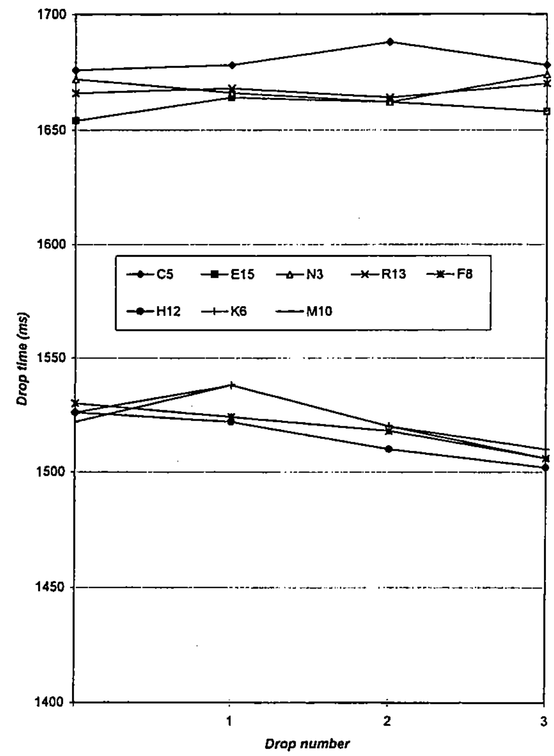


Figure 26
Results of 3 Drops in Hot Leg under Hot Shutdown conditions

Optimization Study of AP-600 Grey Control Rod Design

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ENUSA

W. R. Carlson, T. Marita
Westinghouse

Introduction

The Westinghouse-designed AP-600 advanced 600 MWe PWR design employs a core control concept, designated boron-free Mechanical Shim Reactivity Control System (MSHIM), that provides significantly enhanced load following capabilities. The core control strategy employs groups of low reactivity worth control rods, referred to as grey rods, that work in combination with high worth control rods to control both the reactivity changes and core power distribution during load following maneuvers, with no change in reactor coolant soluble boron concentration required during the maneuver. With this strategy, the reactor core is able to respond quickly to fast rate power demand changes and to rapidly "toggle" between load follow and base load operation during the cycle. Furthermore, the strategy can provide full load follow capability up to 95% of cycle life.

The grey control rods employed with the MSHIM strategy are needed both to help effect the desired reactivity changes during the load follow maneuver, and to compensate for the lower mean xenon reactivity worth that's present when the reactor is load following rather than base-loading. The reactivity worth of the grey control rods must be sufficiently large for these rods to be effective for load following, yet be low enough to minimize the perturbation produced on the core power distribution by the insertion or withdrawal of these rods.

The current grey control rod design uses four full-length absorber rodlets per control spider, with the remaining 20 rodlets of solid stainless steel (see Figure 1). While this design meets reasonably well its design objectives, certain improvements were nevertheless sought, motivating an optimization study to determine how the design could be improved. The improvements sought in the grey control rod design are the following:

- The current grey control rods have a reactivity worth that's somewhat too high compared to the worth that provides optimum load following performance. It's desirable that the worth be reduced 10 to 20%.
- The withdrawal of the control rod can also produce a local xenon transient near the tip of the control rod. The consequence is that the fuel near the tip is at greater risk of fuel rod damage from Pellet-Clad Mechanical Interaction (PCMI). The presence of a control rod in a fuel assembly strongly depresses the power in the fuel rods adjacent to the absorber rodlets. This leads to a "reconditioning" of these rods where the clad, under the influence of the fast fluence, creeps down and ultimately makes contact with the fuel pellet. When the control rod is later withdrawn, the power increases abruptly and the fuel pellet expands more quickly than the clad, overstressing the clad. The effect is aggravated at higher burnups due to the loss of ductility of the clad and the presence of certain corrosive fission products in the pellet-clad diametral gap.

This paper summarizes the optimization study that was performed to identify potential improvements in the current AP-600 grey control rod design to remedy the deficiencies discussed above. This study was performed by ENUSA for the Westinghouse Electric Corporation under the coordination of the D.T.N. (Agrupación Eléctrica para el Desarrollo Tecnológico Nuclear), as part of the participation effort of the Spanish electric utilities in the PWR passive reactor designs.

Proposal and optimization of alternative designs

To meet the first objective of reducing the control rod reactivity worth by 10 to 20% relative to the current grey control rod design, the neutron absorption in the absorber rods has to be decreased - i.e., the absorber loading has to be reduced relative to that in the current design. To meet the objective of minimizing the local power depression and potential PCMI concerns, as well as the localized transient xenon effects from the prolonged insertion of the grey control rods, the absorber needs to be as homogeneously distributed as possible within the fuel assembly. This helps avoid excessively large local power depressions and the "re-conditioning" of the fuel. However, when the absorber loading is reduced and is more evenly distributed (e.g., diluted), the absorber tends to deplete out more quickly and its useful lifetime in the reactor is reduced. Thus, the selection of the optimum design is forcibly a compromise between the desire for a more optimum control rod worth (e.g., reduced worth for better load following performance) and greater PCMI margins vs. the need to minimize the control rod lifetime loss (with the result of more frequent changeout of control rods, increased costs and potential plant downtime).

The current AP-600 grey control rod design uses four absorber rodlets of Ag-In-Cd (80, 15, and 5 w/o abundances, respectively), with the remaining 20 rods of solid stainless steel 304 (see Figure 1). The reactivity worth ideally should be reduced 10-20 % relative to that of the current grey control rod. In the current design, the reactivity worth is due approximately in equal parts to the Ag-In-Cd absorber rods and to the stainless steel rods. As such, the desired 10-20% reduction sought in the reactivity worth corresponds to the elimination of one of the Ag-In-Cd absorber rods, from four to three absorber rodlets in the control spider. However, as discussed further on, the required reduction factor for the optimum design is greater than 3/4, or 0.75, since the absorption strongly depends on the absorber geometry and dimensions because of the self-shielding effects in the absorber.

Various configurations and number of absorber rodlets per spider were looked at as possible alternatives to the current AP-600 grey control rod design. A configuration of eight absorber rodlets with the remaining 16 of stainless steel, was found to work best. Both axial and radial dilution/dispersion of the absorber were evaluated - these alternative designs are the following:

- Axial dilution of absorber ("sandwich" design):

In this design, the absorber dilution is accomplished by axially stacking in an alternating manner stainless steel segments with absorber discs of Ag-In-Cd in a sandwich-type design (see Figure 2). The number and height of the absorber discs are selected to meet several design objectives: a) provide the RCCA optimum reactivity worth; b) minimize the local power depressions in the fuel rods adjacent to the absorber and mitigate PCMI concerns; and c) pack the most amount of absorber in the control rod to maximize its useful design lifetime. The diameter of the absorber discs is maintained the same as that of the current grey control rod design. The solid stainless steel segments are of the same diameter as the absorber discs.

- Radial dilution of absorber ("wire" and annular designs):

Two alternative radial dilution schemes of the absorber were evaluated. In the first, the absorber diameter is reduced with the absorber maintained full length axially. In the second, an annular absorber disc design is used (also full length axially) with its outer diameter maintained the same as that of the current grey control rod design. To conserve the RCCA driveline weight for the first alternative, the Ag-In-Cd "wire" absorber is imbedded inside of a stainless steel bar segment having the same outer diameter as that of the current grey control rod design (see Figure 2). In the case of the annular absorber design, a stainless steel rodlet fills the central void of the annular absorber discs.

The notion of a dilution factor (axial or radial) was developed as an indicator of the quantity of absorber present in the proposed designs relative to the quantity present in the current control rodlet (full length axially). Consequently, the dilution factors, fx and fr, for the different designs considered are defined as follows:

$$\textit{Sandwich:} \quad f_x = \frac{\sum_{i}^{n} l^i_{Ag\text{-}In\text{-}Cd}}{L_{total}}$$

$$\textit{Wire:} \quad f_r = \frac{r^2_{wire}}{R^2_{std}}$$

$$\textit{Annular:} \quad f_r = \frac{(R^2_{std} - r^2_{interior})}{R^2_{std}}$$

R_{std} denotes the absorber radius of the current grey control rod design, r_{wire} is the absorber radius of the wire design, $r_{interior}$ the inner radius of the annular absorber design, $l_{Ag\text{-}In\text{-}Cd}$ the height of the absorber discs in the sandwich design, and L_{total} the total height of the absorber column in the current grey control rod. The dilution factor, fr, completely defines the geometry in the case of the radial variations. In the case of the sandwich design, an additional parameter is needed which is the number of absorber discs loaded per absorber rodlet, n. The axial dilution factor (fx) and number of absorber discs (n) uniquely defines the axial dimensions of the absorber discs and stainless steel bar segments.

A parametric study was performed to determine the geometrical dimensions that yield the desired optimum control reactivity worth. For the "wire" design, a value of fr = 1/6 (assuming 8 absorber rods per RCCA spider) is required, while for the annular design, the fr needed is less than 1/10. However, a value of fr less than 1/10 for the annular design represents too small a quantity of absorber in the RCCA spider, rendering this design impractical because of mechanical and fabrication considerations, and because of having too short a useful lifetime. Consequently, this design was discarded.

For the sandwich design, there is not a unique pair but rather an infinite combination of (fx,n) that yields the desired optimum reactivity worth. The presence of alternating Ag-In-Cd absorber discs and stainless steel segments (with less absorption power) in a sandwich-style axial stacking, produces an oscillatory-type axial variation in the flux and power profiles of the fuel rods adjacent to the absorber rodlet. The thermal flux is depressed at the elevations of the Ag-In-Cd absorber discs and peaks at the elevations of the stainless steel segments. The axial power oscillation depends on the height of the absorber discs and separation distance between the discs. The selection of the optimum axial dilution factor, fx, must necessarily be a compromise between the desire to pack the maximum amount of absorber material in the control rod (e.g., high value of fx) to maximize its useful lifetime in-reactor vs. the need to maintain the size of the absorber discs small enough to minimize the axial power oscillation peaks. However, as the disc size is reduced, the absorber is less able to self-shield itself and so depletes out faster, which penalizes the control rod lifetime. The compromise design uses an axial dilution factor of 0.3, with 68 absorber discs per control rodlet (assuming 8 absorber rods per RCCA spider).

As can be seen, the more even distribution of the absorber within the fuel assembly to minimize the local power depressions and fuel "re-conditioning" requires a reduction of the absorber loading in the control rod that's greater than the strict proportional reduction of 3/4 (e.g., removing one absorber rod per spider) indicated at the beginning of the paper. The Ag-In-Cd absorber material is quite opaque to neutrons so that the outer edge of the absorber is much more effective in capturing neutrons, while the inner region of the absorber disc remains as a reserve. Eventually, the outer region depletes out and the inner zone exposed to depletion. By distributing the absorber more evenly in the assembly, the surface to volume ratio of the absorber is increased, the absorber is less able to self-shield itself, and its reactivity worth is increased. Consequently, to maintain the desired absorption rate and reactivity worth, the absorber loading must be decreased by more than a simple proportional amount which has, however, the undesirable effect of reducing the control rod lifetime.

A study was performed to quantify the depletion behavior and reactivity loss of the alternative designs with burnup. Three absorber designs were evaluated and compared: a) the current AP-600 RCCA design (fr=fx=1), b) the wire design (fr = 1/6) and c) the sandwich design (fx= 1/3.33). As the absorber depletes out, the control rod reactivity worth decreases and reaches a point where the control rod no longer maintains the minimum reactivity worth required to be effective for load following and to provide the necessary return to power capability. At this point, the control rod must be replaced. The absorber depletion effects are much more consequential in the AP-600 reactor since the grey control rods can see prolonged operation at deep insertion levels in the core. Figure 3 compares the relative reactivity worth with burnup of the three designs above. As expected, the design that depletes out the least and is better able to maintain its reactivity worth is the current AP-600 RCCA design since the absorber is more concentrated and therefore better self-shielded. The wire and sandwich designs, on the other hand, are more evenly distributed (e.g., smeared) over the assembly to minimize the local rod power perturbations and to yield a lower reactivity worth, closer to the optimum worth. The undesirable effect of this more even distribution, however, is that these designs benefit less from the absorber self-shielding and, consequently, deplete out more quickly. The dilution factor is useful as a figure of merit to rank the alternative designs considered in terms of the useful in-reactor design lifetime that can be expected from each.

As regards the effect on the power of the fuel rods adjacent to the absorber rodlets, Figure 4 compares the unit assembly peaking factor (e.g., peak rod power) that results from withdrawing the control rod at various points during the depletion, with the assembly depleted with the control rod fully inserted. As expected, the peaking factor is worsened when the assembly is depleted with the control rod inserted. However, both the wire and sandwich designs show a smaller peaking factor increase/penalty than the current AP-600 control rod design during the depletion.

Calculational procedure

The control rod reactivity worth and absorber depletion calculations were made using two transport theory codes: PHOENIX-P, a two-dimensional code (Ref. 1) that uses the discrete ordinates method to solve the differential form of the transport equation; and KENO V.a, a three-dimensional criticality analysis code (Ref. 2) based on Monte Carlo theory that solves the integral transport equation. The KENO V.a code is included in the SCALE-4 code system provided by ORNL.

Figure 5 shows the calculational flow and codes involved. The SCALE-4 218 energy group ENDF/B-IV based cross-section library was used for the cross-section data. The SCALE-4 codes BONAMI-S (for nucleides with Bondarenko cross-section data) and NITAWL-S (Ref. 3) were used to calculate the resonance integrals for the silver (Ag) and indium (In) isotopes and to prepare a working cross-section library for XSDRN-PM. The latter, a 1-D discrete ordinates transport theory code, was then used to group-collapse the Ag, In, and cadmium (Cd) cross-sections for their direct use in KENO V.a. The XSDRN calculations were performed in 218 energy groups, with a fine spatial mesh, S-8 angular quadrature, and P-3 Legendre scattering order. The KENO calculations were made in 27 energy groups.

XSDRN-PM was also used to generate fine group Ag, In, and Cd cross-sections for a special version of the ANISN code, also a 1-D discrete ordinates transport theory code (Ref. 4). This code was modified to deplete out individually the Ag, In, and Cd isotopes at the individual absorber rodlet level of detail, and to generate effective absorber cross-sections, flux and volume homogenized over the absorber volume, to be used by PHOENIX-P. The ANISN absorber depletion calculations were performed in 40 energy groups, with a fine spatial mesh, S-8 angular quadrature, and P-3 scattering order. The control rod designs (e.g., current

AP-600 grey control rod, wire and sandwich alternative designs) were then depleted out in PHOENIX-P in 42 energy groups for a typical 17x17 PWR fuel assembly geometry.

To model the axially heterogeneous sandwich design in the 2-D PHOENIX-P code, energy and burnup-dependent axial flux self-shielding factors calculated by KENO V.a were first applied to the radially flux and volume homogenized (e.g., over absorber disc) Ag, In, and Cd cross-sections.

Conclusions

The optimization study shows that the "wire" and "sandwich" designs are the best alternative designs to the current AP-600 grey control rod design.

- Both the wire and sandwich designs meet the design objective of reducing the grey control rod reactivity worth by 10 to 20% compared to the current design.

- Both the wire and sandwich designs significantly reduce the undesirable effects of the prolonged insertion of the control rod on the rod powers of the fuel rods adjacent to the control rodlets, and thereby reduce PCMI concerns. The rod peaking factor increases that result from the withdrawal of the control rod after a prolonged insertion in the fuel assembly is reduced by half relative to the current design. The fuel rod burnup shadowing effect is also reduced for these alternative designs, as well as the local transient xenon effects.

- The control rod useful lifetime is, however, reduced for these alternative designs, with the reduction somewhat less severe for the sandwich design. With a longer lifetime, the preferred design is judged to be the sandwich design. However, the designs need to be further evaluated from the mechanical design and cost/fabricability standpoints before a final selection can be made.

While this study is specific to the AP-600 reactor design, the results also have relevancy to present-day commercial PWR plants. These optimized grey control rods can significantly improve the load following capability of the reactor, extend that capability further into the cycle, and reduce the need to change the reactor coolant soluble boron and thus the amount of water and radioactive waste that must be processed. The undesirable effects of prolonged control rod insertion are also minimized - namely the increased power peaking, fuel burnup shadowing, local transient xenon effects, and increased PCMI concerns.

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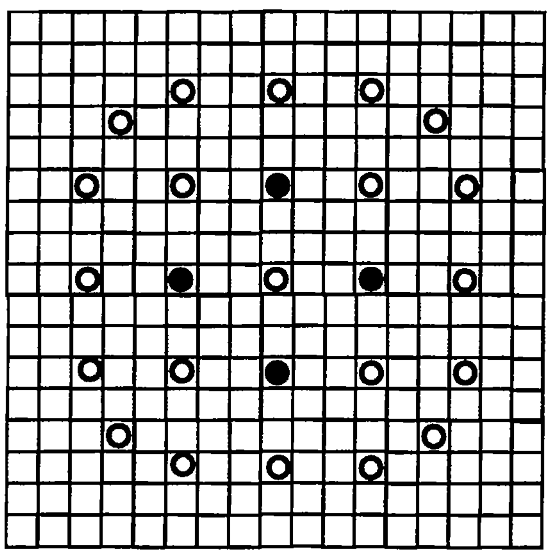

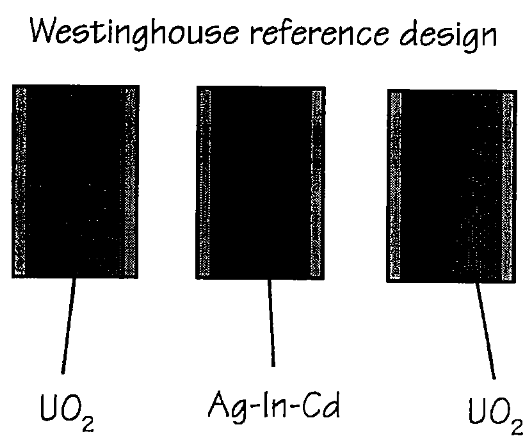

Current Base Grey RCCA
Design of Westinghouse

Figure 1

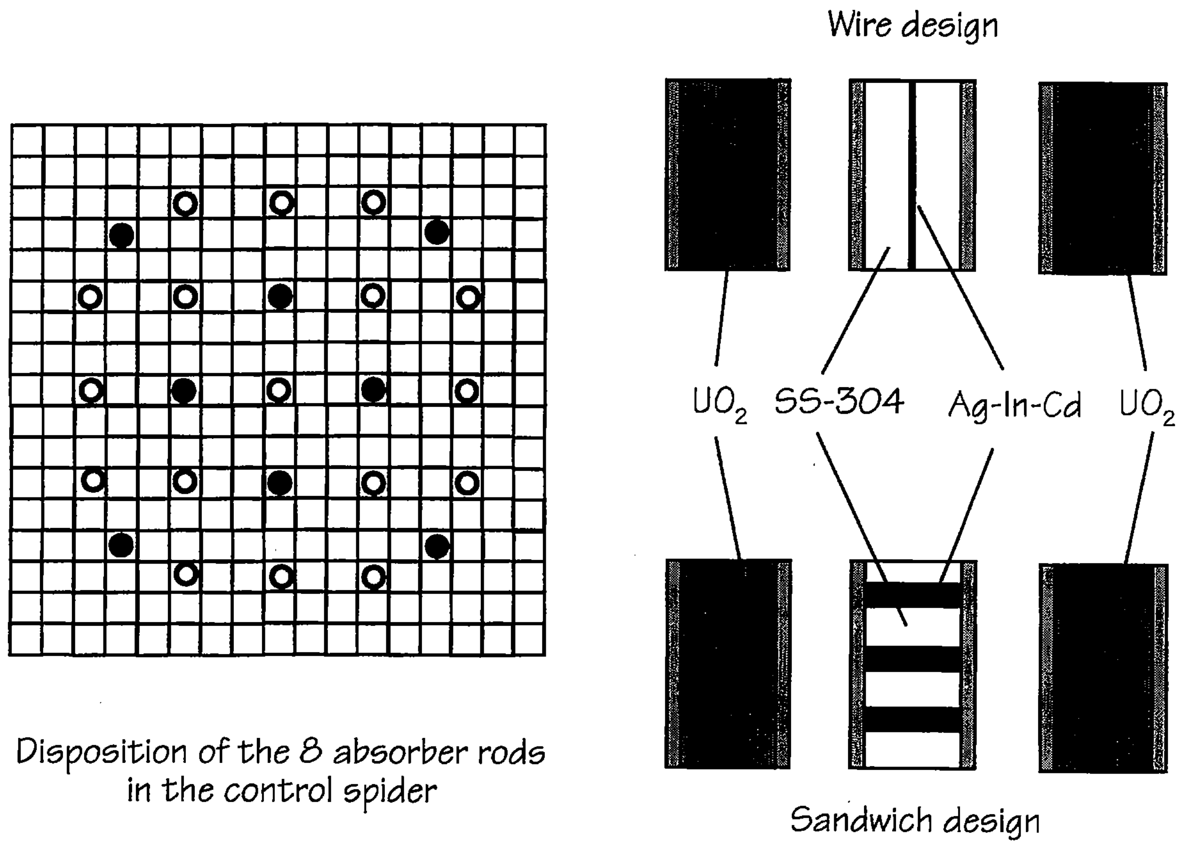

Figure 2

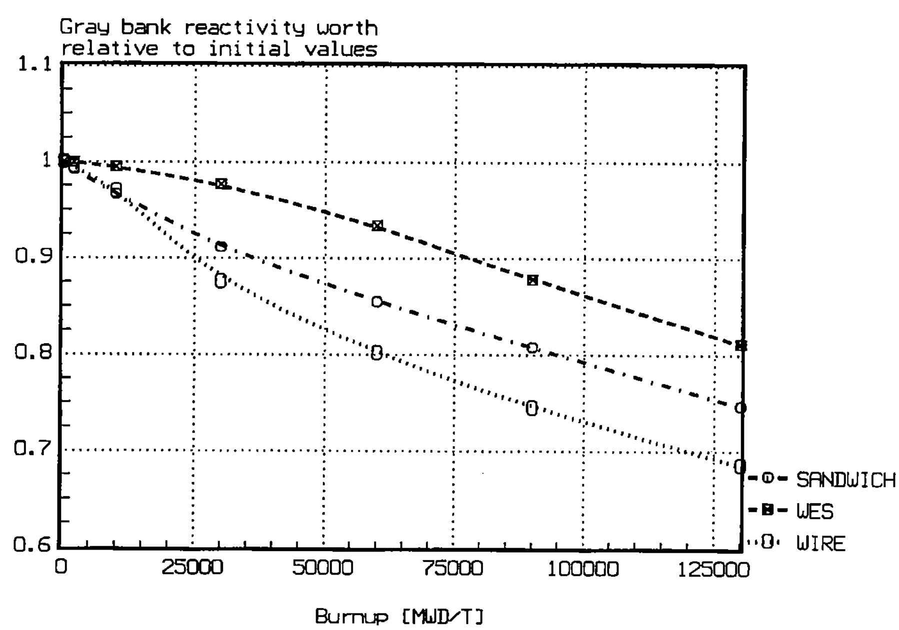

Figure 3
Evolution of control rod reactivity worth with burnup for current AP-600 grey control rod design and proposed alternative wire and sandwich designs

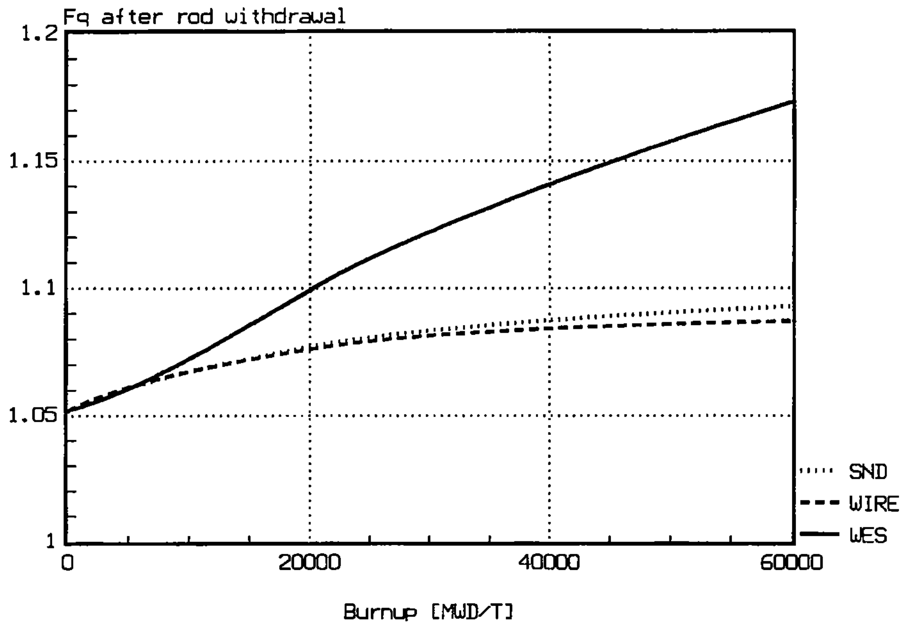

Figure 4
Assembly peak rod power after withdrawal of the control rod
depletion with the control rod inserted

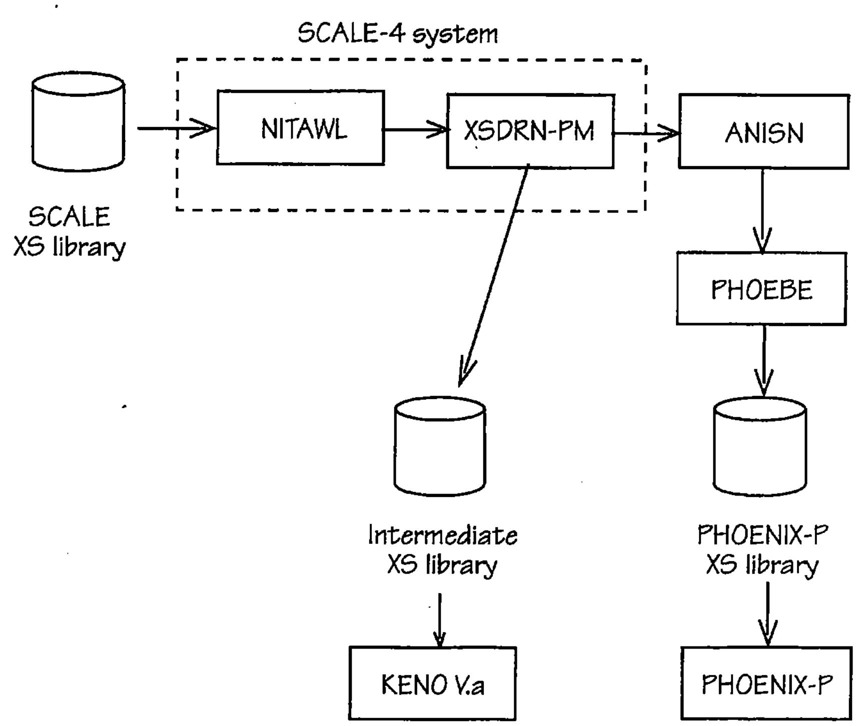

Figure 5

Belgian Operating Experience with RCCA Behaviour

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LABORELEC

1. Belgian power plants presentation

The Belgian Utility ELECTRABEL operates seven PWR nuclear reactors located on two sites (see Table 1). Three of them have been uprated (4 to 10% thermal uprating, 7 to 11% electrical uprating), partly through steam generators and low-pressure turbine rotors replacements.

These units are all operated as baseload plants, with all rod cluster control assemblies (RCCA) fully withdrawn, except one control bank. From an operational point of view, RCCAs have been submitted to a limited number of axial steps.

2. Characteristics of the initial RCCAs

The characteristics of the RCCAs delivered by the NSSS supplier with the first core are given in Table 2. Cladding material is made of AISI 304 stainless steel. Absorbing material is made of bars of Silver-Indium-Cadmium (AIC) alloy for all RCCAs, except in Doel 4 and Tihange 3, where mixed AIC + B4C control rods were delivered, with thicker cladding material.

All the delivered RCCAs have absorbing material with constant diameter from bottom to top.

3. RCCAs inspection history

Expected lifetime of RCCAs was declared by NSSS vendors to be in the range of 15 years. Two limiting phenomena were considered at that time :

- the possible depletion of the absorbing material that could reduce the efficiency of the control rods;
- the thermal creep and swelling of the AIC absorber material caused by own weight and acceleration forces due to the axial rod stepping, that could induce cracking of the tip of the clad material, this one being embritteled by the neutron flux.

A third phenomenon was discovered later on, i.e. flow induced fretting wear of the rodlets at the interface with the RCCA guides of the upper internals, mostly at the level of continuous guiding and guide cards and at the interface between the tip of the rodlets and the guide tubes of the fuel assemblies.

To help address this problem a non destructive inspection system was developed by Laborelec, financed by the Belgian nuclear plants and Westinghouse Europe, which acquired a license for foreign countries.

4. Development of the inspection technique

Research began with the compilation of the degradation shapes and types observed in control or shutdown banks. These appeared amenable to multifrequency Eddy Current testing of the type used for steam generator tubes. Since Eddy Current inspection is known to be able to detect wear and cracks reliably, as well as being easier to implement on site than ultrasonics, this was the method that was settled upon.

Two types of probes are used : bobbin coils for fast detection, and a set of eight pancake coils for accurate characterisation of the defects. Each bobbin coil surrounds a rodlet and provides an average value of the tube wall thickness. The pancake assembly measures the rodlet at eight different points located at 45∞ intervals around the tube section. Up to eight rodlets can be measured simultaneously with the bobbin coils, while one rodlet at the time can be characterised with the pancake coil assembly. This follows from the eight channel capability of the eddy current instrument used (Zetec's MIZ-18).

The signal interpretation being dependent on the geometry of the defects, very accurate laboratory reproductions of the sliding and fretting wear types were made using the spark erosion technique. Tests were carried out using a variety of probe windings and Eddy Current frequencies. The inspection parameters were chosen to ensure the highest accuracy, taking into account the influence of the absorber material.

Five main parameters were measured from the Eddy current signals :

- The defect location (bobbin coil).
- The estimated global penetration depth (bobbin coil).
- The axial defect lengths (pancake coil).
- The circumferential defect lengths (pancake coil).
- The remaining local wall thickness in front of each coil (pancake coil).

Laboratory tests based on artificial defects confirmed that the accuracy and the reproducibility were within less than 5 per cent margins.

Since 1996, the same inspection method has been qualified for nitrated and chromated RCCA.

To achieve practical implementation of the concept, the main challenge was to design the probes and probe-assembly to accommodate both the high-radiation levels (up to 10,000 R/h) and the underwater conditions (between 10 and 15 m below water pool level) whilst maintaining the +/- 5 per cent accuracy.

The coil material and wiring were subjected to a simulated working environment (pressurised borated water). Several designs were needed before the desired dimensional stability was finally obtained.

The probe assembly is equipped with a fuel top nozzle that allows it to be moved using the fuel handling equipment. The RCCA inspection system is immersed in the pool either within the fuel elevator or in an unused storage rack location. The data acquisition equipment (including the multi-frequency eddy current instrument and a special coil multiplexer) is installed within 20 m of the probe assembly.

The RCCA inspection system is designed and equipped in such a way that the inspections can be done leaving the fuel handling machine free for other operations during the inspection.

Four 90∞ rotations of the RCCA are needed to access the 24 rodlets, which are measured six at a time with the bobbin coils. Measurement is performed in a two-step sequence :

- Location of the degraded areas using the bobbin coils,
- Characterisation of the most severe defects using the pancake coil assembly.

The total sequence time can be summarised as follows :

- RCCA movement from its fuel assembly to the measurement stand : 10 to 12 min.
- Bobbin coil measurement of the 24 rods with 4 rotations of 90° : 20 to 25 min.
- Complementary defect characterisation with a pancake probe : 5 to 6 min for each rodlet with the most severe bobbin coil indications.
- RCCA movement from the stand to the original fuel assembly : 10 to 12 min.

Experience has shown that most of the RCCA measurements are carried out within 45 to 55 min, with fuel movements taking 20 to 25 min. For example, the inspection of the 52 RCCAs at Tihange 3 was performed during the normal fuel shuffling sequence without any effect on the duration of the refuelling operation.

Since February 1987, seven Belgian plants and two South African plants have been inspected by Laborelec with the RCCA inspection system. The technique was also applied by Westinghouse Europe to PWRs in France, Korea, Taiwan, Sweden, Switzerland, Brazil, Spain and Slovenia.

Fretting wear was observed with penetration depth ranging from 10 per cent to more than 90 per cent of the wall thickness. The profile of the fretting wear was clearly related to the card location and the RCCA service type (control or shutdown).

The most severe degradations were observed at the top of the RCCA used in shut down banks (close to the spider fixture) and in the rods close to the centre. There should be a correlation between the circumferential location of the wear and the edges of the slots within the card cage. The control RCCAs showed fretting wear along a wider length due to the stepping of the RCCA during reactor operation. Evidence of hairline cracking and swelling could be observed at some plants. The cracks were confirmed by visual examination using an immersed video camera.

5. Inspection strategy

Two inspection strategies were followed :

- In some units, partial inspection is performed during each outage on a sample of the RCCAs and from time to time on all control rods. This allows the identification of both the degradation phenomena and the degradation kinetics (wear or cracking). The interval between successive complete inspection is adjusted following the inspection results. The goal of this inspection strategy is to minimise the number of preventive replacements.

- In other units, complete inspections are performed, but not at every outage. The inspection interval (typically : 3 to 5 cycles) is adjusted following the inspection results and rejection criteria. The goal of this inspection strategy is to minimise inspection cost and outage downtime, with some risk of preventive replacement.

All inspection results are combined to describe the degradation phenomena and define the strategy by plant type.

6. Acceptance / rejection criteria

Immediate rejection criteria of inspected RCCAs were established by the RCCAs suppliers to prevent the following unacceptable situations :

- rodlet break due to excessive loss of material through fretting wear;
- rodlet break to excessive rodlet tip cracking;
- release of the B4C absorbing material due to cladding perforation.

These criteria were established considering inspection after each operating cycle. They only include a provision for just one cycle of operation for RCCAs, passing the rejection criteria.

The actual criteria used on site have to take into account the following additional elements :

- measurement uncertainty and errors;
- additional margin for delayed rejection, if the next inspection is planned after several cycles only. The additional margin is based on the degradation kinetic deduced from inspection results.

Actual rejection criteria used in Belgium are as follows :

- tip cracking : the RCCA is downgraded if ECT probes discover cracks in the 8 sectors which are examined. On the basis of an estimated propagation of the cracks, certain RCCA can be used for a limited number of cycles;
- fretting wear at Doel 4/Tihange 3 : the RCCA is downgraded in case of perforation or wear corresponding to a 30% loss of the circumferential material;
- fretting wear at Doel 4/Tihange 3 : the RCCA is downgraded in case of local wear of more than :
- at the guide cards : 75% for the shutdown control rod group, 68% for the control group D and the shutdown control rod group SD;
- at the continuous guiding : 68%.

7. Corrective actions and RCCA replacement plans

Several corrective actions were decided in the eighties to minimise or delay RCCA degradation for existing and new RCCAs, with a view to achieve the 15 years design lifetime as far as possible. These actions are :

- for initial RCCAs :
- limit the residence time of a single RCCA in a control position (i.e. with tip deeply inserted in the active length of the core), to delay tip cracking;
- apply an axial repositioning strategy for RCCAs subjected to heavy fretting wear.

The axial repositioning was applied in Doel 3/4 and Tihange 2/3 from 1988. A monthly repositioning policy on a 6 steps length is applied;

- for replacement RCCAs :
- use absorber with reduced diameter at the bottom part, to delay hard contact with the cladding;
- use RCCAs with surface treated cladding material (nitriding or chroming) for the RCCAs subject to fretting wear (17 x 17 plants).

Current replacement plans are dependent on the degradation phenomena. The Belgian units can be divided in three groups :

- Doel 1/2 and Tihange 1, where fretting wear is limited and were tip cracking was the rejection phenomena. All control rods were replaced after about 15 years, as expected, by new RCCAs having absorber with reduced diameter at the bottom part, but with standard (AISI 304) cladding;
- Doel 3 and Tihange 2, where all RCCAs suffered from accelerated fretting wear at the upper guide cards. Most RCCA had to be replaced after only 7 to 9 years of operation.

In Doel 3, standard replacement RCCAs (the only available on very short notice at that time) were used. In Tihange 2, nitrided replacement RCCAs were loaded.

Inspections are currently performed in Doel at a three years interval. Rejected RCCAs are replaced by nitrided ones. Inspections performed in both units on nitrided RCCAs have shown very low wear rate, indicating that the 15 years projected lifetime will be achieved.

In Doel 4 and Tihange 3, important fretting wear was found at the level of upper guide cards, but lower than in Doel 3 and Tihange 2.

A limited number of RCCAs were replaced in 1990 en 1994 by standard ones. Complete replacement of RCCAs by surface treated ones has been decided. It will be achieved in 1997 for Tihange 3 and in 1999 for Doel 4, this after 12 and 14 years of operation.

8. Incomplete RCCA insertion at the Doel 4 plant

When the Ringhals 4 incident was reported, based on the fuel vendor's evaluation, it was considered that this was an isolated and plant specific problem that did not affect other power stations. However, incidents in the past year, in particular involving 14 ft fuel assemblies, made the Doel 4 management reconsider their opinion. Consequently it was decided to perform a rod drop test at the end of cycle 11. The test was scheduled for 14 June 1996.

On 9 June 1996 a defect on the external grid caused the reactor to scram. During the analysis of the incident the position of the control rods was verified. Five control rods had a different position compared to the requested position and the deviation varied between 9 and 19 steps. (The control rods are listed in Table 3.) This deviation was followed and determined at different positions of the control rods and it was concluded that the deviation between the different control rods was less than 12 steps. Criticality was re-established and the unit was brought back to the grid. During the start-up operations the position of the control rods was readjusted. Two rods of subgroup C2 were lowered by 10 steps. The rod of subgroup D2 was lowered 2 steps. The next days a safety evaluation was performed to demonstrate the availability of shutdown margin and an elaborate test program established.

The safety evaluation considered three scenarios :

- all control rods which showed incomplete insertion are blocked out of the core (required shutdown margin, i.e. 3000 pcm met)
- on the assumption that all control rods in assemblies belonging to the same region will block before complete insertion, an evaluation was made to determine at what level they had to block with respect to the shutdown margin criterion (138 steps)
- same evaluation as the previous but including the assumption that the most reactive rod is out of the core (33 steps).

The test scheduled for the end of cycle was extended to get absolute certainty about the incomplete rod insertion. The purpose was to verify if the findings after the scram of 9 June could be confirmed and if the phenomenon was reproducible. Not only rod drop timing was performed. As the accuracy of the rod position indication system is only 7 steps, the position of the involved control rods was checked by counting the steps needed to reach the driveshaft lock out position.

In hot shutdown, a manual scram order was given; rod drop timing was measured twice. Finally a reactor scram signal was given to the two subgroups C2 and D2 with their control rods at 100 steps.

The analysis of the rod drop timing registrations was given to the fuel vendors FRAGEMA and ENUSA.

Main observations from the rod drop tests :

- 47 out of the 52 control rods always reached rod bottom position in the core.
- The same control rods as those involved at the scram of 9 June 1996 did not fully insert. These control rods are all in FRAGEMA fuel assemblies of the type AFA-XLR. They belong to the same reload batch (11D4) and they have a burn-up varying from 29400 to 32500 MWd/tU.
- At the first scram, control rod 45 did block at 71 steps.
- Control rod 45 did sometimes slowly continue to come down during further execution of the tests.

Table 3 gives a summary of the most relevant test results. Figure 1 gives the location of the concerned control rods in the core.

Following inspections were planned on the reactor before unloading the core :

- verification of the position of the control rods
- measurement of friction forces during insertion and extraction of a selected lot of control rods
- control of the thermal sleeve guide funnel
- measurement of the height of the assemblies in the core

After the unloading of the core, the inspections on the assemblies continued in the spent fuel pool, including :

- visual inspection of assemblies
- measurement of the deformation of a number of selected assemblies
- measurement of friction force during insertion and extraction of a control rod
- cross-checks with different assemblies and control rods.

The main conclusions regarding the friction forces were:

- The friction force in once burned assemblies was much less than in assemblies irradiated two or more cycles.
- A good correlation could be established between the measurements on reactor and in the spent fuel pit for once and twice burned fuel assemblies.
- For the assemblies involved with the incomplete insertion, the friction force in the dashpot exceeded 60 daN.

The main conclusions regarding the deformation of the assemblies were:

- Banana shaped deformation were preferential in the west to east direction with a maximum amplitude of 19 mm.
- S-shaped deformation were preferential in the north-south axis with a maximum amplitude (peak to peak) of 21 mm.

The cross-checks with different assemblies and control rods showed that the incomplete insertion was due to the deformation of the assemblies and that the control rods were not involved.

Good correlation could be shown by the vendors between :

- time to enter the dashpot and the friction force above the dashpot
- time to transit the dashpot and friction force in the dashpot
- measured rod drop time and calculated rod drop time
- deformation and droptime
- deformation and friction force.

On the other hand, no clear correlation could be observed between the friction force and the burn-up.

Based on their observations both fuel vendors made similar recommendations :

- only fresh or one-cycle irradiated fuel under control rod positions
- for the insertion of a control rod in the once burned fuel assemblies that will be loaded under a control rod, the observed friction force shall be less than 20 daN in the dashpot region and less than 15 daN in the region above the dashpot.

The operator has adopted the vendor's recommendations and loaded under control rod position 32 fresh fuel assemblies and 20 one-cycle burned assemblies. In addition we can mention that, compared to the previous cycle, 52 fresh fuel assemblies will be loaded. At cycle 11 only 32 fresh fuel assemblies were loaded. Another beneficial effect is obtained due to the replacement of the steam generators as the RC flow is increased by 10 % compared to the measured flow at cycle 11. Finally, the maximum burnup at EOC for assemblies under control rods will be less than 28000 MWd/tU.

All these corrective actions have been accepted by the Belgian authorities. As rod drop timing at BOC showed satisfactory results, they allowed to start up the plant with a permit to operate till mid-cycle. By that time the operator will either perform rod drop timing or demonstrate by other means safe continuous operation.

Unit	NSSS Supplier	Type	Thermal Power	Net Electrical Power	Fuel	First start-up
Doel 1	Westinghouse	PWR 2 loops	1192 MWth	392 MWe	14 x 14,8 ft	1974
Doel 2	Westinghouse	PWR 2 loops	1192 MWth	392 MWe	14 x 14,8 ft	1975
Tihange 1	Framatome	PWR 3 loops	2875 MWth	960 MWe	15 x 15,12 ft	1975
Doel 3	Framatome	PWR 3 loops	3053 MWth	1006 MWe	17 x 17,12 ft	1982
Tihange 2	Framatome	PWR 3 loops	2895 MWth	955 MWe	17 x 17,12 ft	1982
Doel 4	Westinghouse	PWR 3 loops	3000 MWth	1000 MWe	17 x 17,14 ft	1985
Tihange 3	Westinghouse	PWR 3 loops	3000 MWth	1015 MWe	17 x 17,14 ft	1985

Table 1
Belgian Nuclear Power Plants

Units	Number of rodlets	Number of RCCAs in reactor	Cladding material	Clad thickness (mm)	Clad outer diameter (mm)	Absorbing material
Doel 1/2	16	33	AISI 304	0.49	11.15	Ag-In-Cd (AIC)
Tihange 1	20	48	AISI 304	0,49	11,15	AIC
Doel 3/Tihange 2	24	48	AISI 304	0,47	9,68	AIC
Doel 4/Tihange 3	24	52	AISI 304	0,978	9,68	AIC + B4C

Table 2
Characteristics of the initial rod cluste control assemblies (RCCA)

Control rod	Subgroup	Core position	RPI after manual scram	RPI after test one	RPI after test two	RPI after scram 100 steps	RPI after test three
45	C2	H6	71	35	38	52	36
46	C2	F8	17	16	17	16	17
47	C2	H10	24	22	24	24	24
48	C2	K8	30	28	29	29	30
52	D2	L5	17	17	17	18	18

Table 3
Results of rod drop tests at end of cycle D4C11

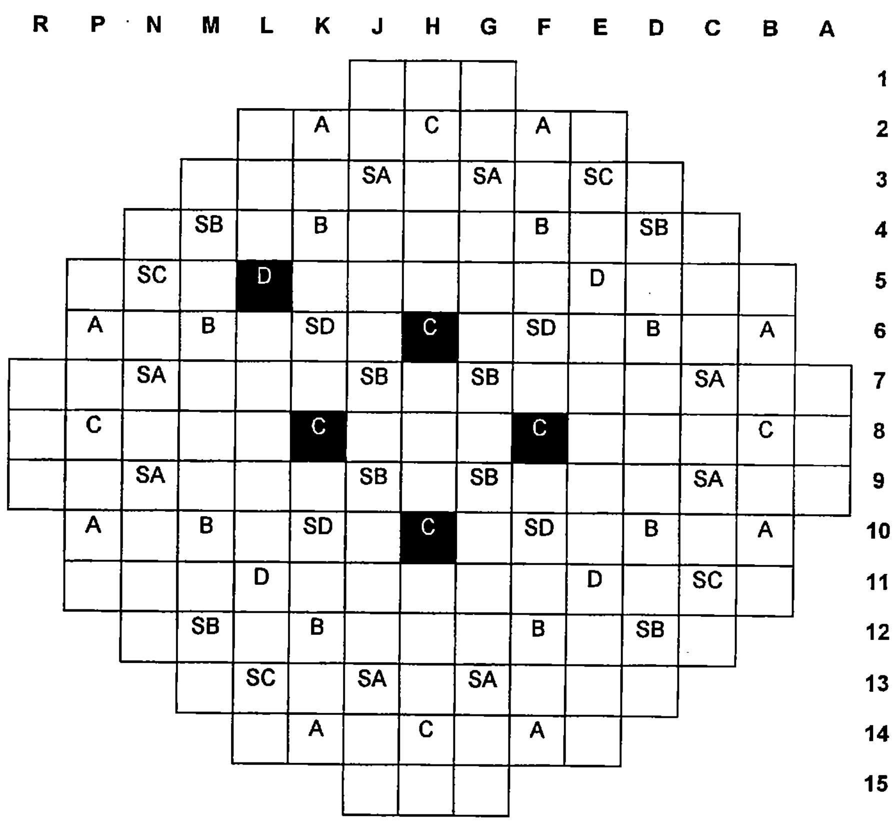


Figure 1
Location of the RCCAs witch incomplete insertion

TECHNICAL SUB-SESSION IV.A

ABB fuel Design and Development

G. Vesterlund
S. Helmersson
ABB

Current fuel designs have reached a high degree of maturity and sophistication and have also proved to be very reliable. High reliability is one of the most important properties and any design evolution step - small or large - have to be thoroughly tested, validated and proved not to compromise on this property. The development efforts to ensure a safe and reliable operation are therefore of utmost importance.

BWR Development

The original SVEA water cross concept was introduced in 1981 and set a new standard for BWR fuel economy. In 1986, ABB introduced the 10x10 lattice in the same SVEA channel to increase inherent margins and improve burnup capability. Substantial quantities of SVEA 10x10 were delivered from 1988. Reload batches have passed 40 MWd/kgU on the average and peak assembly burnups of 49 MWd/kgU have been achieved.

	SVEA-64	SVEA-100	SVEA-96
Assembly geometry	4x(4x4)	4x(5x5)	4x(5x5-1)
Number of rods per assembly	64	100	96
Overall assembly length, mm	4 398	4 398	4 481
Channel outside width mm	139,6	139,6	138,6
Fuel rod length, mm	3 887	3 989	4 147
Fuel rod outside diameter, mm	12.25	9.62	9.62
Pellet length, mm	10.44	8.19	8.19
Average linear fuel rating, kW/m	19	12	12.7
Cladding material	Zy-2	Zy-2	Zy-2
Cladding thickness, mm	0.80	0.63	0.63
Spacer grid material	Inconel	Inconel	Inconel
Average discharge burnup, MWd/kg U	40	40-50	40-50

Table 1. ABB BWR fuel design data.

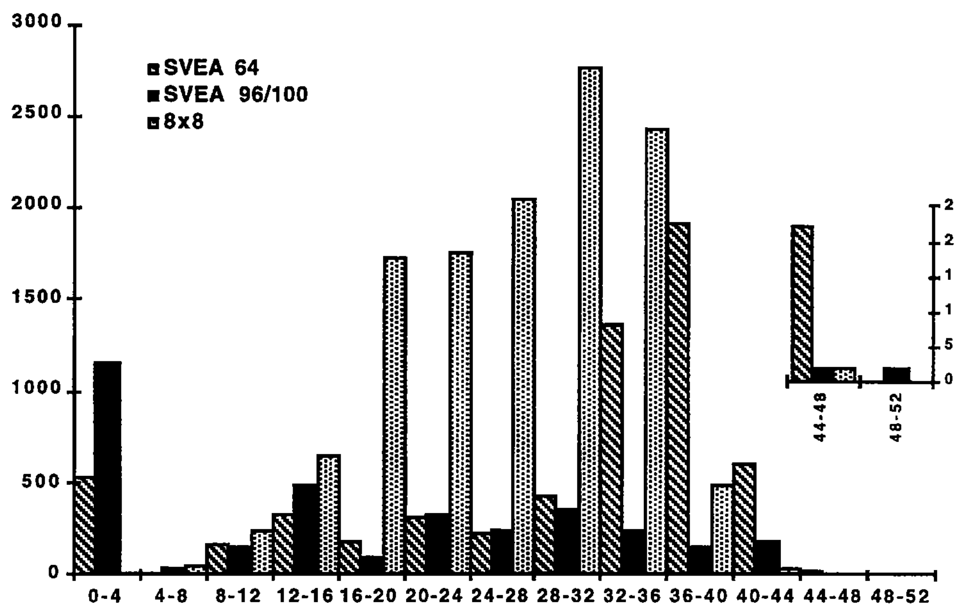


Figure 1. ABB BWR fuel burnup experience, as of October 1995.

A second generation of SVEA 10x10 fuel was introduced in 1994. By improving the spacer grid design, increasing the number of grids from 6 to 7, and by reducing the spacer grid distance in the upper part of the bundle, it was possible to improve the critical power by about 12% compared to the standard SVEA-96. Compared to the original 8x8 fuel the cumulative increase in critical power is about 40%.

Stability and transient behaviour are always important issues in BWR fuel development, particularly in view of increased thermal performance, burnup increases and power uprates of the plants. The experience with the SVEA fuel, both 8x8 and 10x10, is excellent. The methods are now validated against full cores of 10x10 fuel and behaving as predicted.

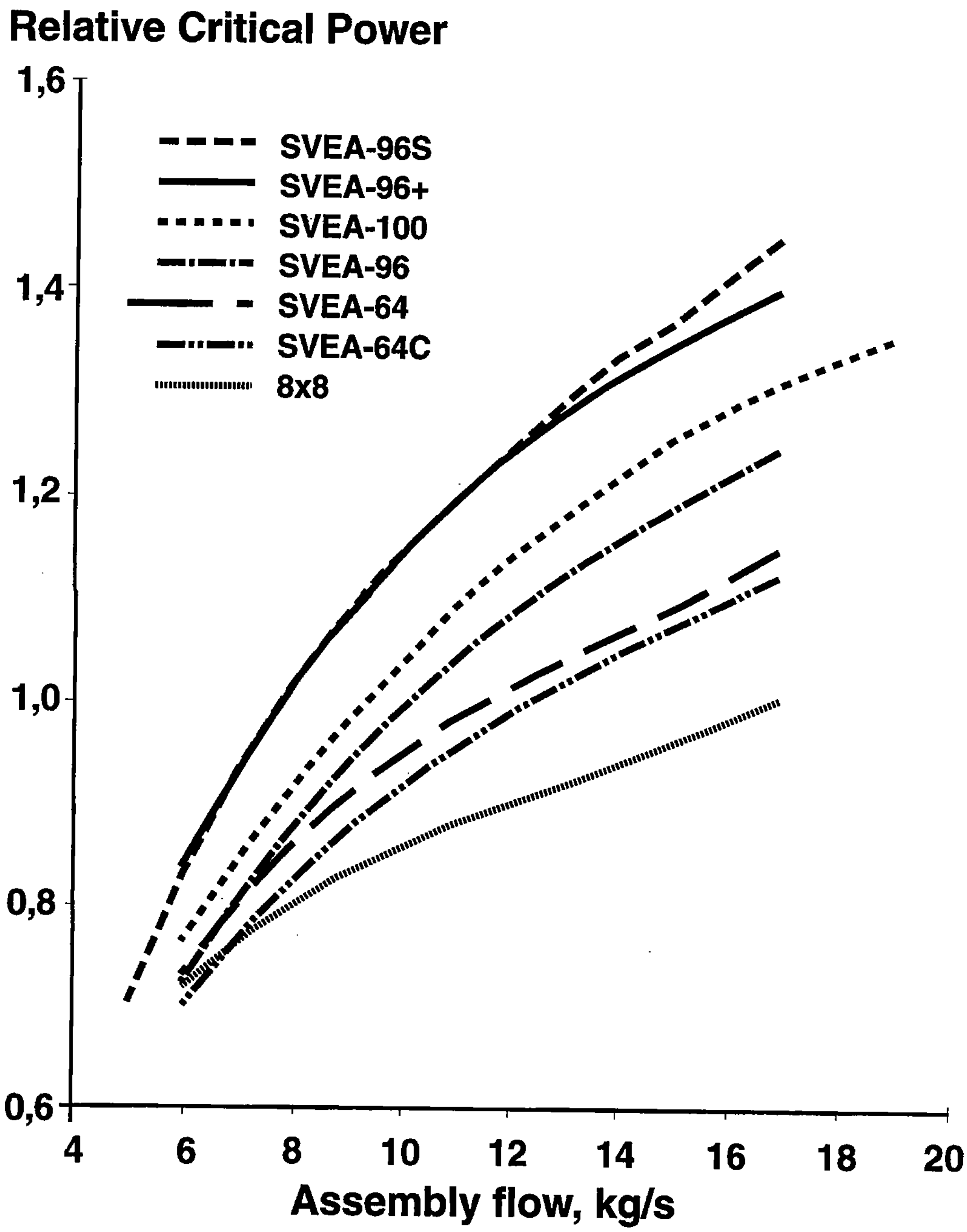
Fuel Reliability

BWR fuel of the SVEA design, featuring a water cross, was first introduced in 1980 and has since then revealed an excellent in-reactor behaviour. Currently, 51 SVEA rods are confirmed as failed, which corresponds to 2,3 per 105 rods per cycle - a number similar to recent experience of some other major BWR fuel vendors (Ref. 6,7). The total can be divided into 35 failures for the 8x8 lattice (SVEA-64), and 16 leakers in 10x10 fuel (SVEA-96 and SVEA-100).

Eight SVEA-64 liner rods have failed to date. One of these liner rods had a conventional sponge liner and the remainder were Sn-alloyed liner rods. The Sn alloyed liner concept was introduced in the mid 1980's to mitigate the secondary degradation effects that were postulated to occur on sponge liner fuel.

Primary causes and remedies

All except three of the failed rods have, to date, been carefully inspected in the fuel pool and altogether 18 rods have also been, or will shortly be, subjected to hot cell examinations at Studsvik. All examined rods could be assigned a definite cause of penetration: Apart from a single-event dryout incident,



only debris fretting and PCI have been observed. 31 rods revealed a debris fretting penetration and 13 failed due to PCI. For 10x10 fuel, only debris fretting was found (15 cases, 1 non inspected), and likewise for liner fuel (eight rods). No manufacture or materials related failures have been observed.

The PCI failures have occurred in non-liner SVEA-64 fuel in conjunction with withdrawal of deeply inserted control rods (CCC operation), i.e., large local power increases. Due to successively higher target burnups and fuel enrichments, the number and severity of such power ramps have increased, leading to a non-zero risk of failure even though the normal global PCI-mitigating operating rules were applied. The remedy consists of modified operating strategies for non-liner SVEA-64. No PCI failure has occurred after the implementation.

No PCI failure has occurred on ABB Atom liner fuel.

With the introduction of a debris filter, a SVEA-100 failure rate of less than 1 rod per 1 million rods per cycle is within reach.

Secondary degradation and remedies

After primary penetration, steam enters the rod and generates H2 due to radiolysis and oxidation of fuel and cladding. Where the H2/H2O partial pressure ratio is sufficiently high, i.e., beyond a certain distance from the primary penetration, local gross hydrogen pick-up can occur. Such hydride accumulations are readily identified by an eddy-current measurement (in the fuel pool or hot-cell), neutron radiography, or by metallography. Altogether, 18 failed SVEA rods have been, or will shortly be, examined at the Studsvik hot-cell facility. Detailed study of these and others together with poolside inspections/measurements reveal a secondary degradation process governed by gross local hydrogen pickup, leading to embrittlement of the Zircaloy tubing, in conjunction with fuel pellet oxidation/swelling and, if applicable, sponge liner corrosion.

Typically, the primary defect is very small as compared to the secondary failure, i.e., significant washout of non-volatile fission products and UO2 occur only as a consequence of significant secondary defecting.

The linear heat generation rate of a failed rod is one major parameter governing the rate of degradation as well as the onset and rate of fission product release and UO2 washout. A high LHGR will lead to a high temperature, i.e., exponentially enhanced chemical reaction and diffusion rates, as well as a high production of hydrogen in the pellet/cladding gap.

Observed degradation and release rates reveal no correlation to cladding manufacture: Types "LK0" and "LK1" represent a conventional processing route with annealing parameters around log A = -14 and an average secondary particle size of 0,1 μm or more. Type "LK2" has been subject to a late beta quench, resulting in an annealing parameter of log A = -16 and a considerably smaller average precipitate size.

Axial splits

Axial cracking is the most severe type of secondary failure since it often leads to significant UO2 wash out. Observations of axial secondary cracks have been made in both liner and non-lined fuel from ABB and other vendors (Refs. 1, 2, 3, 4). Single severely degraded rods are known to have recently forced at least 6 utilities to undertake unscheduled outages (Ref. 4).

A large circumferential cladding stress together with an incipient are required for an axial crack to propagate (Ref. 5). In ABB fuel, secondary hydride blisters as well as primary PCI cracks have functioned as incipients. This conforms with the experience of other vendors (Ref. 1,2, 3); in addition cladding manufacture flaws are reported to have acted as incipients (Ref. 3).

The development of long axial cracks correlates with the withdrawal of deep control rods (Refs. 2, 3). In particular, primary PCI incipients have a relatively high probability of propagating axially, since a large hoop stress is already present.

It is the experience of ABB that sponge liner fuel may give much more severe axial splits than non-liner fuel. This conclusion is based on in-pile experience and hot-cell examinations and is consistent with experience elsewhere (Refs. 2, 4, 9), but has also been contested by another vendor (Ref. 3, 8). The oxidation of the liner after the failure can give rise to a circumferential cladding stress capable of driving an axial split, as well as a high hydrogen content in the cladding, causing embrittlement.

Laboratory tests as well as in-pile experience and hot-cell examination of ABB Atom's Sn alloyed Zr-liner have demonstrated a post-failure corrosion behaviour fully comparable to Zircaloy-2 without liner. The Sn alloyed product is the standard liner since 1989: 22 reloads have to date been irradiated more than one cycle and 8 reloads have attained 5 cycles or more.

Altogether 7 axial splits longer than 15 cm (6 inch) have been observed on ABB Atom SVEA fuel. Of those, five are associated with primary PCI failures and one (the worst case) to sponge liner corrosion. No long splits have been observed in ABB 10x10 fuel.

We conclude that long splits have been practically eliminated by actions already taken, i.e., are no longer a concern for ABB fuel.

Circumferential breaks

Circumferential cracking leading to a break is a somewhat less severe mode of degradation, in terms of potential fuel washout. The occurrence of such cracks is generic to Zircaloy (Refs. 1, 2) and is demonstrated by post-irradiation examination to be a consequence of massive local pickup of hydrogen, occurring after water intrusion. Like blisters and bulges, circumferential breaks occur relatively far away from a primary: A large H2/H2O partial pressure ratio is a prerequisite for local hydrogen pickup.

Circumferential cracking occurs mainly at low burnup, presumably due to the open pellet-cladding gap leading to good communication of steam. We see no correlation to the detailed properties of the cladding, such as Zircaloy with large or small precipitates.

The first fuel rods with a remedy against local hydrogen pickup will be loaded in 1996.

PWR Development

ABB was the first PWR fuel supplier to develop an all-Zircaloy assembly, and the first to introduce a fully reconstitutable design, i. e., with an easily removable top nozzle. To improve cladding corrosion ABB has developed an optimized Zircaloy-4 which is the current standard cladding material in the US and in Europe for 17x17 fuel.

Large scale experience to burnups exceeding 50 MWd/kgU is already available. In the ABB design's for German 16x16 and 18x18 reactors ABB has developed a duplex clad material in which the outer surface of the Zr-4 cladding has a layer of another soft but extremely corrosion resistant Zr-alloy. This

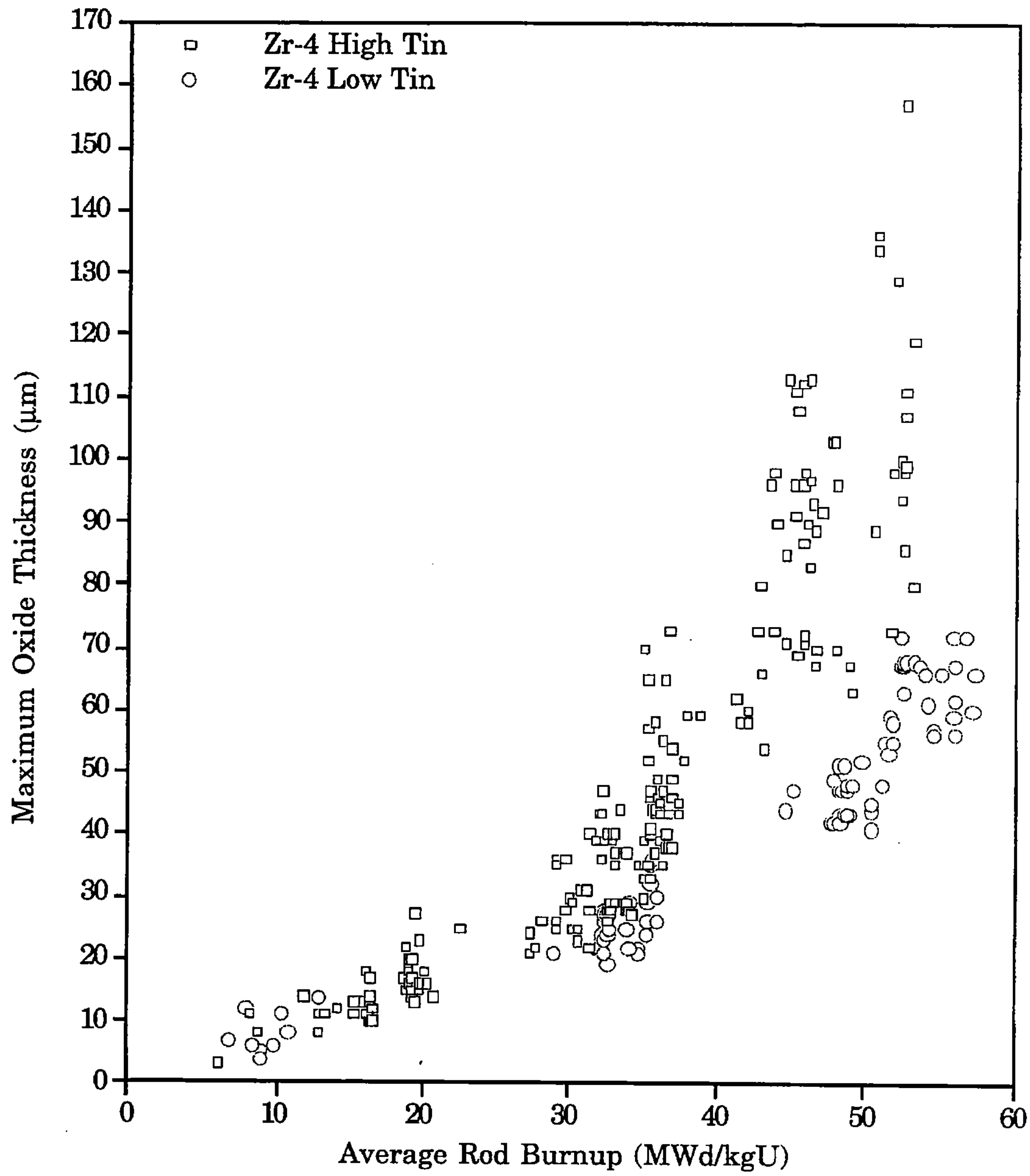


Figure 3. ABB data base on low Sn and standard Sn PWR clad corrosion.

alloy is characterised by its excellent corrosion resistance at higher temperatures, thus allowing higher burn-up to be achieved.

An other advanced feature of all, both ABB CENO and ABB Atom, designs, is the Guardian debris trapping bottom grid. This grid type was introduced in reload quantities in 1991 and has proven very successful.

The first demos of ABB PWR fuel for the European market were developed based on the very successful ABB CENO design and were introduced in 1991. Demos have since been delivered to 17x17 12 ft long cores, 17x17XL 14 feet long cores, 16x16 German reactors and 18x18 German reactors. Reloads have been delivered since 1994.

Parameter	16x16	18x18	17x17	17x17L
Assembly geometry	16x16-20	18x18-24	17x17-25	17x17-25
Number of rods per assembly	236	300	264	264
Overall assembly length, mm	4 827	4 827	4 053	4 790
Overall assembly width (mm)	229.6	229.6	214	214
Fuel rod length, mm	4 393	4 395	3 857	4 481
Fuel rod outer diameter, mm	10.75	9.50	9.50	9.50
Pellet diameter, mm	9.11	8.05	8.19	8.19
Average linear fuel rating, kW/m	20.7	16.7	18.7	17.8
Cladding material	Zy-4, Zy-2 or Duplex	Zy-4,Zy-2 or Duplex	Zy-4, Zy-2 or Duplex	Zy-4,Zy-2 or Duplex
Cladding thickness, mm	0.725	0.64	0.572	0.572
Spacer grid material	Zy, Inc	Zy, Inc	Zy, Inc	Zy, Inc
Average discharge burnup, MWd/kg U	50	50	50	50

Table 2. ABB PWR fuel design data for non-C-E- plants.

The development process

Identification of the customer requirements is of course the first step in the development process. The following is a list of the most common and general utility requirements ABB has found during recent surveys:

- Fuel reliability - zero defects and trouble-free utilization is the goal
- Economic fuel utilization - particularly driven by high back-end costs
- Operating flexibility - load-following and cycle length changes
- Safe Operation - thorough verification process
- Plant improvements - longer cycles, power uprating

Many other smaller requirements - sometimes utility specific - also have to be accounted for.

Conceptualization of the new or modified fuel design is next in the process. It requires a lot of human skill, experience and innovation that can hardly be automated. The design concept has already at this stage to be carefully scrutinized and reviewed, but the real validation process is just started.

The validation process comprises many steps and many areas to be covered. It has to be based on sophisticated testing and simulation methods, adequate for the new design. By necessity those methods have to be ahead of the design evolution and that requirement really invokes a development process in itself. The major validation areas to be covered in most cases are:

- Thermal hydraulic tests - to prove CPR and pressure drop performance for a wide parameter range (axial/radial power distribution, pressure, flow etc.)

Parameter	14x14	16x16	16x16 System 80
Assembly geometry	14x14-20	16x16-20	16x16-20
Number of rods per assembly	176	236	236
Overall assembly length, mm	3 994	4 491	4 528
Overall assembly width (mm)	206	207	207
Fuel rod length, mm	3 740	4 111	4 094
Fuel rod outer diameter, mm	11.18	9.70	9.70
Pellet diameter, mm	9.56	8.26	8.26
Average linear fuel rating, kW/m	21.2	17.7	17.7
Cladding material	Zy-4	Zy-4	Zy-4
Cladding thickness, mm	0.711	0.635	0.635
Spacer grid material	Zy-4	Zy-4	Zy-4
Average discharge burnup, MWd/kg U	46	52	54

Table 3. ABB C-E PWR fuel design data.

- Mechanical tests - to prove acceptable fretting behaviour and mechanical strength during normal and off-normal operation
- Material verification - In pile testing is the ultimate verification but out-of-pile testing gives the bulk of data
- Handling and service and transport - trouble free handling has to be verified
- Manufacturing and process qualification - to validate changes in the manufacturing procedures
- Design reviews - feedback of experience
- Licensing - compilation of design and safety reports for fuel and core design and the behaviour during normal operation and anticipated transients
- Lead Fuel Assemblies (LFAs) - installation of a few assemblies in one or more reactors
- LFA follow-up program - includes usually visual inspections and other pool-side inspections, but may also comprise hot cell examinations and PCI tests if necessary.

ABB has available world class resources for all steps in this validation process either in-house or in cooperation with partners and laboratories world-wide. This of course also includes the strong support given by the utilities in connection with in-reactor verification of new designs. This always involves extraordinary activities for licensing, core management and handling of pool-side inspections.

As examples to ABB's commitment a few examples can be mentioned:

ABB has in 1995 finalised a major upgrade of the FRIGG thermal-hydraulic test facility. The FRIGG loop now has a power of 15 MW and an advanced computerised control system. In FRIGG all kinds of thermal hydraulic testing of BWR fuel can be performed - CPR performance, pressure drop, stability, transients etc. A wide parameter range is also possible including both axial and radial variation of the power distribution within the fuel assembly. A unique feature is a 3D void-scan (tomography) that allows high-resolution void measurements.

The neutronics of a modern BWR core have become increasingly complicated over the years with higher enrichments and burnups and larger power mismatch between different assemblies, more burnable absorbers, MOX, etc. not to mention the complex patterns of unvoided and voided regions in the assembly. The full utilization of the improved fuel designs is coupled to requirements on higher precision and more details in the simulation, beginning with the cross section libraries and up to the 3D core dynamics simulators. ABB has since many years a broad and long term development program, which has resulted in a world class set of computer programs and design tools.

Conclusion

The conclusion is that modern LWR fuel has enabled the utilities to reduce the fuel cycle costs considerably through higher thermal performance, better materials, higher burnups and power uprates while maintaining or improving the safety margins. The development and follow-up work during many years has also provided a better understanding and insight into many key technical areas, that provides a basis for further improvements. For ABB this has also yielded a development process well suited to adapt to shifting demands from the customers.

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An Advanced 9x9 Fuel Design with Offset Water-Channel for D-Lattice BWR Plants

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Abstract

The current trend of the LWR fuel design is to increase the fuel burn-up in order to reduce the volume of spent fuels and the fuel cycle costs. This paper presents the advanced 9 X 9 BWR fuel design for the BWR plants with the so called D-lattice core. The 9 X 9 fuel with the offset large water-channel design is optimized for the overall nuclear characteristics of the D-lattice core with asymmetric configuration of the water gap between the fuel assemblies. The merit of the offset water-channel is crucial for the high burn-up design with higher average enrichment, because the maximum rod enrichment approaches the S wt% li~nit of the current Japanese LWR fuel production facilities and the transportation of fuel materials.

Nuclear design for the fuel average discharge exposure of 50 GWd/t was performed for a reference plant of 800 MWe class BWR. The core calculation demonstrated that the high enrichment design within the 5 wt% limit is feasible without loss of the design margin to the design basis transient or accident events as well as to the steady-state core operation. In comparison with a reference D-lattice fuel assembly without the water-channel offset, the spent fuel volume can be reduced more than 10 % due to the extension of the discharge exposure and the improved nuclear characteristics.

1. Introduction

During the past 20 years, the fuel assembly (FA) design for Japanese BWRs has been changed step by step in order to improve the operational reliability and the performance of the fuel. The maximum design burn-up of the current 8 X 8 FA is 50 GWd/t with an average burnup of around 40 GWd/t. The next generation fuel assembly is of the 9 X 9 type. The maximum design burn-up of the 9 X 9 FA is 55 GWd/t with an average burn-up of 45 GWd/t. Several lead fuel assemblies were inserted this year (1996) in a BWR of Tokyo Electric Power Co.

The NFI 9 X 9 type FA uses the square water-channel design which was developed jointly by NFI and Siemens in mid 1980's [1]. The 9X9 FA with an offset large water-channel has been developed for the reload fuel for D-lattice plants with asymmetric configuration of the water gap between the fuel assemblies [2]. In the D-lattice core, the gap between the fuel assemblies on the sides adjacent to the control blade is greater than the gap on the sides away from the control blade. The asymmetric structure of the Dlattice core has been employed for the old generation BWR plants, while the modem BWR plants use the so called C-lattice core with even gap size on the four sides of the fuel assembly. The difference in the structure of the C- and D -lattices is shown in Fig.1.

The offset water-channel design has been applied successfully for the reload fuel for a German BWR with D-lattice core. However, the merit of the offset water-channel can be exploited further with an extension of the burn-up, because the heterogeneity of the neutron flux distribution becomes more pronounced with the higher neutron absorption rate due to the higher fuel enrichment. The objective of this paper is to demonstrate the performance and safety aspects of the higher burn-up design with the offset water-channel for the D-lattice plants.

2. General Design Feature of NFI 9X9 BWR Fuel Assembly

The NFI 9 X 9 FA uses an optimized size large square water-channel. Figure 2 illustrates the FA structure and its main components. The 72 fuel rods in the NFI 9 X 9 FA provide 20~o less linear heat rate than the 60 fuel rods in the 8 X 8 FA. The fuel rod mechanical design is optimized for the high exposure, e.g. high density pellet (97% theoretical density), 1 MPa Helium pre-pressurization. As in the 8X8 FA, the zirconium-liner cladding is used for a remedy against the PCMI (pellet cladding mechanical interaction).

The increased number of fuel rods and reduced flow area increase the friction pressure drop. This pressure drop is completely compensated with the low pressure drop design of the improved spacer (ring type spacer) and the upper tie-plate. The pressure drop of the NFI 9 X 9 FA h as been proven by thermal-hydraulic testing including the channel stability test [3]. The ring-type spacer provides a high critical power or large MCPR (minimum critical power ratio) margin and its performance has been demonstrated by testing [4].

3. High Burn-up Nuclear Design

3.1. Enrichment Design

During power operation of BWRs, the neutron moderation rate in the non-boiling water between the fuel assemblies becomes comparable with the reducing moderation rate due to boiling in the fuel assembly. As a result, a skewed power generation prevails in the fuel assembly and large local power peaking factors (LPF) cluster on the comer and side rods adjacent to the gap, particularly on the sides adjacent to the wide gap when the control blade is withdrawn. The skewed power distribution becomes more pronounced with higher enrichment, because the neutron absorption rate and power depression in the central fuel rods increase with higher enrichment. The large LPF penalty must be compensated with lower enrichment in the rods located at the assembly periphery and higher enrichment in the other rods. Thus, the maximum rod enrichment approaches the 5 wt% limit of the current Japanese LWR fuel production facilities and the transportation of fuel materials. The 5 wt% limit is more or less common to all Japanese LWR fuel industries due to the past standard practice for the low enrichment LWR fuel production and licensing.

The offset square water-channel design effectively compensates for the asymmetric moderator distribution in the D-lattice core, thereby improving the overall nuclear characteristics of the Dlattice core. As a result, the average enrichment of the fuel assembly can be increased without large penalty of LPF for the high bum-up design, since the LPF is lowered with more uniform moderator distribution. Such merit is crucial for the high bum-up design with higher average enrichment for the D-lattice fuel assembly.

The enrichment design with the offset water-channel is shown in Fig.3(a) for the average discharge bum-up of 45 GWd/t. In order to discuss the merit of the offset water-channel, another enrichment design was performed using the non-offset central water-channel. This design is shown in Fig.3(b). The third design case is shown in Fig.3(c) for the average discharge bum-up of 50 GWd/t. An average enrichment was increased from 3.8 wt% to 4.1wt% in order to increase the average bum-up from 45 to 50 GWd/t. In the following description these three design cases are called A45, S45, and A50.

For all enrichment designs, the maximum enrichment of the rod was limited to 4.9 wt%. However, the high enrichment rods of both designs of A45 and A50 are located closer to the control blade in comparison with the S45 design in which the high enrichment rods are shifted away from the control blade. The effect of this difference of the enrichment distribution on the overall nuclear characteristics of the D-lattice fuel assembly is discussed later. For all designs, the enrichment is axially uniform except for a few rods which have an axial zoning of the enrichment with higher enrichment in the lower part and the lower enrichment in the upper part. The axial zoning was used in order to make a fine tuning of the average enrichment. Natural uranium blanket was used for the top and bottom ends of the active length of around 3.7 m.

3.2. Local Peaking Factor

A comparison of the local peaking factor (LPF) as a function of bum-up is shown in Fig.4. It can be seen that the offset water-channel design provides less LPF than the design without water-channel offset. Even in the case of the higher enrichment design for the 50 GWd/t discharge exposure, the maximum LPF is retained at the same level as the lower bumup design without water-channel offset.

4. Equilibrium Core Design

4.1. Core Design Analysis

The reference plant in this study is a 800 MWe class BWR with 548 fuel assemblies in the core. An equilibrium core of the cycle length of 10 GWd/t was analyzed for the three FA design cases: A45, S45 and A50. The fuel loading pattem with 124 reload FA's, which is commonly used for the case A45 and case S45, is shown in Fig.5(a), while the loading pattem with 112 reload FA's for the case A50 is shown in Fig.5(b). Both fuel loading pattems are of the low leakage pattem in which the highly exposed fuel assemblies were placed preferentially at the core periphery. Table 1 summarizes the core calculation results. All design cases satisfy the design criteria conceming MLHGR, MCPR and shutdown margin.

4.2. Reactivity Performance

As shown in Table 1, the cycle exposure which was determined by the equilibrium core calculation of each core design case is close to the nominal cycle length of 10GWd/t. However, each calculated cycle exposure was normalized to the exact nominal cycle length in order to evaluate the reactivity performance. A linear reactivity relationship was assumed for a small change of number of reload FA's of the given fuel design and enrichment so that the summation of the cycle and the average discharge exposures is approximately constant. Thus, the nominal discharge exposure to the cycle length of 10GWd/t was calculated to be 43.9, 45.0 and 49.6 GWcUt for the design cases, S45, A45, and A50, respectively. Since the average enrichment of cases A45 and S45 is the same, the difference of around 1.0 GWd/t is the discharge exposure gain with the water-channel offset. The gain is equivalent to a decrease of 2% of the spent fuel volume. Moreover, if the bum-up is further increased to 50 GWd/t with the waterchannel offset design, the spent fuel volume decreases about 11%.

4.3. Shutdown Margin

Figure 6 shows the shutdown margin for the three design cases. Even for the higher bum-up design (A50) a large shutdown margin is provided by the offset water-channel. It is discussed later that the improvement of the shutdown margin can be attributed to the increased control rod worth by the offset water-channel. Although the use of higher gadolinia content and the large number of gadolinia bearing rods also improves the shutdown margin, but a loss of reactivity is caused due to the residual absorber effect. It is an important point to be noted that the shutdown margin was improved without increasing the gadolinia content. For all design cases including the case of higher fuel enrichment (A50), the maximum gadolinia content was limited to 5 wt%, since this content is just optimized to the cycle length of 10 GWd/t.

5. Discussion

5.1. Reactivity Characteristics with Water-Channel Offset

The reactivity gain is defined as the difference of the infinite multiplication factors (kinfinity) obtained from the assembly calculation of the two design cases: A45 and S45. The reactivity gain with the water-channel offset is shown in Fig. 7. Note that the same average enrichment as well as the same number of the gadolinia bearing rods and the same gadolinia content are used for both design cases. However, the difference of the water-channel design affects the neutron absorption rate in the absorber rods, hence a superficial reactivity gain occurs in the low exposure range until the burnout of the absorber. Excluding such absorber effect, it turns out that the net reactivity gain amounts to 0.25 % Δ k in the exposure range corresponding to the core average exposure from the beginning of cycle (BOC) to the end of cycle (EOC).

The fuel reactivity characteristics as a function of the core operating state variables are of prime importance for the reactor core design in order to manage all BWR core operating conditions: the core shutdown, the steady-state power operation and the design basis abnormal transients or accidents. The fuel reactivity was obtained by the two-dimensional fuel assembly calculation as a function of the state variables: the void fraction, the fuel temperature and the control rod insertion.

The void reactivity coefficients were calculated for both water-channel designs and they are compared in Fig.8. The magnitude of the void reactivity coefficient is reduced around 3% (about 3% less negative) by the offset water-channel. Even for the higher enrichment design (A50), the void reactivity coefficient is retained at the same level as the lower enrichment design without the water-channel offset. Although not shown in the figure, it was confirmed by the calculation that the Doppler feedback reactivity as a function of the fuel temperature does not change with the water channel offset.

The difference of the control rod worth with and without the offset of the water-channel is shown in Fig.9. The control rod worth is increased with the water-channel offset over the wide exposure range except for a short period due to the gadolinia absorber effect. The absorber effect occurs because the absorber rods away from the control blade burn more rapidly in the case of the water-channel offset. Excluding such absorber effect, the control rod worth is increased because the high enrichment rods are located closer to the control blade (cf. Fig 3).

5.2. High Burn-up Design

The core calculation result demonstrated the steady-state core performance of the high burnup design with the water-channel offset, i.e., the core performance parameters of the 50GWd/t high burn-up design are well within the design limits. The extension of the average discharge exposure from 45 to 50 GWd/t is feasible, because the offset water-channel design allows the higher enrichment design without much increase of the local peaking factor. It was also shown that the shutdown margin of the high burn-up design is retained at a large value without increasing the gadolinia content, since the water-channel offset increases the control rod worth.

5.3. Safety Aspects of Water-Channel Offset

The calculated maximum FA burn-up of each design case is shown in Table 1. Even for the high average burn-up design of 50 GWd/t, the maximum burn-up is only 52 GWd/t against the maximum design burn-up of 55 GWdlt. The power histogram of the fuel pellet as a function of the local exposure at the EOC is shown in Fig.10. Since the high moderator to fuel ratio of the NFI 9X9 FA design provides a soft neutron spectrum, the fuel rod power decreases rapidly at high exposure without much build up of the plutonium isotopes. The fuel rod power at high exposure is also reduced owing to the low leakage core loading pattern in which the highly exposed fuel assemblies are placed at the core periphery. As a result, the core fraction of the fuel rods with local power greater than 20 kW/m becomes very small at higher burn-up. If such effect of the high burn-up is properly taken into account, the consequence of the design basis accident such as RIA (reactivity initiated accidents) will not become a practical issue for the high burn-up fuel.

The core stability is strongly dependent on the void feedback. The core stability of the 9X9 FA with water-channel design has been demonstrated by the reactor core stability measurement and it has been shown that the main contributor to the stability was the less negative void reactivity coefficient of the 9X9 FA [5]. The less negative void reactivity coefficient also increases the MCPR margin against design basis transient events such as a generator trip, since the neutron kinetics with core pressurization events are strongly dependent on the core void reactivity coefficient. Therefore, it is an important merit of the offset water-channel that the void coefficient of the high enrichment design is retained at the same level of the lower enrichment design with the non-offset water-channel.

6. Conclusion

Nuclear design of the high burn-up 9 X 9 fuel assembly with offset square water-channel for the D-lattice core was performed and the reactivity characteristics were discussed in comparison with a reference D-lattice fuel assembly design without the water-channel offset. It was shown by the core design analysis that the advanced 9 X 9 fuel assembly design with the water-channel offset allows an extension of the average discharge exposure up to 50 GWd/t without loss of the design margin to the design basis transients or accident events as well as to the steady-state core operation. It was also shown that in comparison with the reference Dlattice fuel assembly without the water-channel offset, the spent fuel volume can be reduced more than 10 % due to the extension of the discharge exposure and the improved nuclear characteristics.

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Core Design Case	S45	A45	A50
Number of Reload FA (Reload Fraction of 548 FA's)	124 (23%)	124 (23%)	112 (20%)
Cycle Exposure (GWd/t)	9.95	10.15	10.11
MLHGR* (KW/m)	37.8	36.9	39.0
MCPR**	1.42	1.44	1.40
Shutdown Margin (%Δk)	1.21	1.69	1.55
Maximum FA Exposure*** (GWd/t)	48.0	49.0	52.1
Average FA Discharge Exposure (GWd/t) Normalized to Cycle Exposure of 10 GWd/t	43.9	45.0	49.6

* Maximum Linear Heat Generating Rate; Design Limit: 44.0 kW/m
** Minimum Critical Power Ratio; Design Limit: ~1.30
*** Design Limit: 55 GWd/t

Table 1. Calculated Results of Equilibrium Core Performance

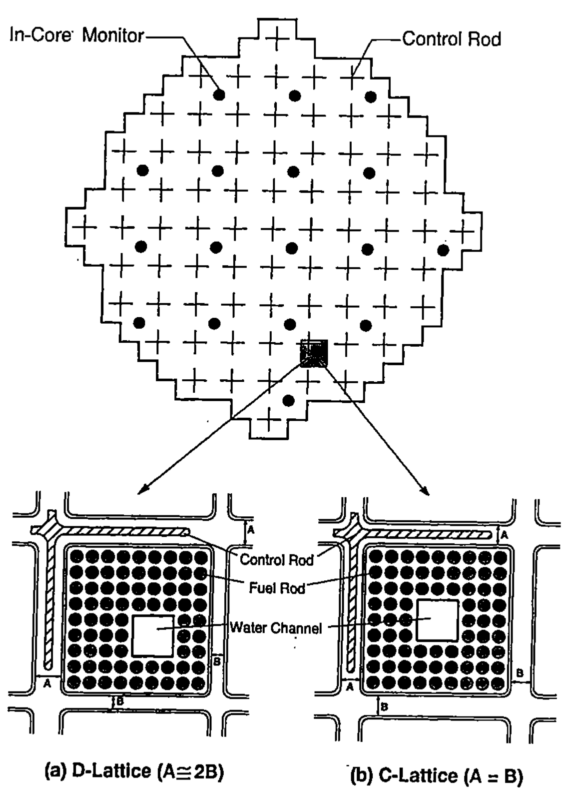


Figure 1. BWR Core Lattice

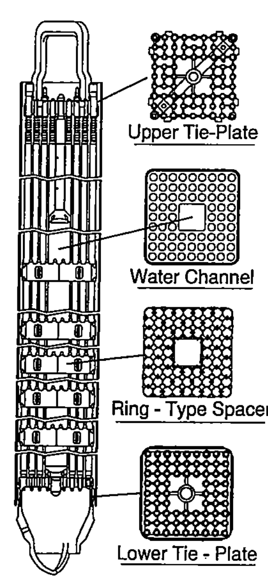


Figure 2. NFI 9X9 Fuel Assembly

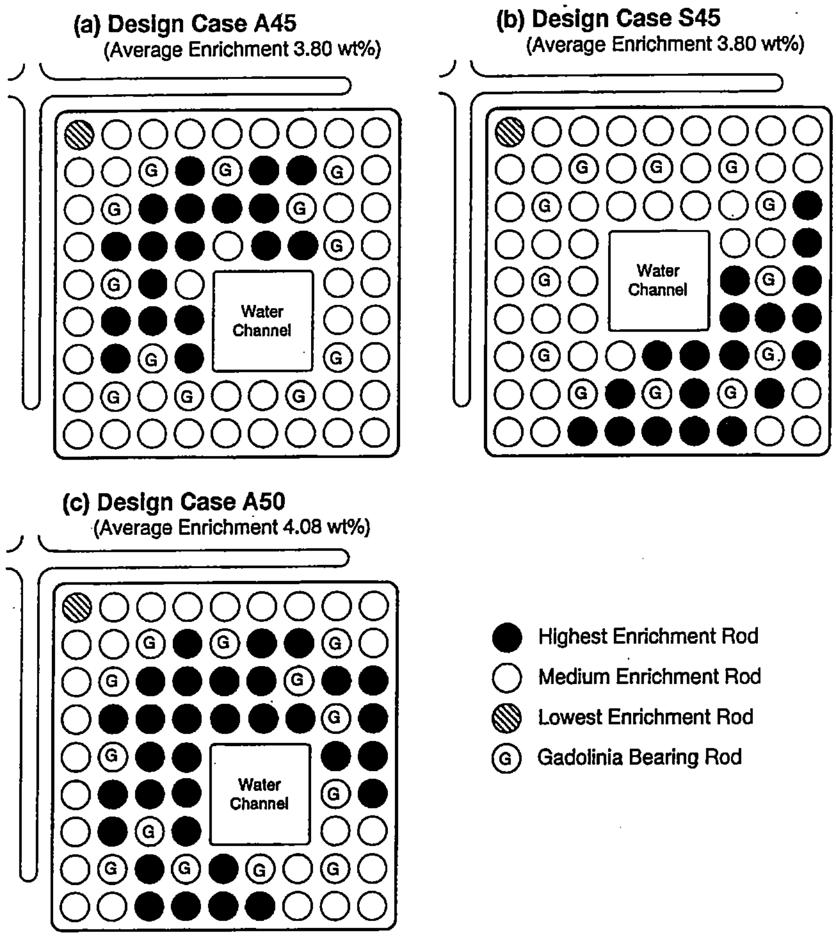


Figure 3. Enrichment Distribution

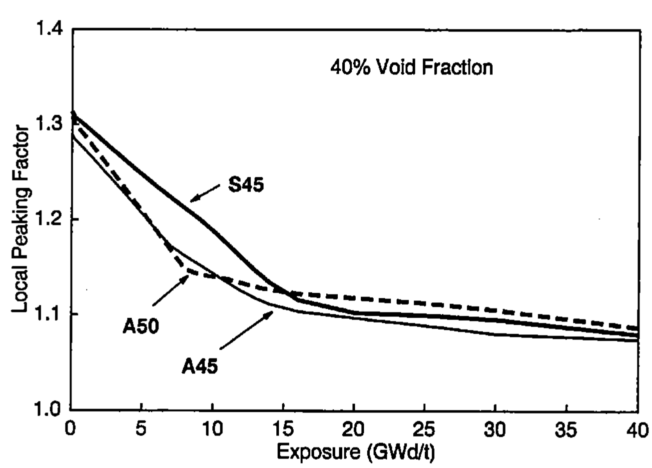


Figure 4. Local Peaking Factor

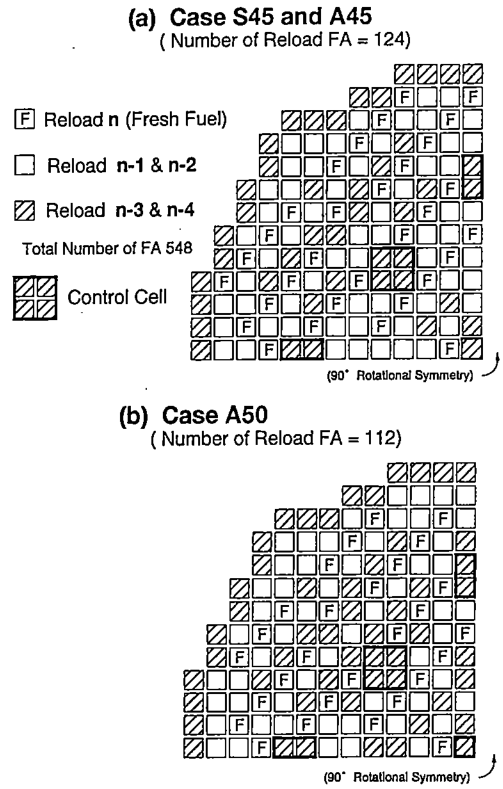


Figure 5. Equilibrium Core Loading Pattern

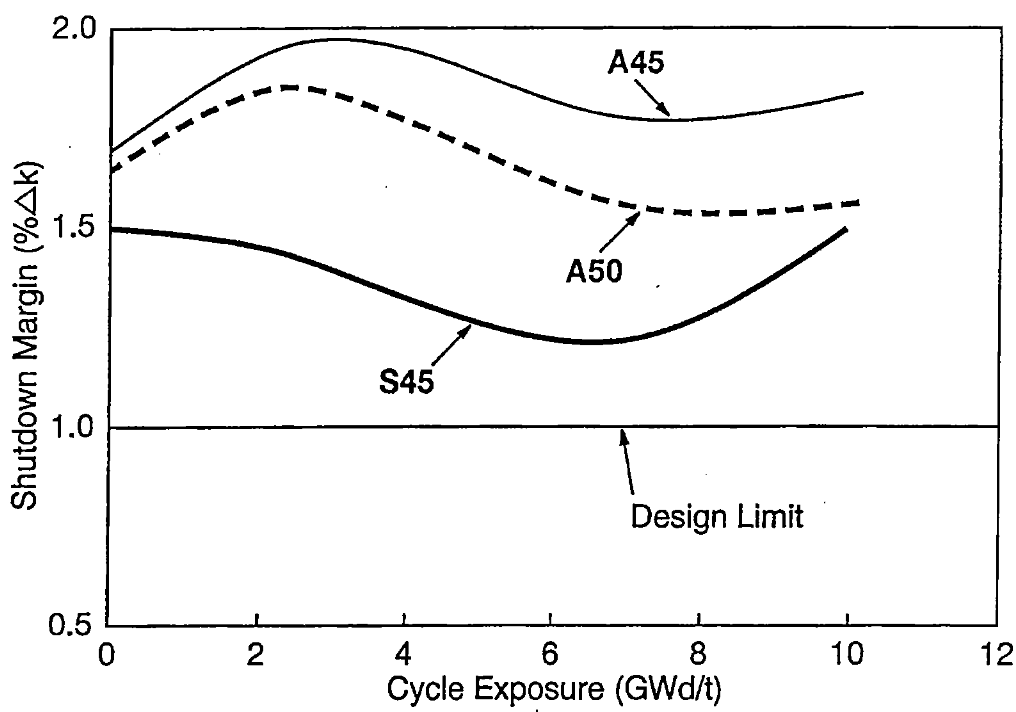


Figure 6. Shutdown Margin

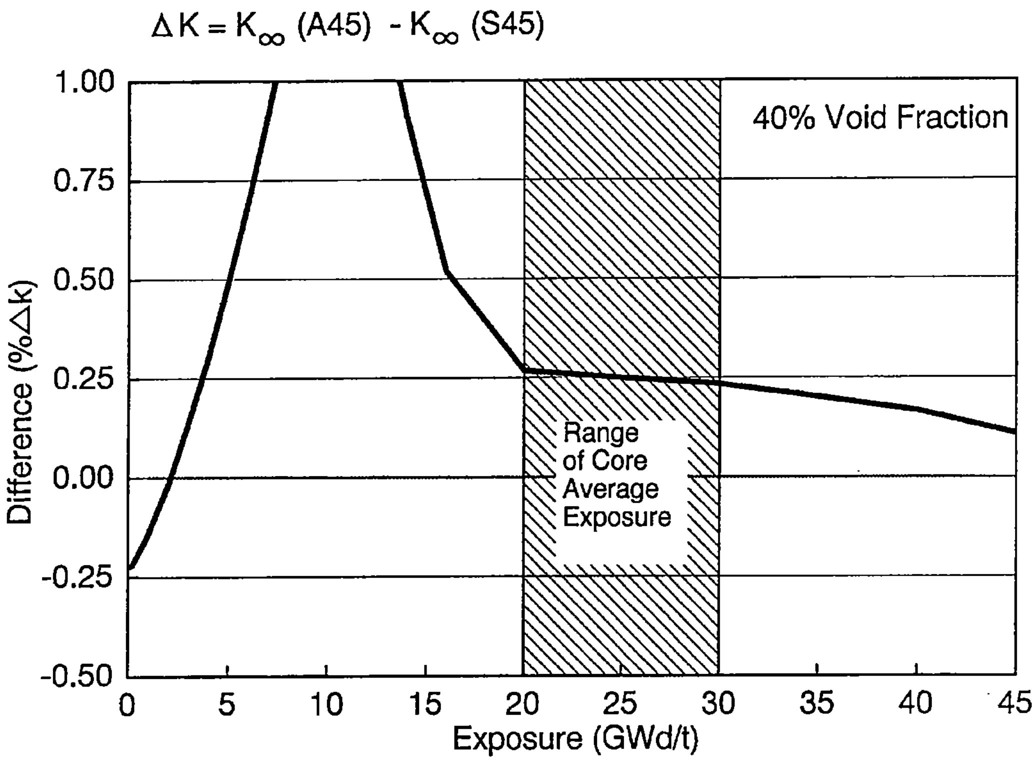


Figure 7. Reactivity Gain with Offset Water-Channel

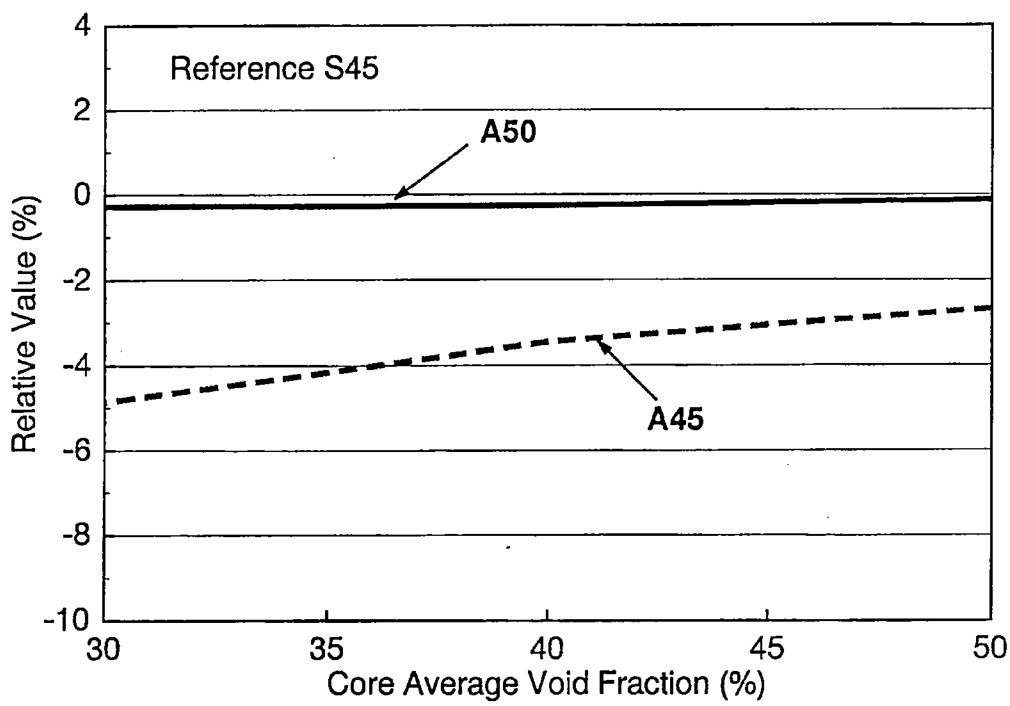


Figure 8. Comparison of Void Reactivity Coefficient

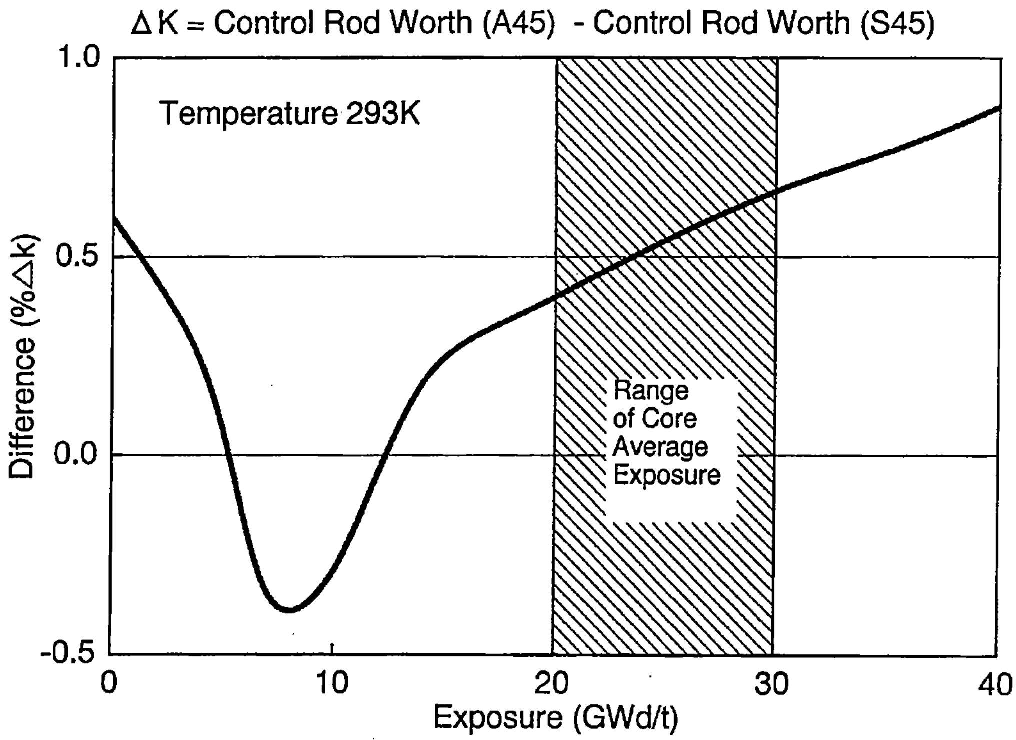


Figure 9. Control Rod Worth

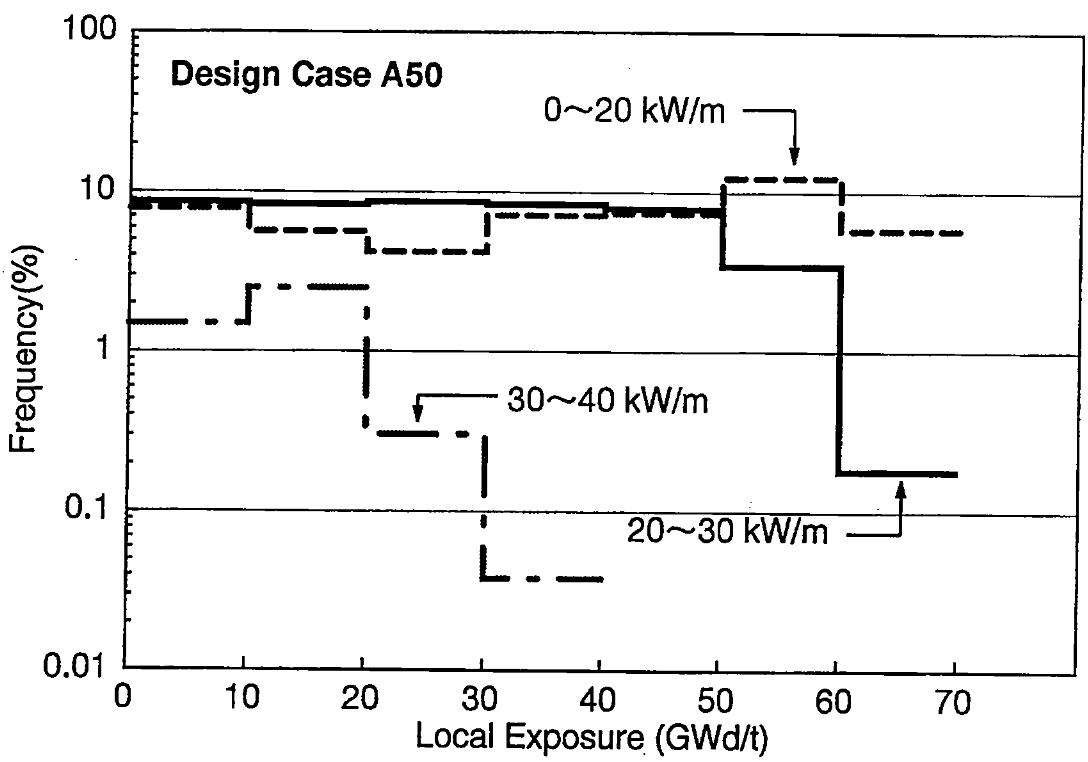


Figure 10. Histogram of Local Power

The Design Method for the ATR High Burnup MOX Fuel

I. Kurita
S. Uematsu
Power reactor & nuclear fuel development corporation

Abstract

The Power Reactor and Nuclear Fuel Development Corporation (PNC) has developed the advanced thermal reactor (ATR). PNC is demonstrating MOX fuel utilization in a prototype of ATR, Fugen(165MWe), in which 638 MOX fuel assemblies have been loaded without a failure since 1979. PNC is developing the high burn-up MOX fuel for the ATR to contribute to MOX fuels for thermal reactors. The statistical design evaluation method that included the MOX fuel rod performance evaluation code "FEMAXI-ATR" was developed for the ATR high burn-up MOX fuel rod.

1. Introduction

The Power Reactor and Nuclear Fuel Development Corporation (PNC) has developed the advanced thermal reactor (ATR), which is the heavy-water-moderated, boiling-lightwater-cooled, pressure-tube-type reactor. Light-water coolant becomes in the core steam-water mixture to be carried to the steam drum for separation of steam before it is directed to the turbine. The coolant temperature and pressure are 284 °C and 6.8MPa respectively, which are comparable to those of the BWR. PNC is operating a prototype of ATR, Fugen(165MWe), in which 638 plutonium-uranium mixed oxide fuel (MOX fuel) assemblies have been loaded without a failure since 1979. The ATR has the advantage of its flexibility in the kind of fuel it can use because fast neutrons are well slowed down in the heavy-water-moderator in the calandria tank (Fig. 1). In the case of plutonium utilization in the ATR, plutonium isotopic composition slightly affects on the nuclear characteristics of the ATR. PNC is developing the high burn-up MOX fuel for the ATR to contribute to MOX fuels for thermal reactors from the economic view point. The target burn-up is 55GWd/t, which is the same as that of the high burn-up UO_2 fuel for the LWR. PNC has a plan of the irradiation experiment with a few ATR high burn-up MOX fuel assemblies in Fugen (Fig. 2). This paper will describe the outline of the ATR high burn-up MOX fuel and the design method for the ATR high burn-up MOX fuel.

2. Outline of ATR High Burn-up MOX Fuel

The structure and main specifications of the ATR high burn-up MOX fuel assembly are shown in Fig. 3 and Table 1. To fit into the pressure tube design of the reactor, the fuel assembly is of a cylindrical configuration in which fuel pins are arranged in three concentric rings. The fuel assembly is composed of fuel pins, spacer supporting rods, the upper and lower tie-plates, and spacers. The upper and lower tie-plates, and spacers maintain the fuel pins in their desired positions. The fuel assembly is about 4.5m long and 110mm in diameter. Each pressure tube houses one assembly. The ATR high burn-up MOX fuel assembly is composed of 54 fuel pins(24 fuel rods in the outer-ring, 18 fuel rods in the intermediate-ring, 12 fuel rods in the inner-ring respectively) of 10.8mm in diameter and is designed for the maximum assembly burn-up of 55GWd/t. The fuel rod has the almost the same structure as that of the BWR fuel rod. The smaller outer diameter of the fuel rod and the increased number of fuel rods in comparison with the driver MOX fuel assembly and the MOX fuel assemblies for irradiation tests (28 fuel pins of 16.46mm in diameter and 36 fuel pins of 14.50mm in diameter respectively) loaded in Fugen contribute to decreasing the average linear heat generation rate. 6 UO_2 fuel rods with Gd_2O_3 in the 18 fuel rods in the intermediatering decrease the power mismatch between the fresh fuel and the remaining fuel. The spacer supporting rod is located in the center of the fuel assembly and is filled with non-void water, contributing to relaxation of the radial power distribution in the fuel assembly.

3. Development of Design Method for ATR High Burn-up MOX Fuel

3.1. Design Evaluation Method for High Burn-up MOX Fuel Rod

A conventional deterministic design evaluation method for the ATR MOX fuel rod has been used for the designings of many kinds of ATR MOX fuels so far, having sufficient reliability. However, it can be considered that the result of the design evaluation is severe in the case of the ATR high burn-up MOX fuel because as the burn-up increases internal pressure and cladding corrosion thickness increase, and the deterministic design evaluation method has an excessive safety margin. On the other hand, a statistical design evaluation method was introduced from the case of designing for the BWR step II fuel rod(maximum assembly burn-up:50GWd/t)from the view point of rationality in Japan. Therefore it was decided to develop and adopt the statistical design evaluation method with considerations of distributions of fuel dimensions in production and operation conditions for the high burn-up MOX fuel rod. It is necessary for having benefit of this method to predict thermal and mechanical performances of the high burn-up MOX fuel rod accurately during the irradiation period. A MOX fuel rod performance evaluation code "FEMAXI-ATR" was developed as a best fit code. Moreover, it was verified that the design parameters, such as fuel rod dimensions can be treated statistically in the FEMAXI-ATR code by researching fabrication data of the ATR fuels which PNC has made so far. (Fig.4).

3.2 The FEMAXI-ATR code [3]

The FEMAXI-ATR code is based on "FEMAXI-III" [4] that has enough experiences in evaluating UO2 fuel rod performances. It has fuel material property models and irradiation performance models specific to the MOX fuel as shown in Fig.5-8. Introduced models of MOX fuel material properties are melting point, thermal conductivity, thermal expansion factor, Young's module, creep rate, and so on. Introduced irradiation performance models are FP gas release rate, Xe to Kr ratio, production of He, and so on which are based on the results of the irradiation tests in Fugen, and so on. In the model of production of He, the production quantity due to a decay of Cm242 corresponding to the initial Pu density is considered, which is calculated by ORIGEN code. A correlation between He gas release and FP gas release[5] is also considered. A code verification was done by comparing calculated values of fuel center temperatures, fuel rod internal pressures, fuel cladding diameter changes, etc., with the results of the irradiation tests in Fugen, and so on. Calculated v,alues and measured ones had a good coincidence at a burn-up of up to about 60 GWd/t as shown in Fig. 9-13. Thus, it was verified that the FEMAXI-ATR code could predict accurately thermal and mechanical performances of the ATR MOX fuel rod during the irradiation period even at high burn-up. Uncertainty of the code predictions to measured values is considered in the statistical evaluation method as a safety margin.

3.3. Statistical Design Evaluation Method

Fig.14 shows flow diagram of the statistical design evaluationmethod. In this method, input values and evaluation values are dealt with statistically. Evaluation values are obtained by applying the sensitiveness analysis method and the standard error propagation theory to the results of analyses by the FEMAXI-ATR code. In the case of the evaluation of the cladding stress the statistical evaluation values of the cladding stresses are obtained by the Monte Carlo method. After that the 95% upper limit of the distribution of the cladding stress is compared with the allowable stress which is based on the shear strain energy hypothesis. This statistical design evaluation method was applied to the evaluation of the performance of the ATR high burn-up MOX fuel rod and it was verified that the integrity of the fuel rod could be maintained over the whole irradiation period. Fig.15 shows example of evaluation result of fuel melting. It is clear that 95% lower limit of the distribution of the power corresponding to fuel melting is sufficiently higher than the design power history in transient.

4. Current Status in Licensing Process

In Japan, there are two stages in the licensing for the nuclear facilities. In the first stage to get a permission, the fundamental design policy and fundamental design of the nuclear facility is assessed by the competent authorities and the Nuclear Safety Commission(NSC) which is an advisory organization of the prime minister. In the second stage, the constructor have to get an approval by the competent authorities for the detailed design and the construction way. PNC has almost completed the fundamental design of the ATR high burn-up MOX fuel and is now preparing for the first stage of the licensing process. The new design method will be assessed in the first stage. The consideration of the characteristics of the MOX fuel, the understanding of unreliability of code prediction and the adoption of the design estimation condition with a safety margin will be the keys from the regulatory view point.

5. Conclusion

The statistic design method for the ATR high burn-up MOX fuel that included the MOX fuel rod performance evaluation code "FEMAXI-ATR" was developed. the FEMAXI-ATR code was verified by the results of the irradiation tests using reactors such as Fugen, and so on. Thus, the statistic design evaluation method for the ATR high burn-up MOX fuel was established. This statistic design evaluation method was applied to the evaluation of the performance of the ATR high burn-up MOX fuel and it was verified that the integrity of the fuel could be maintained over the whole irradiation period.

6. References

[1] Sawai, et al., "Characteristics of plutonium utilization in the Heavy-Water-Moderated, Boiling-Light-Water-Cooled Reactor ATR", Nuclear Engineering and Design 125, 251257, 1991

[2] H. Ikeda, et al., "Operating Experience of Fugen-HWR in Japan", IAEA TC-meeing on Advances in Heavy Water Reactors, Bombay, India, 29 January 29 February 1st 1996

[3] N. Onuki, et al., "Fuel Design Method for the High Burnup MOX Fuel", PNC Technical Review, No 96, December 1995

[4] T. Nakajima, et al., "FEMAXI-III:A Computer Code for the Analysis of Thermal and Mechanical Behavior of Fuel Rods", JAERI 1298, 1985

[5] M. Billaux, et al., "Production of Helium in UO_2-PuO_2 Mixed Oxide Fuel", TWGFPT-32 P.182-186, 1989

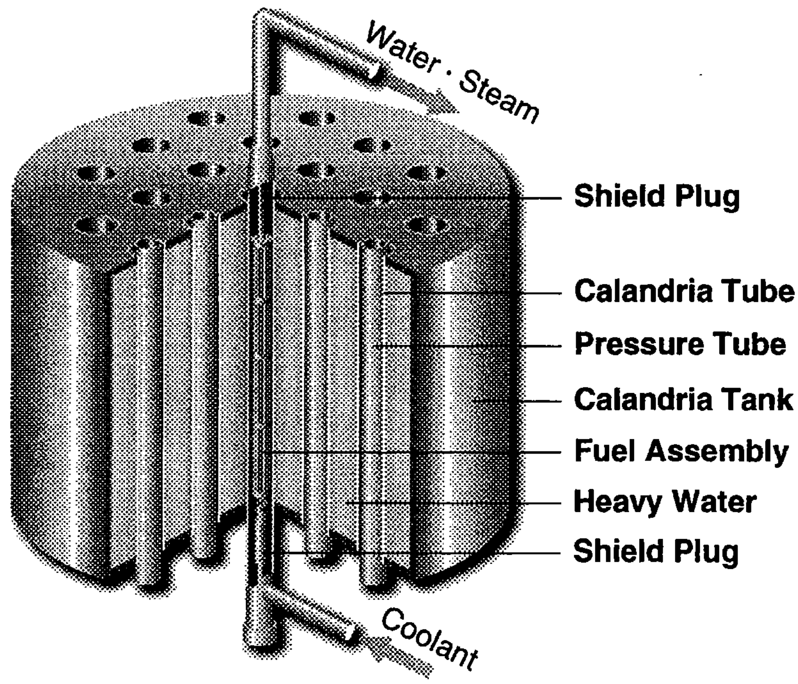
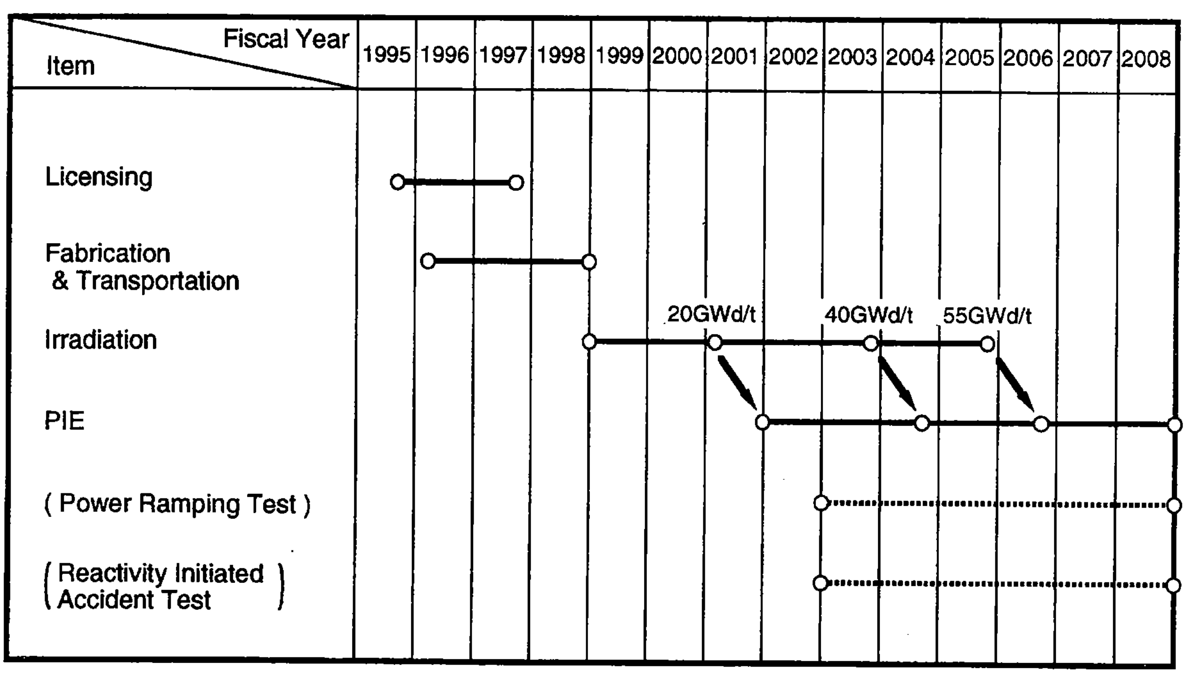


Figure 1. Conception of ATR Core Structure



( ) : Under Consideration

Figure 2. Schedule of ATR High Burn-up MOX(AHM) Fuel Irradiation Test

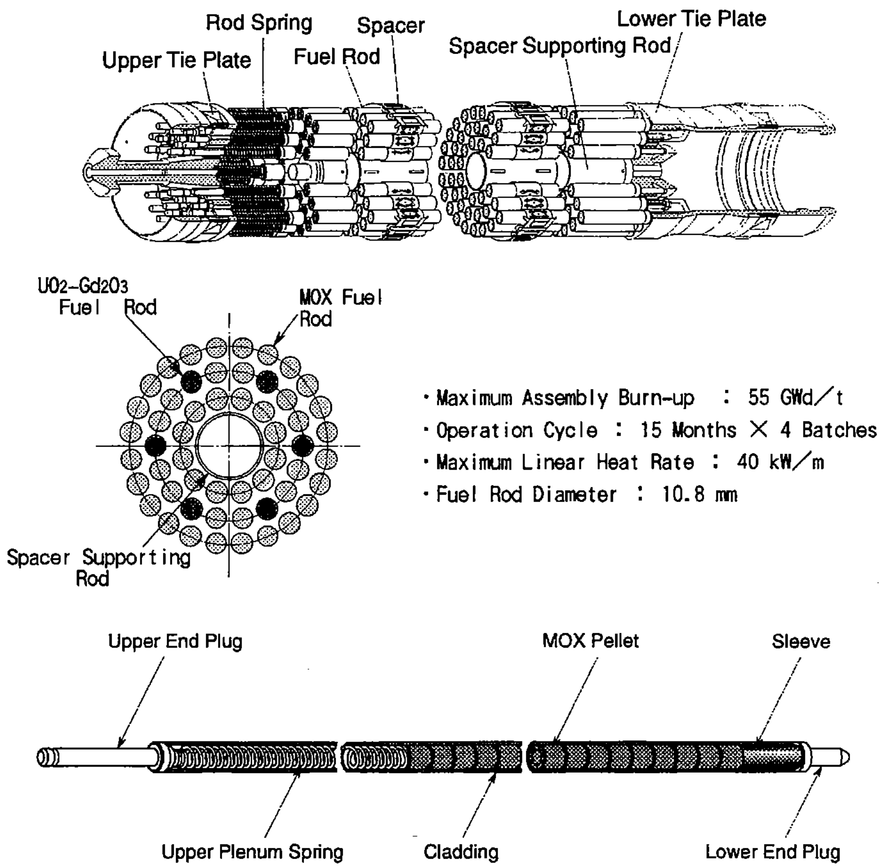


Figure 3. ATR High Burn-up MOX(AHM) Fuel Assembly

Conventional Design Evaluation Method for ATR MOX Fuel Rod

Evaluation method	**Performance code: ATFUEL**
-Deterministic Method All inputs are determined in order to obtain conservative results	-Fuel Performance Prediction with Safety Margin

⇩

Accumulation of Measurement Data (Production, PIE data)

New design Evaluation method for High Burn-up MOX fuel rods

Evaluation method	**Performance code : FEMAXI-ATR**
- Statistical Method Input Data have Statistical Values	-Fuel Performance Prediction with Agreement with Data

Figure 4. Comparison of Design Evaluation. Methods for ATR Fuel Rod

Item	54-Fuel-rod Assembly
1. Fuel Pellet	
Material	PuO_2-UO_2
	UO_2-Gd_2O_3
Outer Diameter(mm)	9.10
Height(mm)	9.5
Density(%T.D.)	95.0
Shape	Solid with Dish & Chamfer
Pu Fissile Enrichment(wt%)	3.6~4.8
2. Fuel Rod	
Cladding Material	Zircaloy-2 with Zr-Liner
Outer Diameter(mm)	10.80
Inner Diameter(mm)	9.30
Diametral Gapsize(μm)	200
Fuel Stack Length(mm)	3,600
Filling Gas	He
Filling Gas Pressure(MPa)	0.5
3. Fuel Assembly	
Length(mm)	4,381
Bundle Outer Diameter(mm)	111.6
Rod Number	54
Inner-Ring Rod	12
Intermediate-Ring Free Rod	12
Intermediate-Ring Tie Rod	6
Outer-Ring Rod	24
Spacer Number	12

Table 1. Fuel Specification

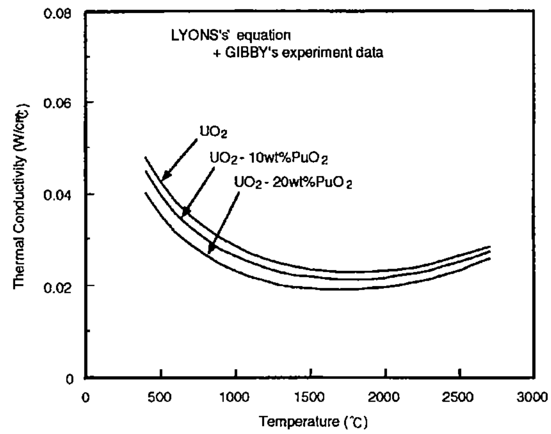


Figure 5. Model of FEMAXI-ATR (Material Property) (Fuel Pellet Thermal Conductivity

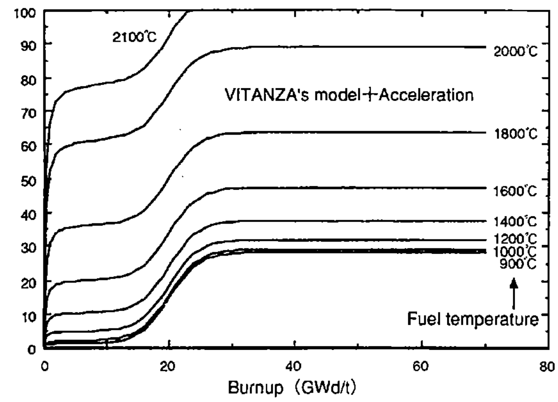


Figure 6. Model of FEMAXI-ATR (Material Property) (FP Gas Release vs. Burnup and Temperature)

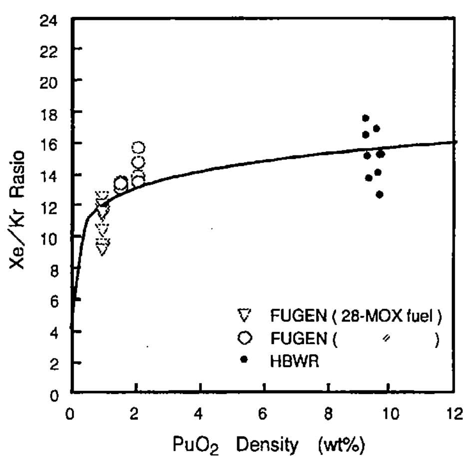


Figure 7. Model of FEMAXI-ATR (Irradiation Behavior) (Xe/Kr Ratio vs. PuO_2 Density)

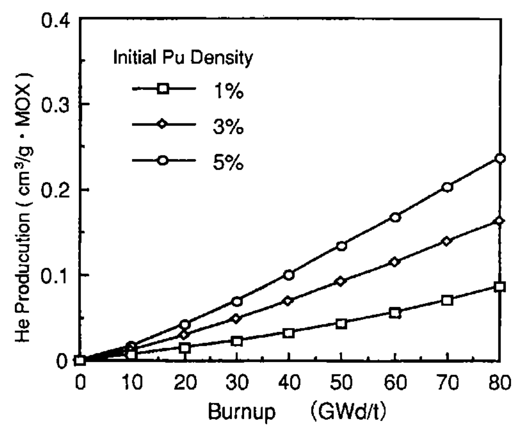


Figure 8. Model of FEMAXI-ATR (Irradiation Behavior) (He Producution vs. Burnup)

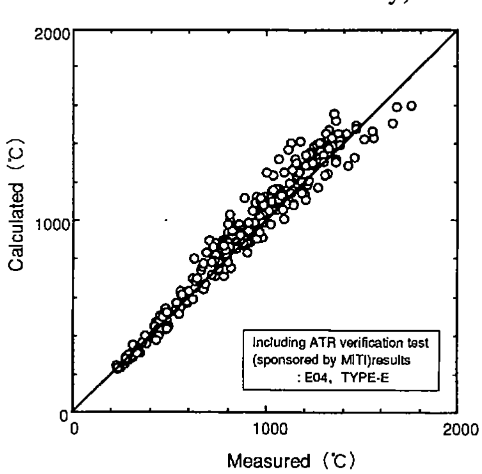


Figure 9. Verification Results of FEMAXI-ATR (Fuel Centerline Temperature)

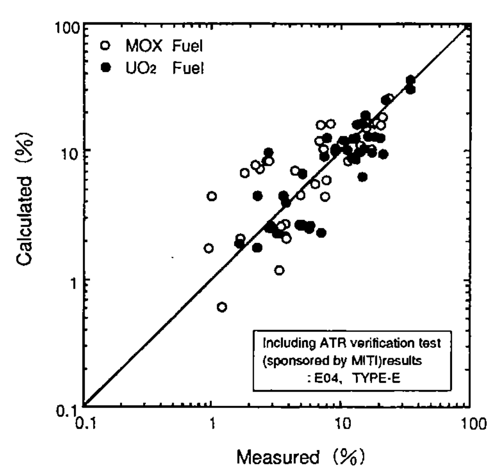


Figure 10. Verification Results of FEMAXI-ATR (Fission Gas Release)

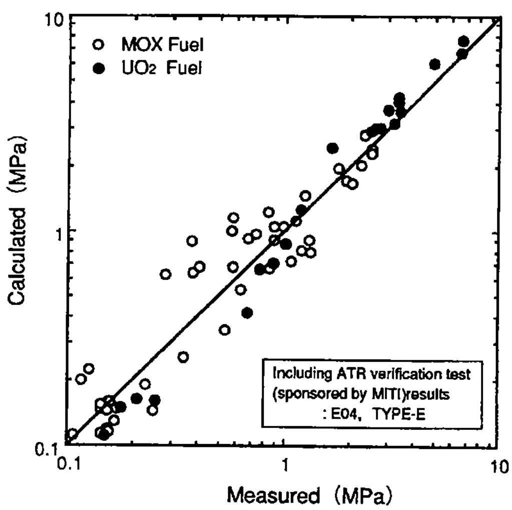


Figure 11. Verification Results of FEMAXI-ATR (Fuel Rod Internal Pressure)

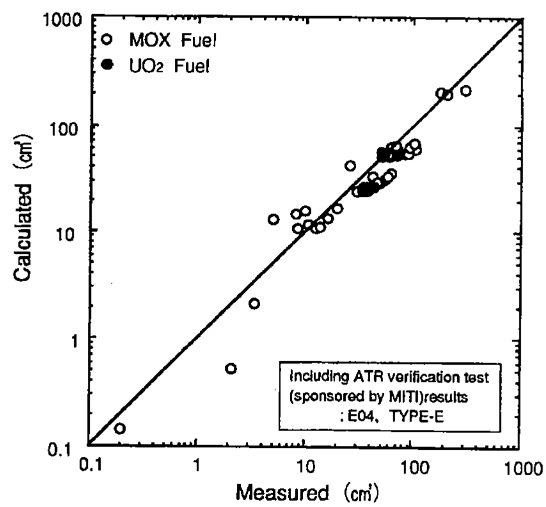


Figure 12. Verification Results of FEMAXI-ATR (Fuel Rod Internal He Gas)

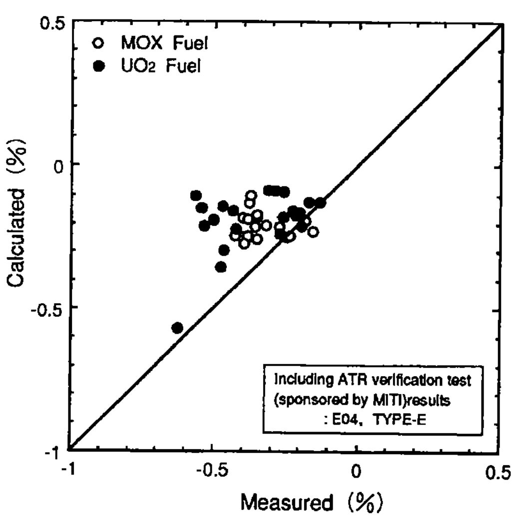


Figure 13. Verification Results of FEMAXI-ATR (Fuel Rod Outer Diameter Change)

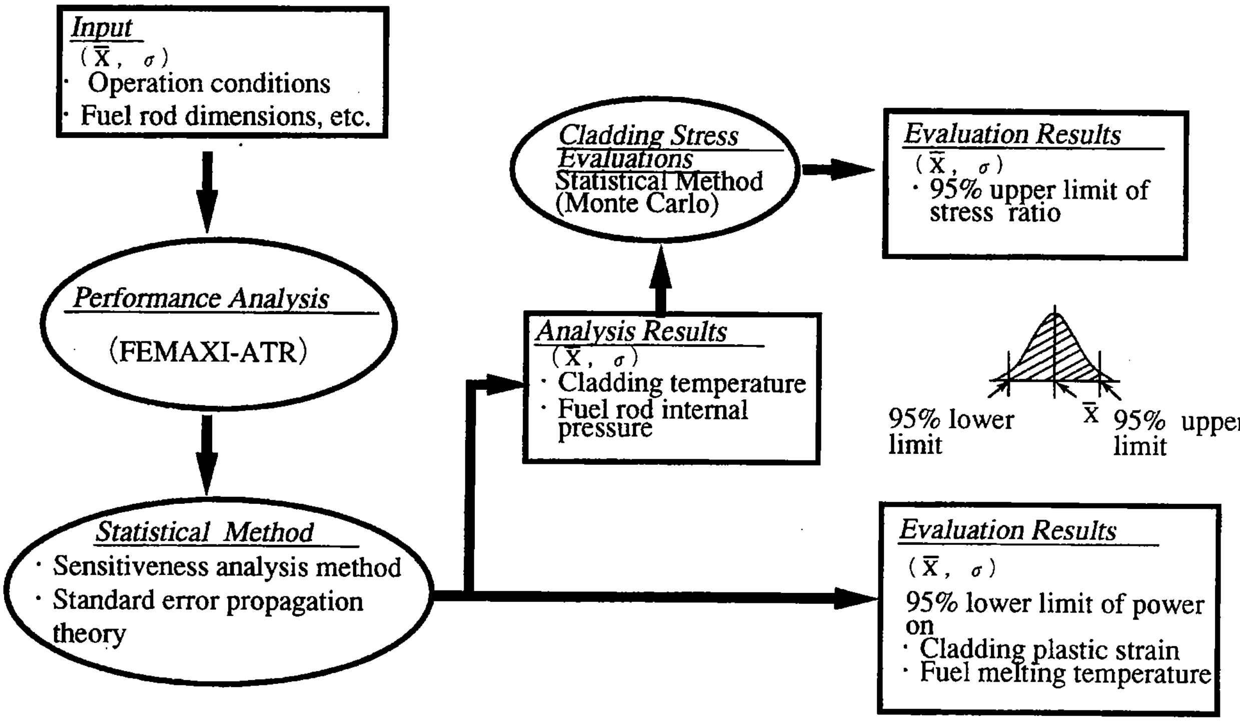


Figure 14. Flow Diagram of Fuel Design Evaluation (Statistical Evaluation Method)

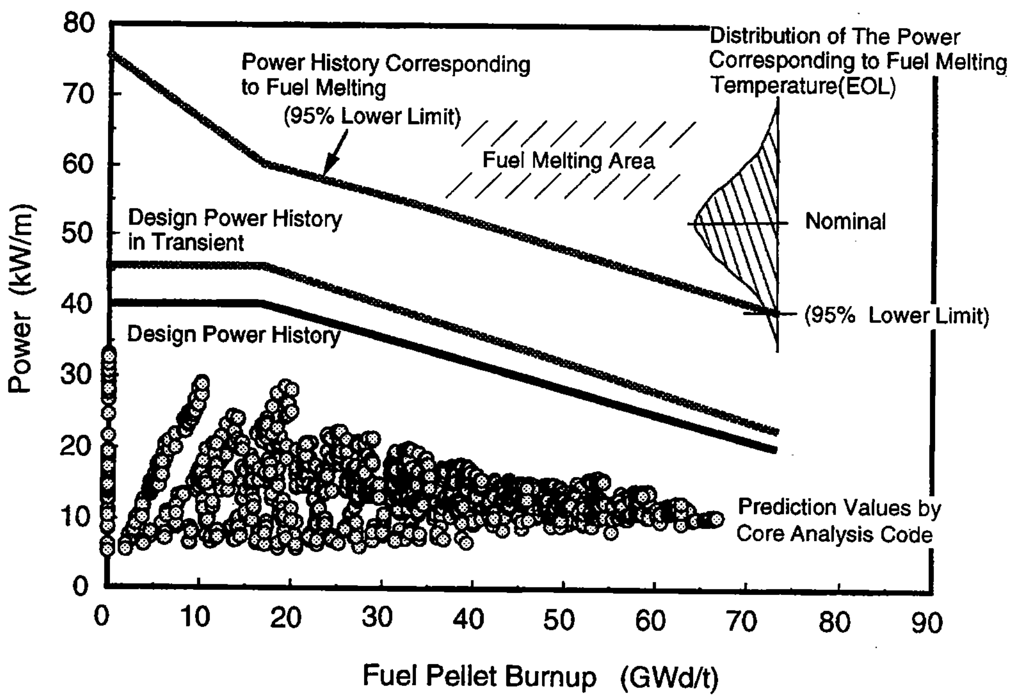


Figure 15. Evaluation Result on Fuel Melting

Segmented Fuel Irradiation Program. Investigation on Advanced Materials

H. Uchida – I. Komine
NUPEC
K. Yamate
KANSAI
S. Abeta
MHI
J. M. Alonso
ENUSA

1. Introduction

The Segmented Fuel Irradiation Program is a collaboration program between Japanese organizations (NUPEC, Kansai Electric Co, representing other Japanese utilities, and Mishubishi Heavy Industries) and Spanish organizations (ENDESA, A.N. de Vandellós-II, and ENUSA), with the participation of Westinghouse.

The objective of the Program is to make substantial contribution to the development of advanced cladding and fuel materials for better performance at high burnup and under operational power transients.

For this Program segmented fuel rods were selected as the most appropriate vehicle to accomplish with the objective above. In this way a large number of fuel and cladding combinations are provided while minimizing the total amount of new material, and, at the same time, facilitating an eventual irradiation extension in a test reactor.

The Program consists of three major phases:

- Phase I : design, licensing, fabrication and characterization of the assemblies carrying the segmented rods.
- Phase II : base irradiation of the assemblies at Vandellós-II NPP, and on site examination at the end of the four cycles.
- Phase III : ramp testing at the Studsvik facilities and hot cell PIE.

The main fuel design features whose effects on fuel behaviour are to be analyzed in this Program are:

- alloy composition (MDA and ZIRLO vs. Zircaloy-4)
- tubing texture
- pellets grain size

This paper presents the Program current status, and the results of the inspections already performed.

2. Program status

Figure 1 summarizes current overall Project schedule.

Phase I finished in 1993 after the segmented fuel assemblies delivery to Vandellós-II.

Phase II is being realized at this moment. The main activities already finished by October 1996 are indicated below:

- Irradiation of cycle 7 and 8 at Vandellós-II
- On-site PIE at EOC-7 (visual, dimensional and corrosion), and at EOC-8 (visual, dimensional and corrosion)
- Removal of 14 segmented fuel rods to be sent to Studsvik.

Phase III of the Program is just started and it will progress in parallel with Phase II. That phase includes: a hot cell non destructive examination of segments to be ramp tested, additional irradiation at the Studsvik R-2 Reactor under power transients, and a very throughfull hot cell PIE.

3. Inspections results

The first on-site inspection campaign took place during the EOC-7 refuelling. The fuel achieved and average burnup of 15 MWd/kgU and a residence time of 350 days.

It follows a summary of the main results obtained in that campaign.

3.1. UT Inspection

Objective:	fuel rod integrity verification
Scope:	all fuel rods of the 4 fuel assemblies at seven elevations
Results:	fuel rod tightness confirmed

3.2. Fuel Rod Length

Objective:	fuel rod growth characterization for every cladding material
Scope:	all peripheral fuel rods in the four fuel assemblies
Results:	– slightly less growth observed in the advance materials – conventional materials grow less than expected – no growth differences between segmented and non-segmented fuel.

3.3. Fuel Rod Spacing

Objective:	fuel rod bow characterization for the different materials
Scope:	rod-to-rod gap measurement along the periphery of the four fuel assemblies at six mid-span elevations
Results:	– very low bow values observed, well below thermal-hydraulic design assumptions.

3.4. Eddy Currents on Cladding

Objective:	cladding corrosion characterization for the different materials
Scope:	oxide thickness measurements of 30 peripheral fuel rods from the 4 fuel assemblies
Results:	– no significant different behaviour among the various materials is observed – the expected corrosion is exhibited by the conventional materials.

3.5. Fuel Rod Profilometry

Objective:	cladding irradiation creep characterization of the different materials
Scope:	fuel rod profilometry of 30 rods removed from 2 fuel assemblies
Results:	– irradiation creep is significantly less for the advanced material – fuel densification is lower for the large grain pellets.

4. Conclusion

The segmented fuel assemblies are operating satisfactorily. At the present time, all the provisions of the Irradiation Program have been successfully met.

The inspections already performed indicate the following major conclusions:

- advance cladding materials are dimensionally more stable (less growth and creep)
- no significant differences in the corrosion behaviour have been observed
- large grain pellets densify less.

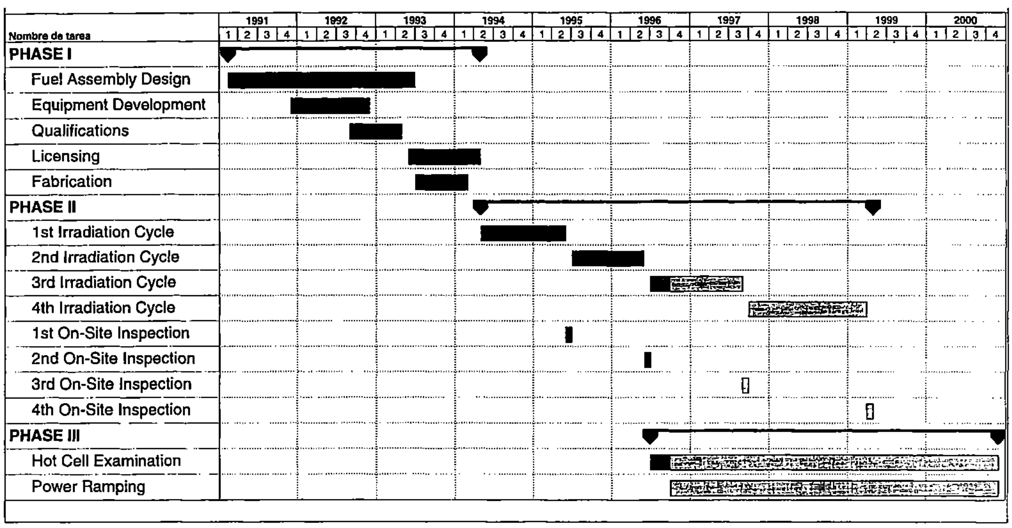


Figure 1. Segmented Fuel Irradiation Program

TECHNICAL SUB-SESSION IV.B

Total Quality Project Initiatives and Fabrication Improvements at ENUSA Factory

J. I. Martín Galán
ENUSA

Introduction

ENUSA is committed to maximize fuel reliability through different improvement programs to cover fabrication processes, inspection techniques bechmarking with different partners and total quality projects.

All these programs focuss on reliability of product trough three main concepts Quality as a priority, continuos improvement and personnel involvement in their work.

Cero defects program

ENUSA's goal is to obtain a factor of 10 improvement and approach the value of 1/1.000.000 or 1 ppm. This goal is realistic and achievable since product from a partner in fuel fabrication improvement programs (JNF) has already achieved this level of reliability.

Fuel bundles of GE BWR design is actually manufactured at three different fuel manufacturing facilities globally.

- GE fuel factory at Willmington, N.C, USA
- ENUSA fuel factory at Juzbado, Spain
- JNF * fuel factory at Kurihama, Japan, (JNF factory is jointly owned by GE, Hitachi and Toshiba)

Although the fuel design and engineering specifications are essentially the same for the three factories, the actual fabrication process is different. This offers a unique opportunity to review the applied fabrication practice and identify the "Best Practices" with respect to quality and reliability.

A task Force was formed for this purpose and to review all aspects relating to fuel rod hermeticity (component suppliers methods, welding techniques, end plugs design, tubes certifications, ... etc.).

The work included the visit to all fabrication plants as well as the supplier's in order to review fabrication processes, quality methods and quality in general at each plant.

Among the many recommendations agreed we can distinguish the following :

- Soft Star in welding
- Compensator on welding chanfer
- Optimized end plug design
- Argon elimination
- Tubes UT inspection improvement ... etc.

Another Best Practices Task Force was formed to review all aspects of pellets manufacturing following the same technique than the previous one.

The purpose was to eliminate any material missing from the pellets and define the best powder purchasing specification to guarantee homogeneity and absence of problems on the fabrication of pellets.

Some of the recommendations we can mention are :

- Soft pellets handling
- Automatic rods loading
- Blending process improvements
- Better control on additives
- Image analyzer inspection techniques ... etc.

Among the Total Quality Projects ENUSA has worked on :

- Documentation Optimization
- Fabrication Time Reduction
- Automatic Design Manufacturing interface
- New Gd line autonomy

The document system was redesigned to reduce the number of documents. The former system revealed an excess on the number of papers, it required an slow and complicated process, and it dismanded an excessive human dedication.

The goals were fixed on a 25% of reduction in Volume, an standardization of the documents, a flow simplification and improve in the overall quality.

The fabrication time reduction project was intending to reduce the fabrication time in a 30-40 %.

Enusa considered there as a lot of margin for cycle time reduction, there were an excessive number of bottle necks in production, the planning was complicated and there was small flexibility.

All these aspects were attacked and the result is that working in a more efficient manner the goal was succeeded and accomplished in a 100 % and fully implemented in Juzbado.

Another important need was the improvement and simplification of all Engineering-Manufacturing interface aspects.

For this reason ENUSA developed an automated Engineering and Manufacturing interface system called ALMA. This system replaces a paper based system with an electronic interface that eliminates all manual transposition of information from the nuclear and mechanical engineering definition of the bundle through the assembly of the bundle on the factory floor.

The new Gd line is working with also a new Total Quality Fabrication Concept.

The whole line is run by only two operators in a two shifts schedule and another operator between both shifts.

This has proven to be efficient and feasible.

Conclusion

World is moving so fast, actions have to be implemented without delay-

All these design and manufacturing modifications will give beneficial results to the cero defect program implemented in ENUSA.

Application of Ultrasonic inspection Technique on Fuel Rod Seam Weld

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M. Inatani
N. Kamata
JNF

Abstract

As of the end of March, 1996, 26 BWR power plants of which station capacity has reached up to 23000 GW in total are in commercial operation in Japan, Japan Nuclear Fuel Co., Ltd. (JNF), a BWR fuel fabricator in Japan, has supplied fuels to those power plants for 25 years. This paper presents refinement of inspection technology applied to enhance completeness of fuel rod welding at JNF, which has cumulatively produced approximately 50,000 fuel bundles to the date in Japan.

In this operation, TIG method has been employed for plug to tube welding of fuel rod, and X-ray radiography was formerly applied as nondestructive testing (NDT) means in order to verify weld integrity of every fuel rod. As there was limited capabilities of X-ray radiography such as shooting time and direction, and also inspection of fuel rod weld integrity is one of key characteristics of regulatory inspection according to the law, JNF has developed and applied more reliable and effective probe rotation type ultrasonic method. This paper presents refinement of inspection technology applied to enhance completeness of fuel rod welding at JNF.

1. Introduction

Japan Nuclear Fuel Co., Ltd.(JNF), a BWR fuel fabricator operated since 1970, has cumulatively produced approximately 50,000 fuel bundles to the date in Japan. The majority of them are of 8x8 design feature which has demonstrated fairly good performance of 4.5E-06 failure rate out of 2.47 million fuel rods and current product is also of 8x8 STEP-II design, a high burn up fuel(38 GWD/T).

It has been used TIG weld method for plug to tube weld under 25 years operation experience. Traditionally JNF had employed corner TIG weld which makes round shape bead, in combination with X-ray radiography for nondestructive means of weld integrity.

It took cumulatively a half hour per testing of 18 fuel rods for X-ray method to put into operations through shooting a picture, film development and fixture, and visual observation. In order to verify the metal thickness of welded portion, it had been taken in two or three shooting angles out of 360 degree round shape. Because traditional x-ray radiography has the resolution limit of detecting 0.3 mm diameter porosity, it required inspector having his skill and experience.

In order to solve the problems including precision, time and labor consuming practice, and human skill dependence, JNF have developed more reliable nondestructive testing method having following features;

1. enhancement of inspection precision removing dependence on human skill,
2. replacement of time and labor consuming practice by an advanced fast mechanicalelectric means,
3. weld design change, if necessary, to be applicable by new inspection method but without change TIG welding practice,
4. be compatible with the mechanized fabrication stream.

X-ray radiography and ultrasonic testing(UT) are popular as nondestructive inspection methods of weld. Regarding UT comparatively more precise and faster, JNF has developed to establish UT microscope for plug to tube weld inspection.

2. Weld UT Microscope

Two incident angle methods are popularly used for UT of metal weldment, angle beam and straight beam. In the case of corner TIG weld having round shape bead, it was difficult to detect reflected echo because straight beam was irregularly reflected at round surface, and angle beam incident from tube side was also irregularly reflected at round bead inside. Instead of corner TIG, flush TIG weld forming smooth cylindrical shape and surface was employed, the former tungsten electrode was positioned in the angle of approximately 45 degree toward the corner of plug flange fitting on the end of cladding tube, the latter is positioned in right angle toward the seam of plug barrel and cladding tube. These two methods are basically same TIG welding technology and it causes no material change or no characteristic change of nuclear fuel regarding reactor operation. One of two UT candidates was found problem that the angle beam for flush TIG weld generates edge echo at the inside weld notch, which cannot distinguish with defect signal. On the other hand, flush TIG weld enables straight beam method to detect the echo from porosity or other reflective defects as well as normal surface and bottom echoes for metal thickness measurement. Then, it was concluded that flush TIG weld was employed in combination with straight beam UT.

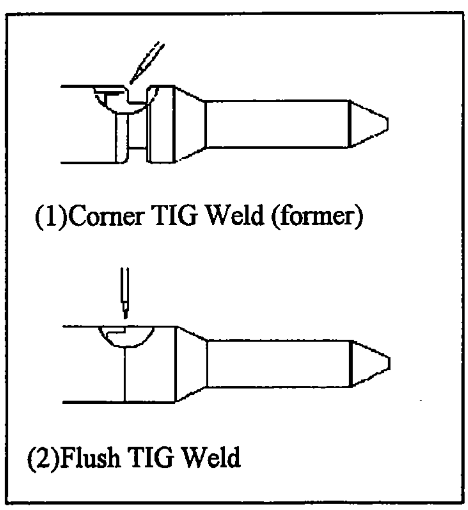

Figure 1. Corner TIG Weld and Flush TIG Weld

Immersion type point focused transducer was selected, which has capability to detect porosity in diameter of 0.2 mm. In order to cover whole welded surface with the focused ultrasonic beam, two spiral scanning methods were considered, one is rotating fuel rod and longitudinally moving UT transducer, another rotating transducer and longitudinally moving fuel rod. The latter was employed, because the former has potential problem that rotation of fuel rod makes pellet chips or tube inside surface scratch. Bubbler method realized transducer rotation, which made water path for ultrasound propagation and made up small parts incorporated with focus adjustment device mounted on rotation mechanism. It was capable of scanning weld surface approximately 50 micro meter mesh in synchronized combination with transducer rotation, longitudinal movement of fuel rod, and firing of UT pulser/receiver.

It measures metal thickness by using time of flight method applying the surface and bottom echoes of weld and tube portions. It is also capable of detecting porosity or reflective defects that signal gate is positioned within the time range of surface and bottom echoes. Synchronous combination of electronics and mechanism enables to obtain metal thickness and defect signal for every UT firing or each mesh, and every data constitutes two dimensional spread of longitudinal and circumferential 0 through 360 degree directions, which indicates characteristic patterns of metal thickness and defects for each weldment, assigning certain colors for each data according to its magnitude. In the case of porosity, it makes a lump of the detected signals in the defect pattern according to its dimension. Fig.2 shows ultrasonic measurement and two-dimensional pattern of weld UT micro scope.

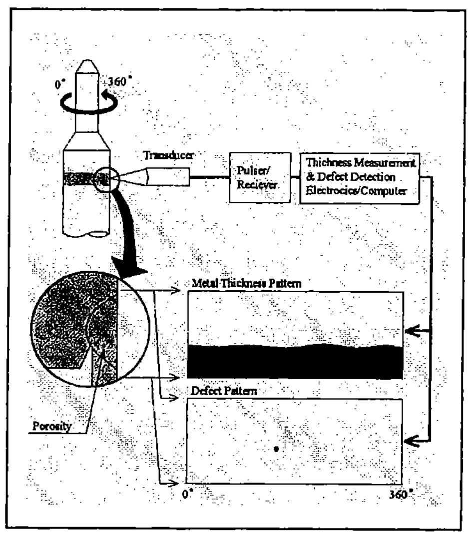

Figure 2. Ultrasonic Measurement and Two-dimensional Pattern of Weld UT Micro scope

This system is capable of measuring metal thickness in 0.001 mm resolution and detecting porosity or reflective defects larga than 0.17 mm in diameter, and pursuing data acquisition, analysis, result display, and recording within 17 seconds. Fig. 2 shows two dimension pattern which clearly displays metal thickness characteristicsand porosity indications quantitative and objective.

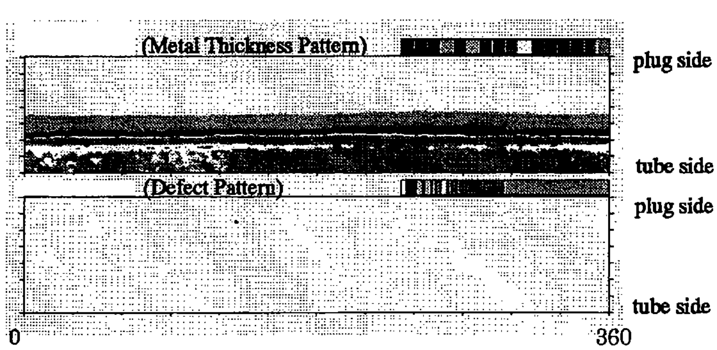

Figure 3. Two-dimensional Pattern of Weld UT microscope

Application of ultrasonic microscope on tube to end plug weld inspection provides many advantages over the traditional practice in terms of inspection technique and weld matrix characteristics:

a) precise detectability of finding porosity of 0.17 mm dia. instead of 0.3 mm dia,

b) inspect 100% cross section of weld bead both for defects in bead matrix and metal thickness instead of X -Y section,

c) ultrasonic imaging telling detail figure of inside bead at any cross section,

d) elimination of ambiguity due to X - ray radiography,

e) revolutional reduction of inspection and cycle time to 15 and 38 seconds respectively versus those of traditional but redundant X - ray radiography,

f) good detectability leading to produce more reliable and high quality product as it is well aware,

g) finally removal of several operators and/or inspectors occurred.

In addition, we established the capability of detecting micro defect on the inside surface of weld bead area as the off-line system. Micro defect is occurred in the specific angle toward tube longitudinal direction and also circumferential welding rotation ,and it is observed that the specific angle is variable to the welding parameters, or in other word, once process is qualified, the angle is constant. Consequently it was considered compound angle method, which three-dimensional rectangular incident toward micro defect makes a part of ultrasonic spiral track on the weld or tube so to be obtained the best signal to noise ratio or detectability theoretically and experimentally. The compound angle method is effective for detecting the micro defect on weld bead inside surface.

3. Conclusion

Thus pursuant to the application of ultrasonic microscope and flush TIG welding ensure that JNF is able to produce fuels with required reliability and positive progress obtained to the date encourages us to employ same technology onto 9x9 STEP-III design, a higher burn-up fuel(45 GWD/t) which has 23 % increase in numbers of inspection per assembly. It would be already acknowledged our technological development at this area is not only meeting the predicted benchmarks, but also greatly contributing general quality up-grade and efficiency improvement as a function of labor and cycle time.

4. Acknowledgment

We would like to express our gratitude to Toshiba Corporation, Hitachi Ltd. and General Electric Company for providing technical assistance as well as fuel performance information to sophisticate our equipment and/or fabrication technology.

Impact of Present Fuel Management Strategies on Maintaining Safety Margins. Mixed Core Aspects. ENUSA's Experience in the PWR Area

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ENUSA

Introduction

The Nuclear Industry has been characterised during the last years by a general trend for improvements driven by an increase of the utility needs related to a series of aspects: Plant availability, plant operability, plant performance and economics, increased regulatory requirements, focus on optimising the operation of existing facilities, etc.

In the Nuclear Fuel Industry this general trend has led to a development effort for designing and introducing new and advanced Nuclear fuel and Software Methodologies capable of responding to these general needs. At the same time, due to other factors like technology, economic competition, security of supply, etc., the qualification of an alternative nuclear fuel supplier is becoming a more common situation. These circumstances have contributed to the spreading and generalisation of the Mixed Cores configurations, i.e. those cores containing fuel assemblies of different designs, either from the same or different Fuel Manufacturer.

This generalisation of the mixed-cores situation has led to the implementation at ENUSA of a systematic design process aimed to the evaluation and licensing of the mixed-cores configurations and the individual aspects of each different fuel design which is involved. Furthermore, with the in-house design capabilities and the introduction of improved design methodologies, the fuel assembly designs are addressed by ENUSA from their inception with the basic design objective of minimising the potential unfavourable mixed core effects affecting the safety margins

Naturally the scope and level of the required analyses depends on the level of fuel design changes and methodologies that are introduced. The basic objective that is pursued is the integral assessment of all the design areas in order to demonstrate the compliance with the licensing basis of each particular application, not impacting or even improving the current plant Safety margins or improving them when possible.

Compatibility Assessment

As mentioned previously, due to the importance of minimising the impact of the introduction of a new fuel design, the potential mixed-core configurations are taken in consideration from the beginning and specific Functional Requirements are established to guarantee the compatibility along the fuel design process for each particular application or for the target plants for which the new fuel design is intended.

This effort oriented to make the new design as compatible as possible may go up to the extent of introducing design changes in the principal components.

Development and Design Verification Process

According to ENUSA´s design experience the development and design process for a new fuel assembly design application is the following :

1) Define the current or future Design and Licensing Requirements

2) Establish the Design Functional Requirements

3) Assemble all the required Plant and Resident Fuels Compatibility Information,

4) Perform a Preliminary Assessment of the Design Compatibility,

5) Develop the Fuel Design, incorporating the required features to guarantee compatibility and minimising the potential unfavourable transition core effects which reduce the Safety Margins or affect the plant power capability

6) Perform the Design Verification including the appropriate test programme (for individual components and full scale prototypes) covering all design areas.

   - Fuel Assembly Mechanical Design
   - Fuel Rod Thermal-mechanical design
   - Hydraulic and Thermal Hydraulic Design including Transition core evaluation
   - Nuclear Design
   - Safety Analysis

7) Preparation of Licensing Reports

8) Review and Licensing of the proposed design

This means that the mixed-cores are configured after a prior and complete design assessment of the compatibility of the new fuel design against the different elements involved. As indicated this can imply that the Fuel Assembly is designed or adapted to the Design Specifications required for each particular application in order to minimise the potential compatibility problems.

The compatibility assessment and design verification is performed to assure in general the following design aspects:

Geometrical and Physical Compatibility

Reactor internals geometric interface: Lower and upper core plate pins and holes interfaces, Core Instrumentation interfaces, Core components interfaces, fuel assembly shipping handling and storage, etc.

Mechanical Compatibility

Mechanical and structural compatibility of the fuel assembly and its components with respect to:
RCCA Compatibility: impact velocities, axial interfaces, drop times,
Mechanical behaviour: fuel assembly growth; grid positioning mismatch, Mechanical loads against core internals. Fuel rod to grid fretting.
Structural Compatibility with respect to impact loads during design transients, etc.

Fuel Rod Thermal-mechanical compatibility

Fuel Rod design compatibility with respect the fulfilment of the required energy and fuel rod performance requirements related with the plant operational duties and fuel duties, power density, coolant chemistry, etc.

Thermal-Hydraulic Compatibility

Hydraulic and Thermal-hydraulic compatibility aspects: Fuel assembly pressure drops: Core design flow, Lift-off forces, localised pressure drops mismatches which create local or axial flow redistributions affecting thermal margins or design transients implications. Core stability considerations

Neutronic Compatibility

Assessment to determine the impact of the potential changes in the geometrical characteristics and material inventories with respect to the neutronic compatibility and the kinetic parameters behaviour of the mixed-core configurations to be considered for the Safety Analysis.
Compatibility with Plant and Core monitoring software

This compatibility evaluation is performed against Plant Specific Compatibility Information. Also the assessment implies the use of particular techniques and methods developed to analyse each design aspect.

The results of this compatibility assessment permit to identify all potential concerns related to the safety evaluation and can identify the need to incorporate methods or safety analysis evaluations to generate additional design margins to overcome the potential transition core impacts where it is required. Also the introduction of new fuel designs can be performed in conjunction with the introduction of new design methodologies like Thermal-Hydraulic methodologies and CHF Correlations, Fuel Rod Performance Models or Transient Analysis Methods improvements, etc. which determine the level of extension of the Safety Analysis effort that is required.

Assuming that no major changes are introduced in the plant licensing basis (as Power Upratings or other important operational parameters like Core flow, and temperature, Peking factor limits, etc.). Apart from the basic compatibility aspects: geometrical and physical interfaces, neutronic compatibility etc., there are two design aspects that require particular study and attention because they are potentially most affected by the transition core configurations impacting Safety Margins. These are, the structural performace of the new fuel design and the hydraulic and thermal hydraulic performance

Structural Performance

Structural integrity of the fuel assemblies, new and resident fuels, must be assured. To this effect, limits are defined on stresses and deformations as a consequence of the design loads and by determining that the assemblies do no interfere with the functioning of control rods .

Different types of loads must be considered:

- Non operational loads due to shipping and handling.
- Normal and upset loads which are defined for condition I and II: Scram loads, Top Nozzle Spring loads, lift forces
- Abnormal loads which defined for conditions III and IV as the safe shut down earthquakes (SSE) and Loss of Coolant Accident events,

The later are those most directly affected by the potential mixed core configurations, as a consequence of the importance of interactions between the fuel assemblies themselves, and also between the fuel assemblies and the core internals resulting from the postulated events .

Time history numerical integration methods are used as a standard design analysis for obtaining dynamic responses of fuel assemblies and their components. The solution based on the time history method is the most representative response resulting from a postulated transient loading. This method requires extensive computational time and analytical modeling techniques.

To verify the analytical models and computational procedures, mechanical tests using prototype assemblies and components are performed. These tests, static and dynamic, are conducted in order to identify the specific parameters and to verify analytical models under design accident loads.

Development of a reactor core model is a complex task, because the fuel assembly and the whole reactor core model representations are non-linear in both axial and lateral directions. By modeling with nonlinear impact elements, employing direct integration schemes, and incremental time step loading techniques of the finite element method approaches, the local responses to transient loadings can be accurately obtained.

During a transition core operation, fresh fuel assemblies are placed next to the existing fuel assemblies. The locations of the different types of fuel assembly designs in a transition core are based on the core loading patterns. In order to assess assembly behaviors and interactions between two designs in a transition core, the impacts of seismic and LOCA loads are based on the expected fuel assembly configurations in the transition cores and are normally provided through the compatibility information .

The fuel assembly parameters and models are properly adjusted to obtain the representative transient response.

Results from these analyses permit to verify the adequacy of the proposed design and in case it is required the need for deign modifications required for improving the structural performance

Hydraulic and Thermal-Hydraulic Performance

The other relevant design aspects that potentially affects the safety margins are the hydraulic and thermal hydraulic characteristics of the new and resident fuel designs

Fuel Assembly Hydraulic resistance characteristics

In a mixed core with assemblies having different hydraulic resistance, the local hydraulic resistance differences are a mechanism for flow redistribution. This redistribution results in the fluid velocity vector having a lateral component as well as the dominant axial component. The lateral component is commonly referred to as cross-flow.

The cross-flow induced by local hydraulic resistance differences will typically impact the safety analyses of the core. In the safety analyses, cross-flow affects DNB. DNB is affected because the flow redistribution affects both mass velocity and enthalpy distributions.

The Methodology developed has been successfully applied for calculating the thermal-hydraulic transition core DNBR penalty for a full range of plants and different fuel designs:

The analysis which determines the transition core DNBR penalty uses a very detailed fuel assembly THINC-IV (standard subchannel analysis Code) code model. The configurations used are three-by-three and five-by-five assembly arrays with different fractions of new assemblies and resident fuel in different arrays. Various plant conditions, including: Nominal, Overpower, Loss of Flow Transient and any other limiting DNB condition, including also the effect of different axial power distributions, are considered for the different homogeneous core configurations and also the different mixed core loading pattern configurations which are analysed. The results compared to a core with all assemblies of the same type yield the transition core DNBR penalty for each mixed core configuration.

To determine a DNBR penalty, an analysis is performed for the limiting transition core pattern, over the noted ranges. The analysis is then repeated for a three-by-three assembly array with all the assemblies being the new fuel design The remaining of the transition core patterns are used to calculate a DNBR penalty at the limiting core conditions previously established. The maximum DNBR penalties are then correlated to the fraction of the limiting fuel assemblies in the core or array being analysed.

Then, as a core transitions from a lower hydraulic resistance fuel assembly to a higher hydraulic resistance fuel assembly, the transition core DNBR penalty starts at a maximum value and continually approaches to zero when the transition is complete. In safety only analysis, a full core of one fuel type is analyzed and the appropriate transition core DNBR penalty is applied.

In the case where the hydraulic resistance of different assemblies are close, which is normally the desired situation , there is no transition core DNBR penalty.

Other effect that should be taken into consideration as a consequence of the crossflow effects is the impact on the mechanical design of the fuel assemblies. This could be affected if the crossflow excited peripheral rods in the fuel assemblies such that the wear mechanisms of fretting or whirling could exist.

Specific analyses are performed in order to assess the acceptability of the potential limiting crossflow conditions that might result from inlet core maldistributions in conjunction with pressure drop mismatches produced in the mixed core configurations

Other aspect taken into consideration, although normally of a minor impact in mixed core configurations, is that due to the fact that the fuel assembly with a different hydraulic resistance changes the flow distribution in the surrounding assemblies. In particular, if this fuel assembly has a higher value of fuel assembly loss coefficient, the surrounding assemblies will see a higher average flow through them than they would in a full core situation. Thus, the lift force on these surrounding assemblies can be expected to increase.

DNB performance compatibility

Besides the hydraulic effects affecting the DNB of mixed core configurations one of the most important aspect affecting safety margins is the intrinsic DNB performace of the fuel. Adequate DNB margin should be demonstrated for the new fuel design by means of the applicable CHF correlation. This is a basic requirement of the acceptability of the fuel design. Nevertheless, it is important to mention that the use of improved methods permits to increase the DNB margins for different uses like, for instance, to compensate for the potential DNB penalties associated with the transition cores, or for other plant needs In particular, the first aspect, i.e. to compensate any margin shortcoming resulting from transition configurations, guarantees that there is will be no reduction in safety margin as a consequence of the introduction of the new fuel designs if used in conjunction with these improved thermal hydraulic tools. These improved methods range from the use of advance CHF correlations like the WRB-1 and WRB-2 (associated with the use of additional flow mixing grids in the DNB limiting region) to the statistical methodologies in which uncertainties in plant operating parameters (Core flow , core power, coolant temperature , system pressure), nuclear and thermal parameters, design and transient codes are statistically combined such the DNB design criterion is satisfied

ENUSA has performed different fuel transition evaluations whether as a consequence of the introduction of advanced fuel design replacing previous fuel manufactured by ENUSA for the domestics plants, or due to the introduction of our fuel designs in other vendors plants in Europe.

Examples of the transitions of the first case are: from the Standard 17x17 to the OFA and from these two, to the AEF and the AEF with Intermediate flow mixers spacers. Normally these applications have been accompanied with the introduction of advance thermal-hydraulic methods and correlations. Examples of the second case are from different members of the Fragema´s AFA (14`) family to the AEF XL designs. These evaluations have also incorporated new materials like the Westinghouse´s Zirlo™ and the use of gadolinium as integral burnable absorber.

Conclusions

ENUSA has been involved in a wide range of fuel transitions, covering both the introduction of major design changes in our own fuel designs and also for the introduction of our products. In all cases a complete compatibility assessment has been performed and the adequate safety margins have been confirmed. In some cases specific design modifications have been introduced to minimise the transition core effects, in other cases the incorporation of advance design methods have permitted increasing the design margins to overcome the associated penalties.

Belgian Licensing Requirements: Mixed Cores and Control Rods Insertion Problem Aspects

N. A. Hollasky
AVN

Abstract

The purpose of this paper is double: on one hand it summarizes the different items to be addressed when performing a compatibility verification for mixed cores, pointing out more particularly the Belgian safety requirements, and on the other hand it approaches the licensing aspects associated with the control rods insertion problem recently encountered in Doel 4.

1. Belgian safety requirements for mixed cores

1.1. Introduction

When fuel assemblies of a new design (different from the design of the existing ones) are introduced into the core, the compatibility of the new reload with the NSSS and with the co-resident assemblies has to be demonstrated. More specifically, it has to be verified that the introduction of the new fuel elements can be realized while respecting the principles, rules and criteria enacted in the Safety Report (Departure for Nucleate Boiling (DNB) criterium, fuel melting limit, cladding maximal temperature, ...).

The compatibility studies may lead to modifications or penalties to be applied to either type of assemblies in order to respect the above-mentioned criteria. For example, the respect of the DNB criterium in mixed cores can lead to impose a penalty on the hot channel factor of either type of fuels; this aspect will be developed in detail further.

1.2. The compatibility report

The compatibility report is established by the supplier of the new type of fuel assemblies and has to be transmitted to the Safety Authorities six months before the planned date of the beginning of the core loading so that they can be convinced of the safety of the new fuel; this requirement is explicitly mentioned in the Royal Decree of Authorization of each nuclear power plant.

The report contains a detailed description of the new assembly and all the results of the studies and verifications performed to demonstrate:

- the geometrical compatibility with the existing surroundings (core, fuel handling equipment, storage, ...),

- the mechanical compatibility with the adjacent elements and with the associated components such as control rods, thimble plug assemblies, ...,
- the thermal-hydraulic compatibility with regard to the pressure losses, cross flows, by-pass flows, ...,
- the neutronic compatibility,
- the loss of coolant accident (LOCA).

These different items are addressed more in detail hereafter.

1.3. The geometrical and mechanical compatibilities

The design requirements to be verified are as follows:

1. Compatibility with the internal core parts

- The alignment holes of the fuel assembly's top and bottom nozzles shall match the upper and lower core plate guide pins without interference, and shall correctly position the fuel assembly.
- The alignment holes in the fuel assembly's top and bottom nozzles shall have sufficient guiding chamfers to allow the core plate guide pins to enter.
- The core plates guide pins shall always stay in contact with the nozzles of the fuel assembly.
- Upper and lower surfaces of the fuel assembly shall be parallel in order to give an equally distributed load.
- The fuel assembly positioning on the periphery shall provide adequate spacing to the baffle plates.

2. Compatibility with the instrumentation

- The instrumentation probe shall enter the fuel assembly without bending.
- The design of the instrumentation tube shall allow for adequate cooling of the instrumentation probe.

3. Compatibility with existing fuel assemblies, loading and unloading

- The fuel assembly shall be designed to provide adequate overlap at each grid elevation between adjacent fuel assemblies of different burnup and with different types of grids in order to preclude cross flow between such adjacent assemblies.
- The beginning of life (BOL) position of the fuel column shall be the same as for the reference fuel.
- The clearance between different fuel assemblies and between fuel assemblies and baffle plates shall always allow fuel handling in the core.
- The fuel assembly's nozzles and grids shall be provided with guiding chamfers to avoid adjacent assemblies getting hooked into each other during handling.

- The fuel assembly's top nozzle shall fit the gripping device and have a hole that uniquely determines the orientation of the fuel assembly.

- The fuel assembly's top nozzle shall have engraved identification numbers.

4. Compatibility with the control rods

- The positions and the dimensions of the guide thimble tubes shall be compatible with the geometry of the rod control cluster assembly (RCCA).

- The guide thimble tubes with their dash-pot zone shall be designed so as to give adequate damping of the drop velocity and acceptable impact load on the top nozzle.

- A sufficient axial clearance must exist to allow full insertion of control rods into the fuel assembly's guide thimble tubes.

All the preceding requirements are related to the geometrical characteristics and the dimensions of the fuel element and are generally satisfied without problem.

The RCCA compatibility is worth being developed a little more:

The RCCA spider hub spring retainer impact velocity during a scram shall be less than the velocity corresponding to the maximum impact force that avoids impact damage to the RCCA spider assembly and the top nozzle adapter plate; the order of magnitude of this maximum impact force is between 3500 and 4000 N depending on the plant.

The rodlets of core component assemblies must be able to be fully inserted into the fuel assembly without contact between the rodlet tip and the bottom end of the dashpot or thimble screw tip. A positive gap must exist under worst tolerances. In addition, the compression of the RCCA spring retainer, the compression of the fuel assembly due to the fuel assembly holddown springs, and the extension of the RCCA absorber rodlets due to deceleration, must be considered. This requirement provides assurance that the RCCAs can function without damaging the fuel assembly.

The RCCA rod drop time shall not be significantly different from the value given in the Technical Specifications so that the accident studies can not be questioned. The main fuel assembly parameters that affect control rod insertion time till the entrance in the dashpot are the inner diameter of the guide thimble and the pressure drop in the core.

Under the action of irradiation the fuel assembly length will increase. Interface between the assembly and the core plates could result in distorted assemblies and overloads on the internals. The distance between the top of the fuel assembly and the upper core plate shall be enough to preclude contact during the fuel assembly life. The maximum predicted growth is calculated at the fuel assembly design discharge burnup.

1.4. The thermal-hydraulic compatibility

The new fuel assembly must meet all the thermal-hydraulic performance requirements and not limit the power capability of the plant; it must be compatible, from thermal-hydraulic point of view, with all the other fuel assemblies potentially loaded into the core. Therefore, some criteria have been defined:

- The overall hydraulic resistance of the new element should be comparable to the one of the existing fuel assembly and should no prevent reaching 100% of rated thermal design flow. Any change in fuel assembly hydraulic resistance with respect to the current fuel assembly design in excess of a few percents would affect the reactor coolant system flow.

  The relative fuel assembly components loss coefficients for the different fuel assemblies present in the core are calculated and compared among themselves.

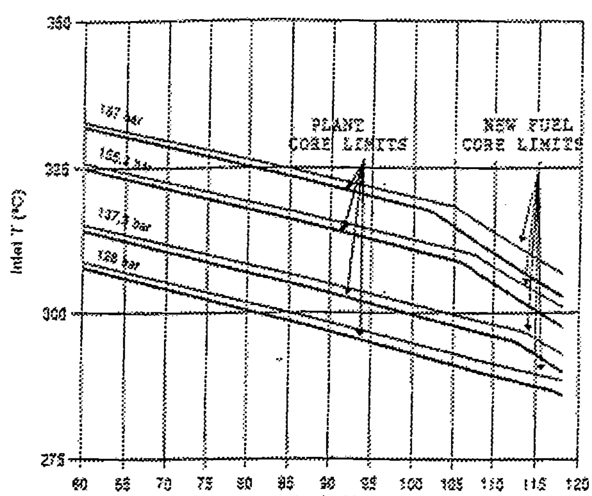
- The new fuel assembly must not increase the core by-pass flow due to the thimble tubes and instrumentation tubes beyond the allowable limit (for instance 2% -for Doel 4- or 2.3% -for Tihange 2- of the primary system coolant flow).

- The fuel assembly shall be protected against lift-off under all conditions I and II events with the exception of the turbine overspeed transient associated with a loss of external load. Under turbine overspeed transient conditions fuel assembly lift-off is permitted; however, no damage to the hold-down springs shall occur which would impair their continued use after the transient. Lift-off must also be limited so that contact between bottom nozzle and the lower core plate guide pins is always maintained.

The lift forces are calculated in hot full power, hot pump overspeed and cold zero power conditions and the corresponding required holddown spring forces in order to prevent lift-off are compared with the actual ones.

In reload mixed cores containing different fuel types, localized hydraulic resistance mismatches may cause local redistributions which can degrade the heat transfer capability and result in localized DNB penalties which could in turn limit power capability.

In order to verify the thermal-hydraulic behaviour of the new fuel assembly in a full core configuration and to justify its compatibility with respect to the currently fuel loaded into the core, a statistical method is used; this method combines statistically the uncertainties on some parameters to derive the design limit DNBR to which the required margins and penalties are applied in order to reach the safety analysis limit DNBR (DNBRSAL). The aforementioned parameters are then considered at their nominal value in the calculations. With the DNBRSAL value, the core thermal limits for the new fuel are generated and it is verified that these limits are bounded by the core limits of the plant. If not, these last ones have to be modified. The figure hereafter gives a typical example of this approach.



Several manufacturers supply fuel elements in the Belgian power plants; consequently, several statistical methods and several critical heat flux correlations have been submitted to AVN for licensing. The present situation for the five most recent plants is summarized hereafter:

Power Plant	Fuel Supplier	Fuel Type	Co-resident fuels	CHF Correlation / DNB code	Statistical Method
Doel 3	SIEMENS (KWU)	17x17 FOCUS	AKA, AKA-PA and AFA-2G	ERB-3 / COBRA3-CP	SSTDP
	FRAGEMA	17x17 AFA-2G MOX	AKA, AKA-PA and FOCUS	WRB1 / FLICAIII-F	MSG
Doel 4	ENUSA	17x17 AEF XLR	FRA STD-XLR and AFA-XLR	WRB-1 / THINCIV	RTDP
Tihange 1	FRAGEMA	15x15 AFA-2G	ANF	WRB1 / FLICAIII-F	MSG
Tihange 2	ABB	17x17 ABB	AFA and AFA-2G	ABB-X1 / TORC	MASC
	FRAGEMA	17x17 AFA-2G MOX	ABB and AFA	WRB1 / FLICAIII-F	MSG
Tihange 3	FRAGEMA	17x17 AFA XR2	AFA XLR and AFA XR1	WRB1 / FLICAIII-F	MSG

In the past, statistical methods such as ITDP and mini-RTDP from Westinghouse, and MS from Framatome, have also been evaluated.

1.5. *Belgian safety requirements*

a. All these original methods combine statistically the uncertainties on the core flow rate and on the by-pass flow rate; now such parameters do not statistically fluctuate as a function of time for a given plant: the associated uncertainties result only from the inaccuracy of the measurements and according to AVN, may not be statistically combined. They have to be considered in a deterministic and penalizing way.

   On AVN request, the two parameters were thus removed from the statistical combination by the manufacturers when applying the methods in Belgium.

b. The most recent methods include, in the statistical combination, the uncertainty on the DNB correlation. This releases an important DNB margin and reduces accordingly the safety margin. In a philosophy of "defence in depth", safety margins are useful to cover unknown phenomena, such as deficiency of rigour in the application of the statistical methods, introduction of unjustified hypothesis, incomplete demonstrations, ...

   Therefore, a full reduction of the safety margin did not seem acceptable to AVN and a 4% generic margin on the design limit DNBR has been imposed.

c. The statistical combination of the DNB correlation uncertainty distribution is not without concern. Indeed, when a DNB correlation is used for a particular type of fuel assemblies, most often the associated data base contains data subsets corresponding to this specific fuel. The uncertainty distribution of critical heat flux data for a subset might be different from the uncertainty distribution of the whole correlation, thus introducing a bias. Rigorously, such a bias cannot be accepted: the statistical combination process for a non uniform biased sub-population would be mathematically wrong. However, from a safety point of view, if the bias is in a conservative sense, - i.e. the mean value of the sub-populations higher than the mean value of the correlation and the standard deviation lower - the approach may be accepted . Moreover, as more than one subset of data should be considered, the normal distribution of each set, the homogeneity of the sets and the normal distribution of the combined sets have to be demonstrated at a 5% rejection level.

   All these verifications have to be performed by the fuel manufacturer.

d. The rod bow penalty has also to be taken into account in the determination of the safety analysis limit DNBRSAL. The rod bow models have been licensed by AVN for all the types of fuel present in the Belgian cores and the DNB rod bow penalty was required to be applied on a deterministic way, i.e. on a multiplicative way on the design limit.

   In the past, the rod bow effect amounted about 4% for most of the fuel types loaded in the Belgian cores, with a few exceptions relative to fuel length, grids number or burnup range. The manufacturers using at that time statistical methods (Framatome, KWU, ...) applied a "provision" for rod bow of 6.5% so that the total penalty on the design limit DNBR amounted about 10% (6.5% + 4% generic margin(see point b. hereabove)). Subsequently, by regard for coherence, AVN has imposed this order of magnitude of global penalty to all the manufacturers.

   More recently, owing to the experience feedback, the rod bow models have been improved by almost all the fuel manufacturers so that the corresponding DNB penalties were inclined to decrease. The value of 6.5% has become exaggeratedly high and AVN accepted the application of the new calculated rod bow penalties on the condition to add a 1.5% margin as "reserve". Moreover, the burnup range covered by the penalty has to be mentioned in the safety evaluation studies and it must be verified, for each cycle, that above the retained burnup breakpoint the power delivered by the fuel assemblies is low enough to prevent rod bow effect.

1.6. *Transition cores analysis: DNB aspects*

Redistribution of flow occurs generally because of thermal-hydraulic fluid condition gradients within the core. In a mixed core with assemblies having different hydraulic resistance, the local hydraulic resistance differences are a mechanism for flow redistribution. This redistribution results in the fluid velocity vector having a lateral component besides the dominant axial component. The lateral component is commonly referred to as crossflow. The crossflow induced by local hydraulic resistance differences will impact the safety analyses of the core, and more particularly the DNB because the flow redistribution affects both mass velocity and enthalpy distributions.

Core configurations with different amounts of the new fuel among the resident one are modelized and it is examined and which loading situation leads to minimize the flow rate in each type of assembly. The DNB code is then used to analyse the impact of this situation on the minimum DNBR of the reference core and a eventual penalty is determined.

If no margin, during the transition this penalty can be offset by a trade off between the DNBR penalty and the penalizing type fuel assembly F(H limit.

1.7. *Transition cores analysis: large break LOCA aspects*

It is a matter of determining whether a greater calculated peak clad temperature (PCT) can occur for the transition configurations considered; the verification is performed for both blowdown and reflood phases.

The initially stored energy within the fuel is fuel dependent and can influence relatively strong the first temperature peak. The difference in initially stored energy is indirectly addressed by the difference between the calculated fuel cladding average temperatures.

In the reflood phase, for a given peaking factor FQ, the only mechanism potentially causing a transition core to have a greater PCT than a full core of either fuel is through the flow redistribution due to fuel assembly hydraulic resistance mismatch. In the presence of assemblies with reduced flow area, the associated PCT penalty must be calculated and compared with the existing margin in the reference study. In case of criteria violation (> 1204°C), the allowed value for FQ must be reduced.

1.8. *The neutronic compatibility*

The differences between the two fuels from a neutronic point of view (pellet density, diameter and wall thickness of the guide thimbles and the instrumentation tube, material and weight of the spacer grids) are analysed.

To demonstrate the neutronic compatibility between a new fuel type and the old one, the following calculations as a function of burnup are performed:

- comparison of the fuel assembly reactivity at different operating conditions (HFP, HZP and CZP),
- comparison of the uranium and plutonium isotopic inventory,
- comparison of the xenon and samarium reactivity worths,
- comparison of the moderator temperature coefficient and Doppler power coefficient. Also, the fuel resonance temperatures used as the basis for the Doppler coefficient calculations are compared,
- comparison of the prompt neutron lifetime and effective delayed neutron fractions,
- comparison of the control rod reactivity worth,
- quantification of the power distributions (rod-wise power distribution, relative assembly power distribution, peak integrated rod power (F(H)) impact of the presence of the new fuel assembly next on the adjacent old one.

The fuel rod behaviour must also be verified: the objective of the studies is to demonstrate that the fuel rod preserve its mechanical integrity during normal operation and under condition II transients.

More particularly, it has to be demonstrated that there is no risk of rupture due to excessive stress in the cladding, more particularly for the condition II transients

Therefore according to the Safety Report, two criteria must be satisfied: the maximum fuel centerline temperature must remain lower than the melting limit and the DNBR limit must be respected; the first one has been expressed in terms of maximum lineic power in overpower conditions (OPDT trip set point 118%).

The melting temperature for fresh UO2 fuel amounts 2800°C: this value is reduced to take into account the burnup effect, fabrication tolerances, model uncertainties, ...

AVN required that an additional margin be considered in order to cover the effect of unknown phenomena or of phenomena not taken into account in the criteria such as internal pressure pellet clad interaction, stress in the cladding, strain of the cladding which can damage the fuel rod. This requirement is especially justified as all the manufacturers tend to increase the enrichments and the discharge burnups.

2. Control rods insertion problems in Doel 4: Licensing aspects

The control rods insertion problems encountered in Doel 4 during the last cycle have been described in detail in [1]; all de verifications and tests performed by the Utility as well as the conclusions of the investigations of the concerned manufacturers are covered by this paper so that only the licensing aspects will be addressed here.

The design of a nuclear power plant requires a very high confidence in the trip system; failure of this system has never been considered except for a number of accident studies in which one stuck rod is supposed. A situation with several control rods stuck at different levels in the core is beyond the design and has never been evaluated

It would be very difficult to analyse these situations because the existing methods and models do not cover such configurations and also because there are to many possibilities to easily define the worst one. Moreover, the root cause of the problem is still unknown.

Consequently, AVN required for the next cycles that the initial level of reliability in the trip system remains unchanged.

The actions proposed by the Utility have been evaluated by the Licensing Authorities:

The loading pattern with only fresh or one cycle assemblies under the control rod, the fresh assemblies preferentially in the center of the cores and a criterion for the friction force measured in the one cycle elements have been accepted.

On the basis of the available measurements results and evaluations performed to day, AVN is not convinced that the actions taken by the Utility are sufficient to prevent any stuck rod problem to occur during cycle 12. Consequently, additional periodical controls of the availability of the control rod system have been required, i.e. rod drop time measurements at different moments of the cycle expressed in terms of average assembly burnup for the assemblies containing a control rod. Accordingly, a start-up license has been delivered for a limited period of time (up to 20 000 MWd/t for the most burned rodded assembly).

Due to the uncertainty existing in the evaluation of the problem and in order to keep a sufficient "defence in depth", AVN required that the effect of a failure of the trip system and its consequences be evaluated.

The shutdown margin has been calculated assuming a number of penalizing configurations with stuck rods. It has been shown that even for the most penalizing situation, the criteria remained met (cycle 12 shows an important margin with respect to the limit: 4380 pcm vs 2700 pcm).

For the same configurations, it has been verified that the anti-reactivity insertion curve considered in the accident studies remained bounded.

Finally, considering that the assumption "all rods minus one" (ARI-1) inserted in the core and the choice of the stuck rod at the beginning of the transient are very important in the steam line break accident evaluation, AVN asks that more than one stuck rod and different levels be considered. These verifications should be finalized before any rodded fuel assembly reaches 20 000 Mwd/tU.

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[1] Belgian operating experience with RCCA behaviour. H. de Baenst et al Muting on Nuclear Fuel and Control Rods: Operating Experience, Design Evolution and Safeti Aspects. Madrid, 5-7 November 1996.

BWR Fuel Designs for Extended Operating Domains

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Abstract

Fuel design entails a complicated process of trade-off actions which impacts core thermal and reactivity margins. The implementation of plant specific operating flexibility options such as the extension of the operating domain requires the expenditure of some of these margins. Therefore the optimum fuel design is very much dependent upon the operational environment and utility specific priorities. This paper discusses recent advances in BWR fuel design which allow for better fuel utilization while extending the plant's operational capabilities.

1. Introduction

Boiling Water Reactor (BWR) operating domain is defined, at a macro level, in terms of the power/flow map, since once power and flow are defined the rest of the reactor heat balance parameters are basically known, i.e., reactor pressure, feedwater temperature, steam flow, core inlet enthalpy, etc. Figure 1 shows a standard operating domain for conventional or advanced BWR with recirculation pumps.

Reduction of power generation costs, improvements in fuel cycle economics and enhanced operational flexibility have lead the entire BWR fleet to extend the initial operating domain, as indicated in Figure 1.

However, power and flow by themselves do not represent the entire fuel operating domain. Fuel operation is also affected by micro parameters, such as core power distribution (radial, axial and local), which in turn are largely controlled by the fuel enrichment and the reload batch size.

Reload batch size is basically set by the desired batch average discharge exposure. Meanwhile fuel enrichment can be determined as a function of the desired cycle length or cycle energy, based upon the linear reactivity model (Reference 1).

Therefore, the fuel operating domain can be visualized as a four-dimensional (4-D) space, with axes being power, flow, discharge exposure and cycle length (or cycle exposure). Of course a 4-D chart can not easily be pictured; instead another complementary 2-D map, the Region of Application chart, can be used (see Figure 2). This chart represents the ranges of discharge exposures and cycle lengths (or exposures) that a particular fuel design can be operated.

Extensions in the power/flow map does not necessarily require new fuel design features, although for optimum fuel cycle economics improved designs are highly advised. On the other hand, extensions of the region of application generally require improved or advanced fuel designs; more specifically improvements in the thermal margins, discharge exposure capability, and reactivity margins are required (see Figure 2).

In this paper we analyze the bases of how changes in the operating parameters impact the fuel operating performance. We also review the more frequent technical solutions adopted by the fuel vendors to cope with the more and more demanding fuel operating environment.

2. Needs for Improved Fuel Operating Margins: Basic Concepts

Extended operating domains generally require operation at higher powers for a larger flow range (see Figure 1). Either the core thermal power is uprated (Power Uprate) or the flow window is enlarged (Flow Control Spectral Shift operation) for the same rated power.

Power Uprate (PU) is becoming an emerging issue in the last few years, and one of the best options to improve, at relatively low cost, the competitiveness of the nuclear industry. Almost every utility is considering or has already implemented PU.

Flow Control Spectral Shift (FCSS) operation, either extensions above the load line limit (ELLL for extended or MELLL for maximum extended) or beyond rated core flow (ICF), has being used for years and is a well known method to increase the nuclear efficiency by means of generating and burning plutonium.

Both PU and FCSS require additional operating margins. Following is a discussion of the basic concepts associated with these two options to extend the BWR operating domain. Core thermal, stability and reactivity margin aspects are considered.

Power Uprate increases the bundle average power proportionally to the uprate, and may cause some reduction of the operational thermal margins if no changes are introduced in the fuel or core design. However a "softer" (larger batch size) core loading or redesign of the bundle nuclear characteristics (improved R-factor and local peaking) could be considered in order to mitigate the impact of PU in the thermal margins.

For example, Figure 3 represents the change in critical power ratio (CPR) as a function of core thermal power, assuming no changes in other parameters (same core loading, same core flow). The variation of CPR with power is almost linear, although it is noticeably less than proportional. This is the result of a radial power peaking flattening due to the increased loss of reactivity in the hot assemblies relative to the average bundles in the core, and the redesign of the control rod patterns to maintain criticality.

PU does not adversely impact stability as long as the control rod lines are not expanded beyond the currently operating ranges, because the stability exlcusion region is not modified.

PU also affects the core reactivity margins, since an increase in the average bundle power reduces both the hot excess reactivity and the cold shutdown margin, i.e., reduces the so called hot-to-cold reactivity window (see Figure 4), which typically requires a core and/or a bundle redesign (see later discussion on this issue).

FCSS operation results in the loss of thermal margins both during the low flow period of operation (beginning through middle of the cycle) and, to a less extent, during the high flow period (end-of-the cycle).

Figure 5 shows a typical plot of CPR versus fuel assembly flow at standard operating conditions. As the flow is reduced the loss in CPR margin is evident. However, near EOC when the core flow is increased to its maximum allowable value, the CPR margin is also slightly penalized due to the very top peaked axial power distribution generated by this FCSS operation. Figure 6 illustrates this issue.

Stability margins tend to be degraded for FCSS operation due to the larger flow range, which produce higher power at low flows for control rod lines beyond the rated line, and the larger axial power peaking associated to this mode of operation.

FCSS operation tends to improve hot excess reactivity throughout the cycle due to the plutonium buildup. Shutdown margin is also improved through the cycle due to the beneficial effect of the power shift down to the bottom of the core during the majority of the cycle, leaving more Gd available at the top of the core where is needed for controlling reactivity at cold conditions. Plutonium buildup tends to penalize the SDM at EOC, but the axial power shape impact on the Gd content is the controlling parameter, and the net effect is favorable.

In terms of the region of application for the fuel (Figure 2), two possible extensions are considered, i.e., higher burnups and longer cycles.

An increase of the discharge burnup can be accomplished by reducing the reload batch size or reducing the bundle weight. Assuming a fixed mechanical design bundle weight will be invariable. A reduction of the batch size will increase the radial peaking, requiring improved CPR margin. Figure 7 presents a plot of the change in CPR margin versus batch average discharge exposure, assuming other parameters remain constant. The higher the discharge exposure the less the CPR margin. The main reason for this is the higher initial U-235 loading (and therefore hot excess reactivity) required for accommodating higher burnups; non-linearity is associated with the hydraulic feedback, which reduces the flow through the hotter bundles. High discharge burnups tend to be limited by the fuel exposure capability and by the fuel thermal margin performance, as indicated in Figure 2.

Higher burnups also tend to penalize the stability margins due to the associated higher radial power peaking.

On the other hand, longer cycles result in higher cycle exposure and require higher gadolinium concentrations for optimum nuclear efficiency. Also, since long cycles need higher enrichments, they require more gadolinium rods to control reactivity. Reactivity margins are typically characterized by the hot-to-cold reactivity window. Hot-to-cold reactivity window is defined as the sum of the minimum hot excess reactivity and the minimum cold shutdown margin (see Figure 4). BWR fuel is undermoderated at hot conditions, but it's slightly overmoderated at cold conditions. This implies that for a fixed fuel assembly mechanical design the hot-to-cold window will decrease with enrichment. Therefore the fuel assembly mechanical design for longer cycles needs to account for this effect in order to optimize the nuclear efficiency.

The necessary increase in the gadolinium requirements to accommodate long cycles penalize the bundle local power distribution, increasing the local peaking. Meanwhile, the higher bundle enrichments required for long cycles drive higher radial power peaking, which also tend to penalize thermal margins.

However, long cycles generally require larger batch fractions. Large batch fractions, on the contrary, tend to decrease the core radial peaking. This decrease in core radial peaking usually more than offsets the increase in local peaking. Thus long cycles are not typically limited by thermal margins, but by reactivity margins, as shown in Figure 2. On the contrary short cycles tend to be controlled by fuel thermal performance

Based upon the above discussions, long cycles don't have any major impact on stability margins due to the beneficial effect of the reduced core radial power peaking.

Before looking at the design solutions, let's review briefly the various phenomena which contribute to boiling transition or dryout in a BWR. Figure 8 shows a pictorial description of the dryout phenomenon. Boiling transition in a BWR is the result of the thickness of the liquid film surrounding the fuel rod approaching zero, and loosing its wetted condition. The spacer function is to replenish the liquid film on the fuel rod as the two phase mixture passes through the spacer. Therefore the limiting condition occurs just upstream of a spacer (typically the first or second spacer from the top of the bundle). As shown in Figure 9, increased spacer pitch is also an important factor in reducing the CPR margin, since the liquid film thins over a longer length prior to being replenished by the next spacer

3. Fuel Design Features for Extended Operating Domains

Once the basic impacts on core margins of the operating domain extensions have been introduced, we now turn to the main subject of this paper, which is how they affect the fuel design. Recent trends in fuel design, which help to cope with the very demanding operating domains, are presented below.

3.1. Increased number of fuel rods

One of the design features, common to all of the BWR fuel vendors is the trend to increase the number of fuel rods. The majority of BWR fuel is still an 8x8 lattice, but the trend to 9x9 and more recently to 10x10 lattices is increasing exponentially. The reasons for this worldwide trend are as follows:

1. More fuel rods per bundle leads to a reduction in the average linear heat generation rate. With this the allowable peak linear heat generation rate (PLHGR) can be reduced. Lower PLHGR limit permits lower fuel stored energy and reduced fission gas release rates. This in turn can be used to extend the discharge exposure or to reduce the fuel rod plenum, increasing the amount of UO_2 in the bundle, necessary for longer cycles.

2. The increased number of fuel rods allows a further optimization of the rod-to-rod power distribution, maximizing bundle nuclear efficiency, while maintaining or improving CPR and PLGHR margins.

3. Increasing the number of rods increases the fuel heat transfer area, reducing the average thermal heat flux. Lower heat fluxes tend to increase critical power capability (assuming an efficient pressure drop performance).

3.2. Additional water volume in the interior of the lattice

The addition of more water volume to the interior of the fuel lattice, by means of larger central water rods, internal square water channel or a water cross, provides increased and more uniform neutron moderation in the bundle (optimum hydrogen to fissile atom ratio). This has two inherent benefits:

1. Better nuclear efficiency: As indicated above, a more uniform power distribution allows for an overall reduction of bundle enrichment. Also the improved moderation provides a larger hot-to-cold reactivity window, helpful to achieve longer fuel cycles.

2. More uniform moderation, reduces local power peaking, which helps to improve thermal limits

3. The more water present in the lattice the less negative the void reactivity coefficient. This has two beneficial aspects:

- Milder neutron flux excursions during pressurization transients
- Improved stability performance

3.3. Part Length Rods

Two BWR fuel vendors have introduced Part Length Rods (PLR's) primarily to maintain pressure drop (DP) performance (the larger the number of fuel rods the higher the pressure drop, degrading core flow capability and most important stability).

A typical pressure drop distribution in a PLR bundle design is shown in Figure 10. Above the PLR the flow area is increased, reducing flow velocity and consequently pressure drop in the two phase region. This two phase pressure drop reduction allows an increase of the single phase DP in the lower tie plate. This drives a further reduction of the two phase to single phase pressure drop ratio, improving significantly the hydrodynamic stability of the fuel channel.

PLR designs also concentrate the fuel near the low void region, so that optimum moderation can be achieved without sacrificing fuel weight. Figure 11 presents a typical full power axial variation in H/U ratio; the PLR's serve to make a more uniform distribution and enhance neutron economy. Figure 12 shows how PLR's allow to increase the fuel weight without penalizing excessively core pressure drop. The increased fuel weight is helpful in maintaining efficient fuel cycle performance for long cycles.

3.4. Improved Spacers

As discussed in Section 2, spacers play a key role in the critical power capability of the fuel assembly. They are also important to maintain rod bow and flow induced vibration margins and contribute to the bundle pressure drop.

The optimum spacer design is the one which can increase the liquid film thickness surrounding the fuel rods, while maintaining the pressure drop at a reasonably low value.

Several different spacer design concepts are available, egg-crate, ferrule, etc., as shown in Figure 13. Two phase flow pattern along the spacer is the key for CPR performance. For a given design, the critical power capability can be increased at the expense of pressure drop, as shown in Figure 14. This is generally accomplished by additional lateral liquid flow (addition of flow tabs or mixing vanes, increased spacer thickness and height).

Also, decreasing the distance between the spacers or adding an additional spacer near the top of the bundle, where dryout is expected to occur, is another potential design option to improve CPR margins (see Figure 15). Therefore the number of spacers and the spacer pitch is a design feature that can be traded-off.

3.5. Axial Zoning (Gadolinium and Uranium Enrichment)

The combined axial enrichment and axial gadolinium loading for axial power shaping and cold shutdown margin have proved to be a very useful way to improve nuclear efficiency, minimizing the use of shallow control rods during operation, and reducing the content of undepleted gadolinium. This feature provides an optimum spectral shift operation, allowing for an overall reduction in the bundle average enrichment.

3.6. Optimized Channel

Fuel channels have been optimized in order to improve nuclear reactivity margins, pressure drop and critical power margins. For instance where possible the channel inner dimension has been expanded for reduced pressure drop and improved shutdown margin. Also, interactive channel flow trippers have been added in the top of the channel for improved CPR margins.

4. Summary and Conclusions

In summary, the fuel operating domain, understood in its more general meaning (power/flow and discharge-exposure/cycle-length maps), determines the required fuel performance. Fuel designers must produce a design able to meet the expected operating performance. Table 1 summarizes the most relevant fuel design features introduced to extend the operating domain.

The highly variable operating environment around the world sets different performance targets. Therefore, utilities need to be more and more involved in defining their specific priorities.

Since most of the design options have both beneficial and detrimental effects, the final optimum fuel design for the specific plant application will be the result of a very complicated trade-off process. Understanding the benefits of these trade-offs must be integrated with the utilities specific needs.

5. References

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[2] J.E. Wood (GE), "Advanced Fuel Design Thermal Margin", Presentation to the Swiss Association of Atomic Energy, May 7, 1991

[3] S.P. Congdon (GE) and J.J. Peña (ENUSA), "Low Linear Heat Generation Rate Fuel Designs for Boiling Water Reactors", Paper presented at the XIX meting of the Spanish Nuclear Society, Cáceres, Spain, October 1993.

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[5] A. Jackson (ABB), "Advances in BWR Fuel Design", Paper presented at the Topfuel'95, Wüzburg, Germany, March 12-15 1995.

[6] P. Urban, "ATRIUM-10 , The Highly Versatile Fuel Assembly for BWRs", Paper presented at the Topfuel'95, Wüzburg, Germany, March 12-15 1995.

Benefits	More Fuel Rods	More Water Vol.	Part Length Rods	Improved Spacers	Axial Zoning	Improved Channel
Improved Thermal Margins	+++	+	-	+++	++	+
Improved Reactivity Margins	+	+++	++	0	+++	+
Improved Stability Margins	---	++	+++	+	0	+
Improved Nuclear Efficiency	+++	++	++	+	++	++
Increased Burnup Capability	+++	0	0	++	+	0
Increased cycle energy	+++	++	++	+	+	+
Reduced core pressure drop	---	0	+++	++	0	++
Reduce Fuel Cycle Cost	++	++	++	+	++	+

0 means no impact

- means adverse impact

+ means positive impact

the relative importance is from 1 to 3

Table 1
Relative Importance of BWR Fuel Design Features
for Extended Operating Domains

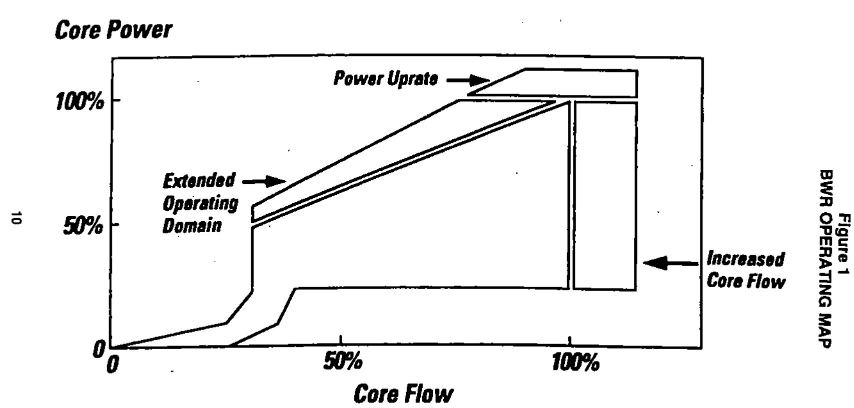

Figure 1. BWR Operating Map

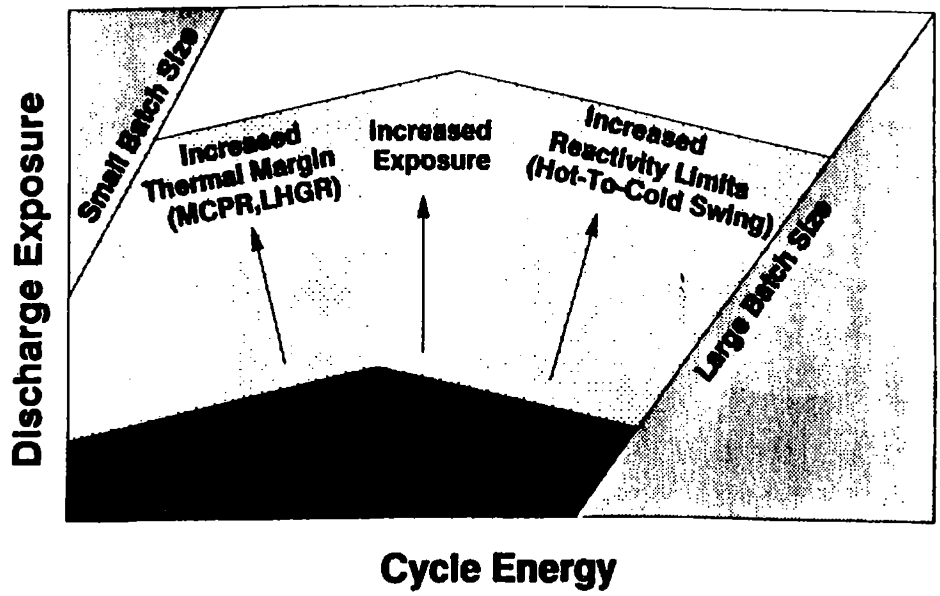

Figure 2. REGION OF APPLICATION. Cycle Energy vs. Discharga Exposure

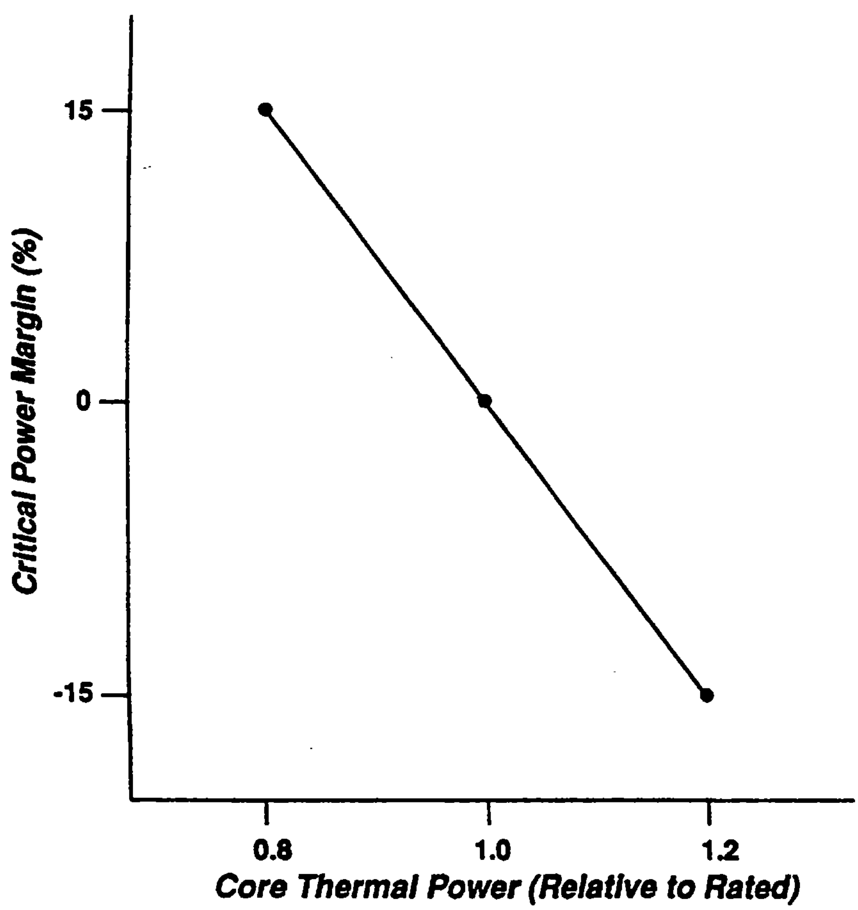

Figure 3. POWER UPRATE IMPACT ON CPR

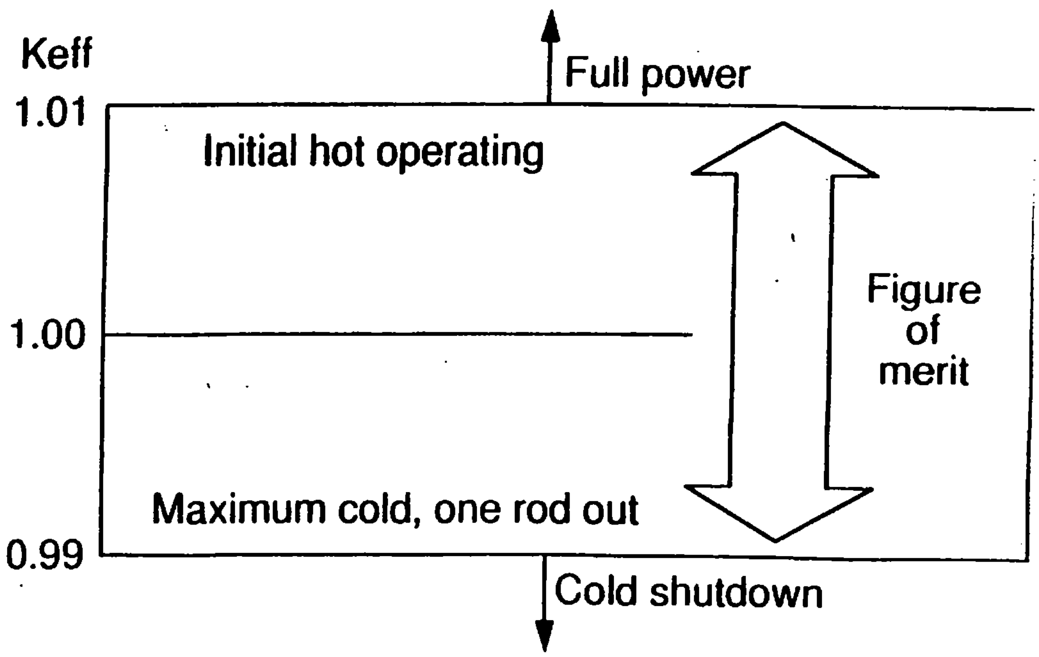

Figure 4. HOT-TO-COLD SWING

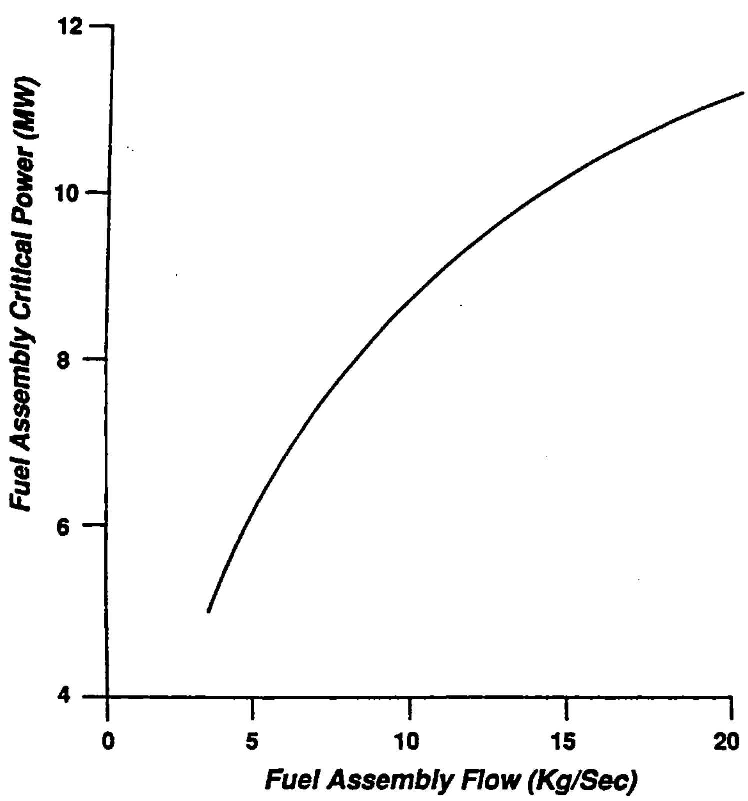

Figure 5. CRITICAL POWER Vs. FLOW

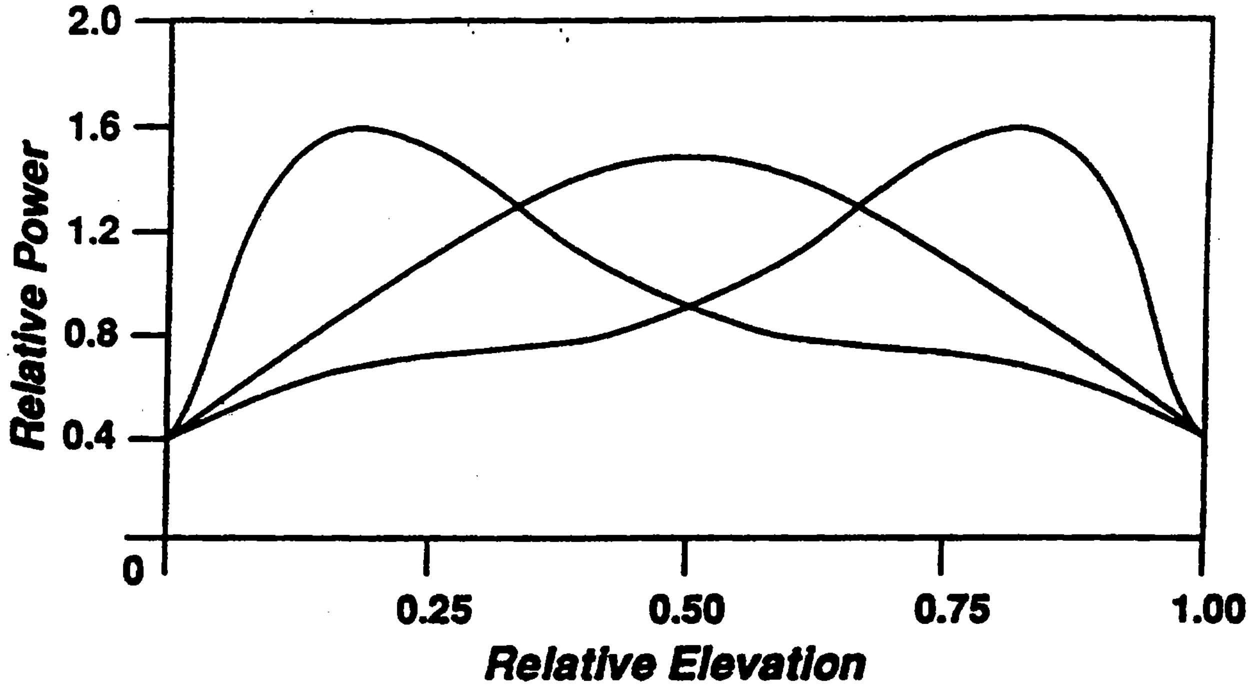

Figure 6A. Power Distribution

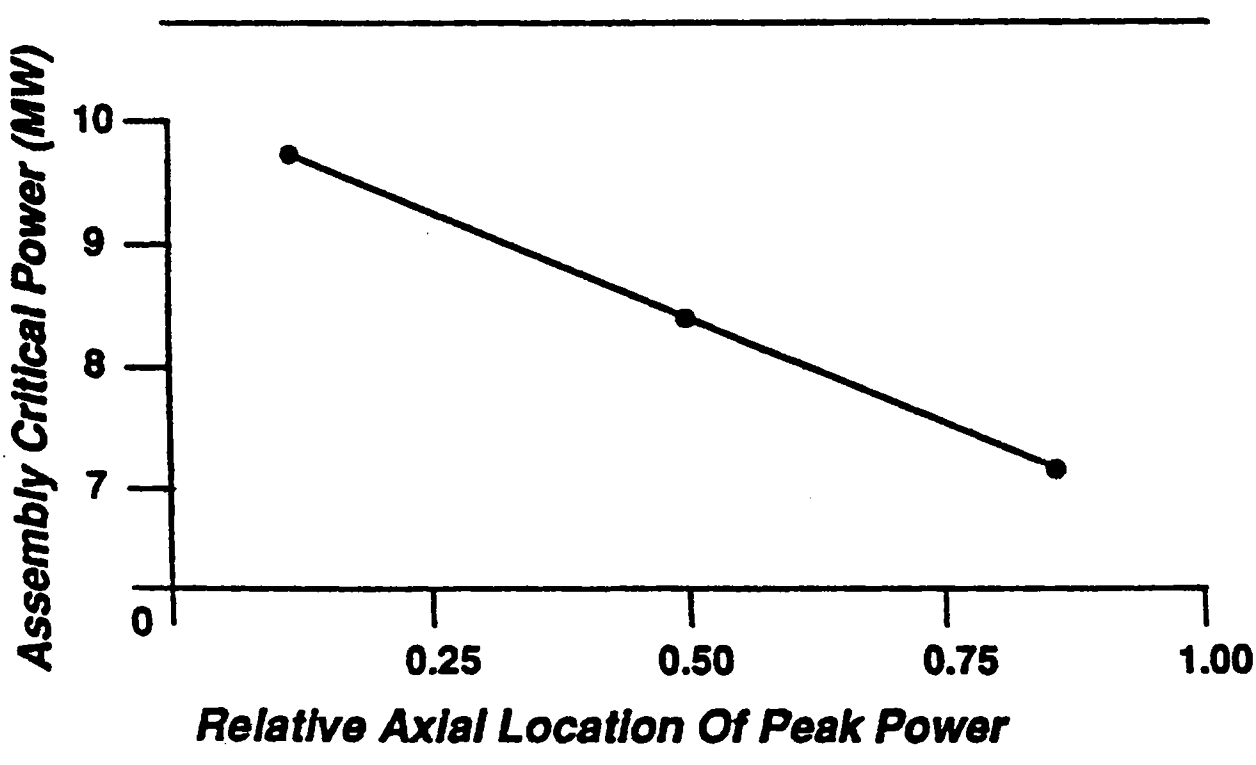

Figure 6B. Critical Power Location

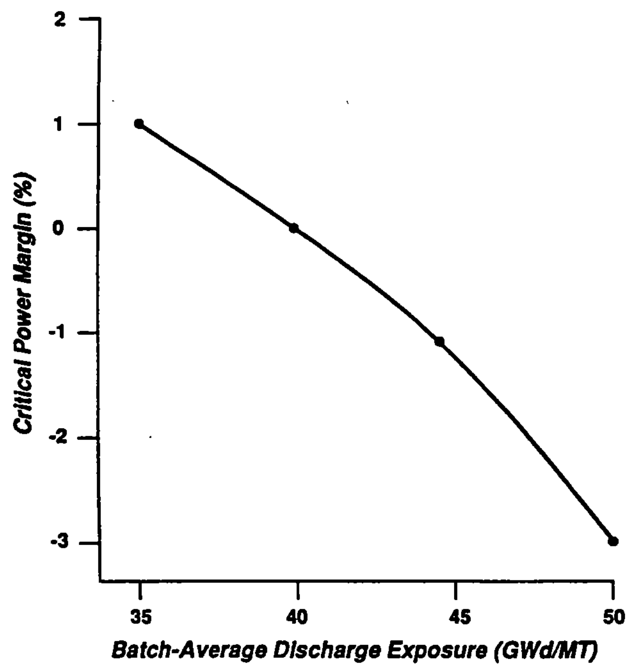

Figure 7. INCREASED BURNUP IMPACT ON CPR

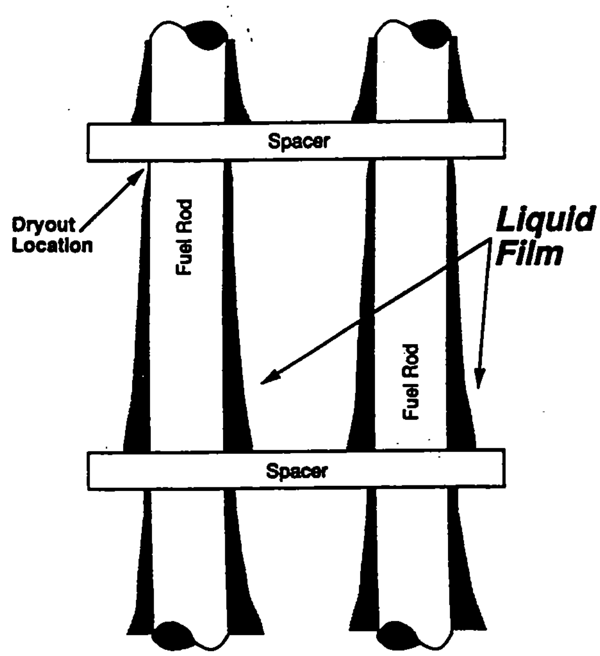

Figure 8. BWR LIQUID FILM DRYOUT PROCESS

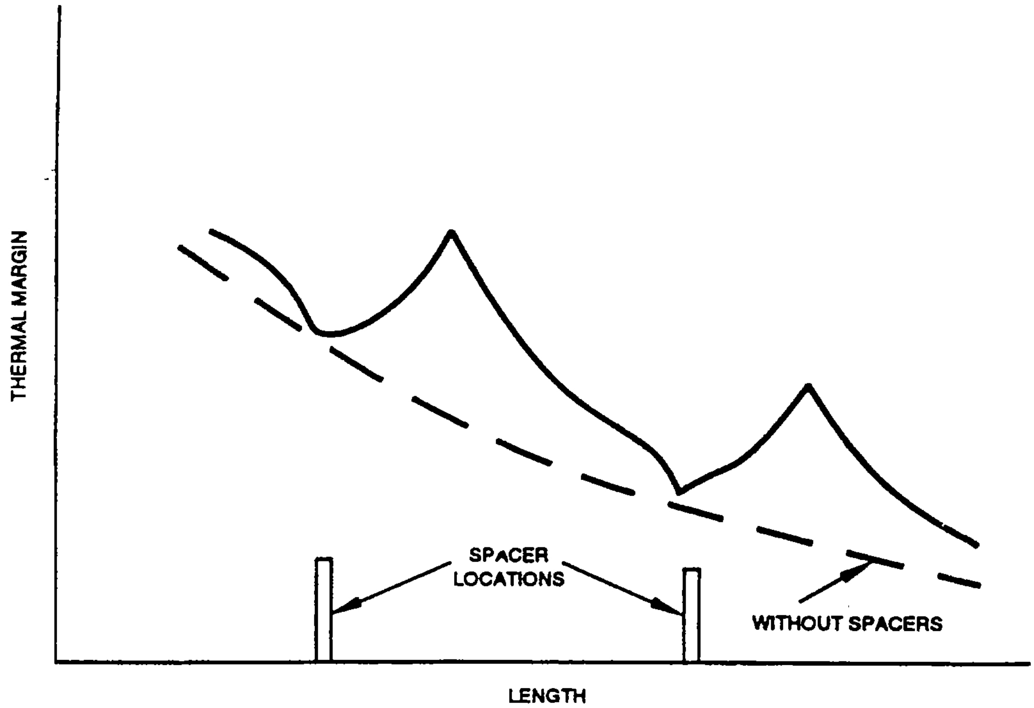

Figure 9. VARIATION OF THERMAL MARGIN ALONG BUNDLE LENGTH

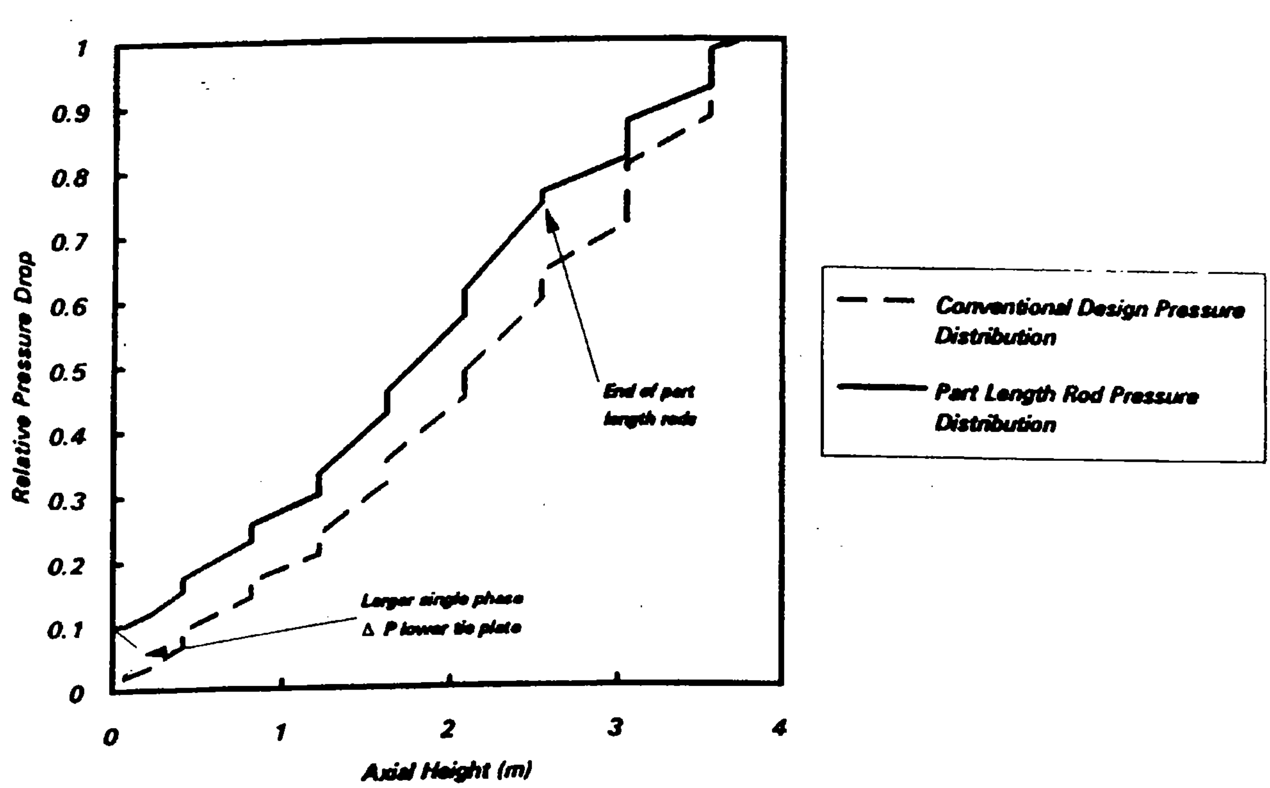

Figure 10. PART LENGTH FUEL RODS AXIAL PRESSURE DROP DISTRIBUTION

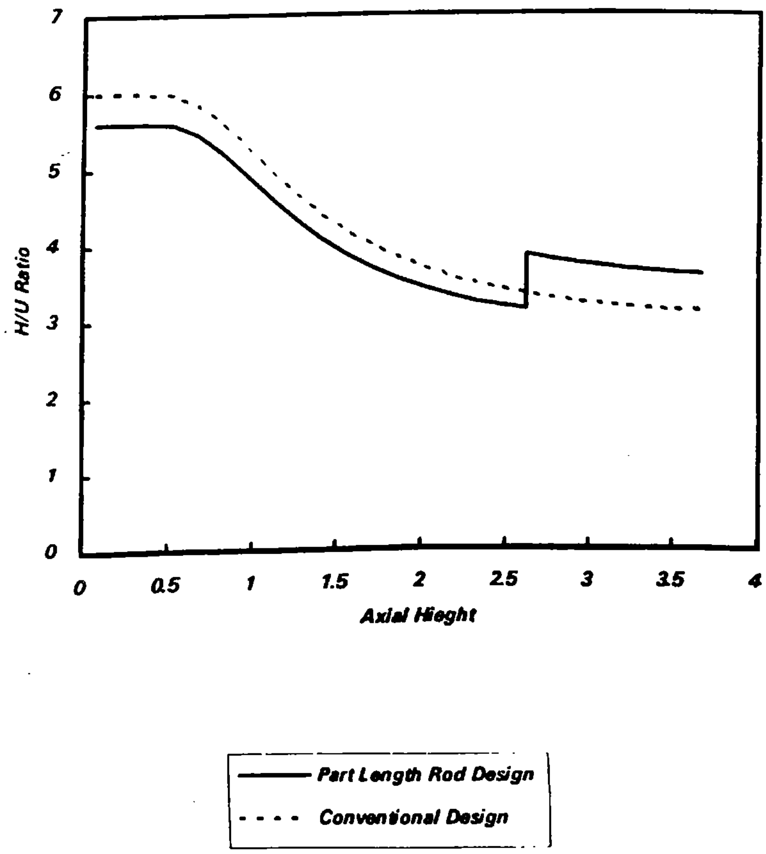

Figure 11. PART LENGTH FUEL RODS AXIAL H/U DISTRIBUTION

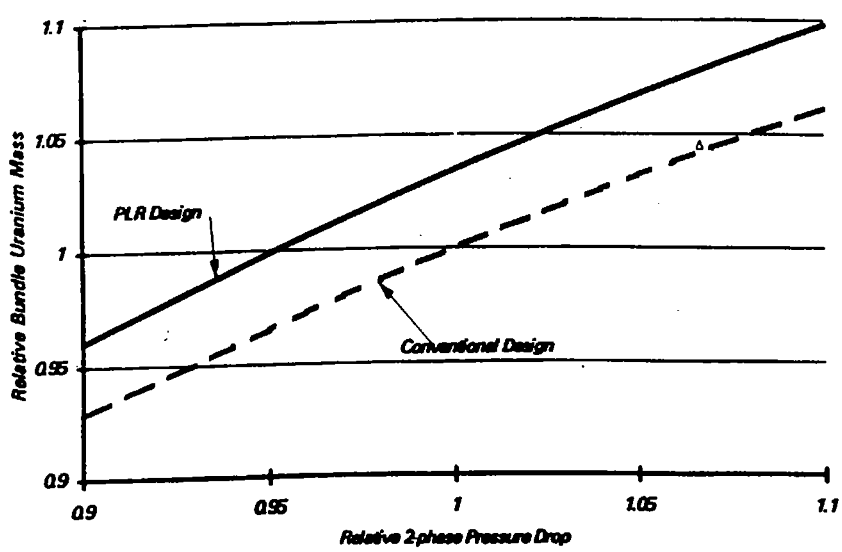

Figure 12. PART LENGTH FUEL RODS MORE WEIGHT PER BUNDLE

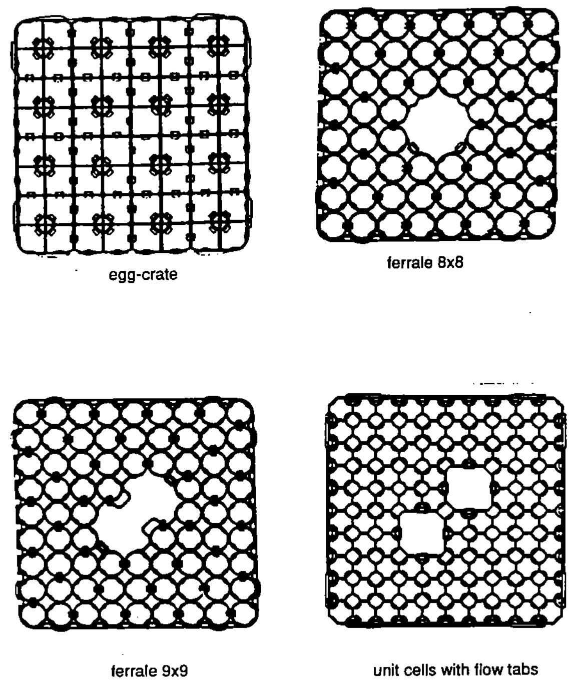

Figure 13. HIGH PERFORMANCE SPACER

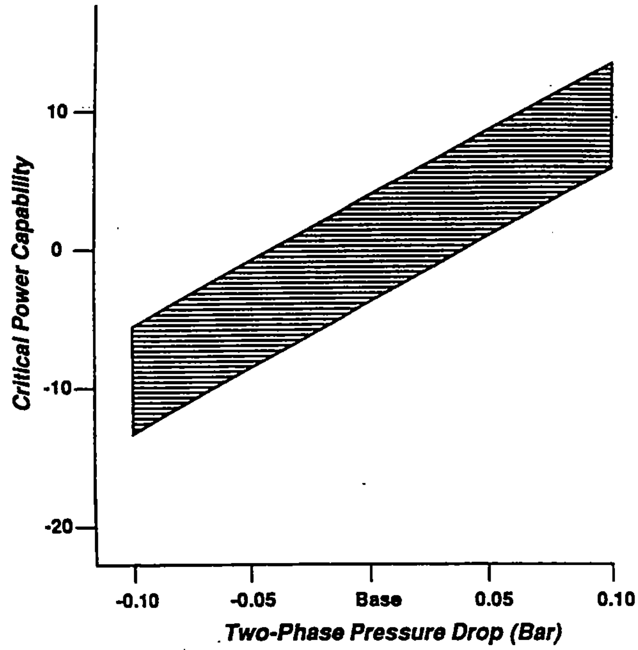

Figure 14. CRITICAL POWER - PRESSURE DROP RELATIONSHIP

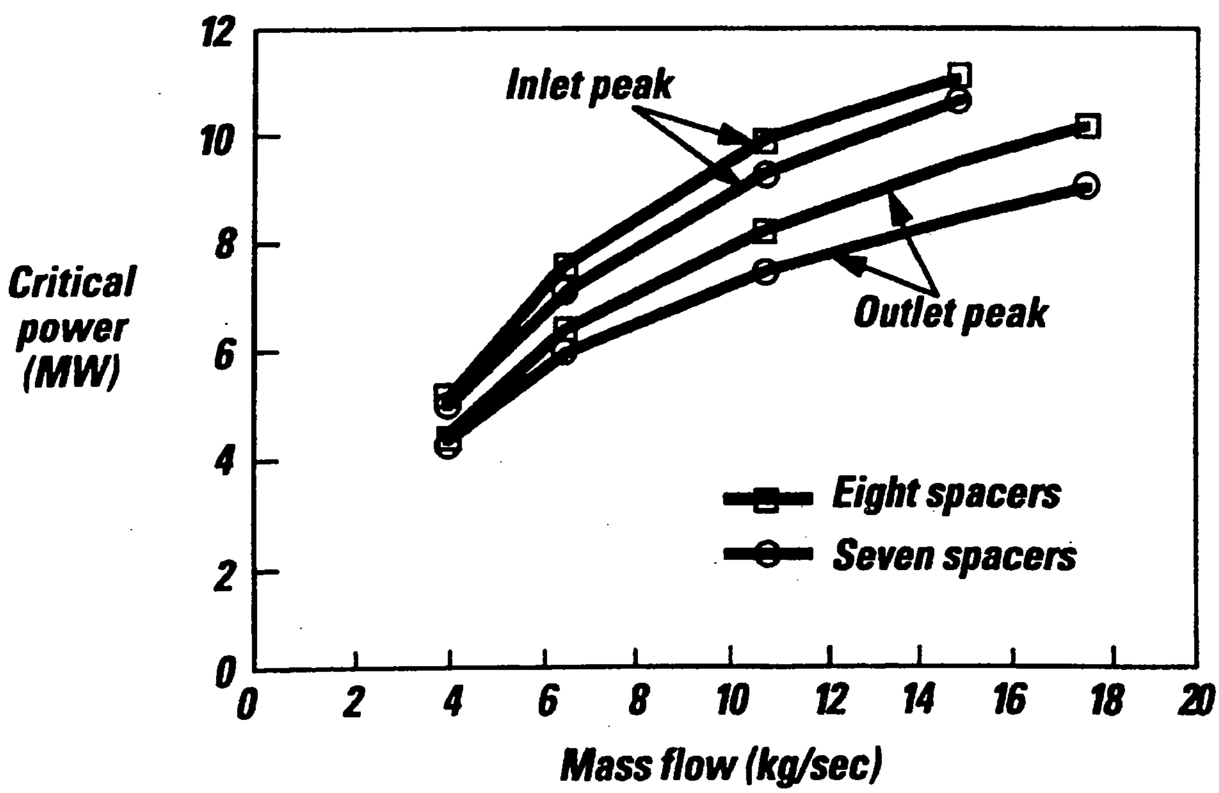

Figure 15. CRITICAL POWER IMPROVEMENT FOR ADDITIONAL SPACER

Impact of Low Leakage Pattern Strategy on Vessel Neutron Fluence

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ENUSA

Introduction

The fast neutrons irradiation of the reactor pressure vessels produces displacements of iron atoms in their metallic structure leading to interstitial relocation of the iron atoms and creation of void positions. The macroscopic expression of these alterations is an increase of the strength resistance of the metal, but also leads to higher temperature for the non-ductile transition, with the risk, if the damage is important enough, of reaching the normal operation temperature of the reactor vessel, permiting a fragile rupture in case a pressurized thermal shock occurs.

The above mentioned problem is the same for the base metal of the vessel as for the weldings of the different pieces that make up the vessel. Nevertheless the content of impurities such as Phosphorus, Nickel and Copper, that have a critical importance in this phenomenum, are normally less characterized in the welding material than in the base metal, due to the welding process itself.

Therefore the reduction in the rate of neutron vessel irradiation is of major importance for guaranteeing a long operating life for the vessel and an increase in the operating margin for pressure transients. In the BWR type, the important thickness of the downcomer water layer reduces in an order of magnitude the level of vessel fluence with respect to the PWR vessels, making the latter the most limiting in terms of vessel embritlement.

The design of the central piece of the PWR vessel has the longitudinal weldings turned 45 degrees relative to the core axes, then being located in the angles of lowest neutron fluence, i.e. the core corners. On the other side, the external flat faces of the core are the closest points to the reactor vessel producing the maximum azimuthal values of the fluence. Those positions are protected with pieces of thermal shield, called "thermal pads". In the PWR vessel there are also two circumferential weldings affected by neutron irradiation and located close to upper and lower core ends, though their relative positions depend on core type (12 or 14-foot cores).

Not only the vessel is important to safety and susceptible to irradiation embritlement. Recent appearances of cracks in the core barrel of BWR reactors have underlined the need to accurately estimate the neutron fluence in inner components and also the opportunity to adopt the necessary techniques to reduce the neutron fluence that impact them.

Low Leakage loading pattern

Since long ago the BWR reload strategies have tried to reduce to a minimum the radial neutron leakages by locating burned assemblies in the core periphery. Also in the last decade the low-leakage loading patterns in PWR reactors have become the normal practice, due to the need for increasing the core reactivity without excessive requirements in feed enrichment. This is due to the general increase of cycle length from the initial 12 months with low capacity factor to the currently usual of 18 months with capacity factors close to 90%.

The loading strategies have become more and more agressive, by filling up the peripheral core locations with twice-burned fuel assemblies in the periphery of the core. While a relative power close to 1 was usual in the periphery, now this value has been reduced to 0.4 or less. Of course this has led to power distribution problems in the inner part of the core, requiring the use of burnable absorbers in the feed fuel assemblies. But, as a secondary beneficial effect, the peripheral fuel assemblies release less and less fast neutrons to the vessel and other internal components.

In Figure 1 the relative power of the peripheral assemblies is presented for the different load strategies of a single PWR reactor core along its life. Strategy 1 corresponds to a clasical out-in loading pattern operating in annual cycles. In strategy 2 a reduction in radial leakages is implemented though cycle length is maintained. Strategy 3 and 4 correspond respectively to the transition and equilibrium status of a low-low leakage pattern with operation in eighteen month cycles.

As it can be seen, there is a drastic reduction of power in all the peripheral positions, but especially in locations (8,1) and (8,2) which are facing the vessel point of maximum fast neutron fluence.

In Figure 2 the fast neutron fluence in vessel is presented along the azimuthal angle for the extreme strategies presented above. It can be seen the global decrease in neutron fluence and it is very obvious the greater reduction in the azimuthal zone, around the core axes, where fluence is more critical.

Axial Blankets

As a natural extension of the radial leakage reduction concept, the use of natural or low enriched uranium in the upper and lower edges of the fuel rod has clear advantages in terms of fuel cycle cost reduction by decreasing the required enrichment for a fixed cycle length. This reduction, which has been estimated in 1% of total fuel cycle cost, also has, as a secondary advantage, an important reduction in the fast neutron fluence impacting the upper and lower core plates.

Also in this case the BWR design was the leader in this move due to the especially high leakages in the upper part of the core due to the reduced coolant density, even increasing the axial blanket thickness to 30 cm. in recent designs.

In the Westinghouse 3-loop (2700 MWt) reactor design, which is the existing in Spain, the fuel assemblies are supported on an intermediate "lower core plate" which in turns stands on the core support plate. But in the XL version of this design, the fuel assemblies are directly supported on the core support plate, locating the edge of the active length closer to this critical component. Consequently the XL fuel designs use taller bottom nozzles in order to reduce as much as possible the fast neutron fluence in the bulk of this component.

In Figure 3 the reduction in fast neutron flux is presented, along the distance from the lower end of the active length for the 17x17 14 foot design. The situation of the support plate is clearly limiting as far as fast neutron fluence is concerned.

Following, a set of examples will be presented with the purpose of illustrating how the fuel design can help to mitigate or to solve specific internal components problems wit hout penalty to the fuel cycle cost or even getting some benefit in terms of burnup limit and cycle energy.

Examples of fluence reduction through fuel design changes

Former Plate Bolts

Recently the bolts that fit the elements of the baffle to the core former plates have shown indications of an accelerated process of stress corrosion cracking assisted by irradiation (IASCC). This is especially important in the baffle corners where the baffle material is in contact with three fuel assemblies, one of them having a relatively highpower rate due to its more inner location.

Until now this problem has not been taken into account in Core Design tasks, being these limited to the reduction of global neutron leakages through adequate care to the peripheral locations of the reactor core.

In fact, the setup of new limitations in core loading pattern search makes still harder the already dificult task a reactive enough, flat enough and stable enough, spatial combination of spent and fresh fuel assemblies.

The possibility of fluence reduction in the critical bolts by locating solid stainless steel rods in the fuel assembly corners has been studied. In order to keep the necessary freedom to locate fresh and burned fuel assemblies in core, it has been assumed that all the fuel assemblies around the baffle corner have incorporated such design modification.

The relative fluence reduction in the most representative points of the baffle comer area are presented in Figure 4. These results correspond to the replacement of four fuel rods by solid stainless steel rods. There is a clear spatial variation of the reduction obtained, which indicates the need for a detailed study in order to evaluate the real benefit to be obtained.

Vessel protection for short thermal shield

The use of axial blankets has been investigated as a possible solution for the Westinghouse 3100p XL reactors where thermal shields are shorter than the fuel active length, leading to some lack of protection of the lower part of the vessel.

In Figure 5 the axial variation of fluxes in the internal surface of the vessel is presented for different axial blanket lengths. There is a clear reduction of fluence in the concerned area (under thermal shield end), but the increase of power density required in the rest of the core, required in order to keep the total power, produces a clear fluence increase in the area inmediately adjacent to the axial blanket, area not protected by the end part of the thermal shield.

Core support plate protection

As indicated above, the use of axial blankets in the fuel design can provide an increased margin in core support plate fluence by reducing the source intensity in the lower end, where most of the neutrons impacting this plate come from.

In Figure 6 the reductions in fast neutron fluence at the plate surface and at one quarter of plate thickness are presented for different length of the axial blanket, from 0 to 30 cm. It can be seen that the maximum fluence reduction obtained is close to 70%.

In contrast it would be interesting to see the opposite impact of reducing the bottom nozzle height from the standard value in AEF XL and XLR designs. Some results are presented in Figure 7, corresponding to a reduction of the bottom nozzle legs of 3 and 12 cm. corresponding the latter to the maximum available reduction, i.e., saving holes plate and skirt part of the present XL bottom nozzle design. As it can be seen the fast neutron flux value is practically three times bigger than the value corresponding to present design. Then it can be concluded that 12 cm. of additional active length can be obtained by creating a 30 cm. natural uranium bottom axial blanket. The benefit in terms of fuel rod burnup corresponding to a 3% increase in active length, and consequently on total energy to be obtained per fuel assembly and total cycle energy cannot be underestimated.

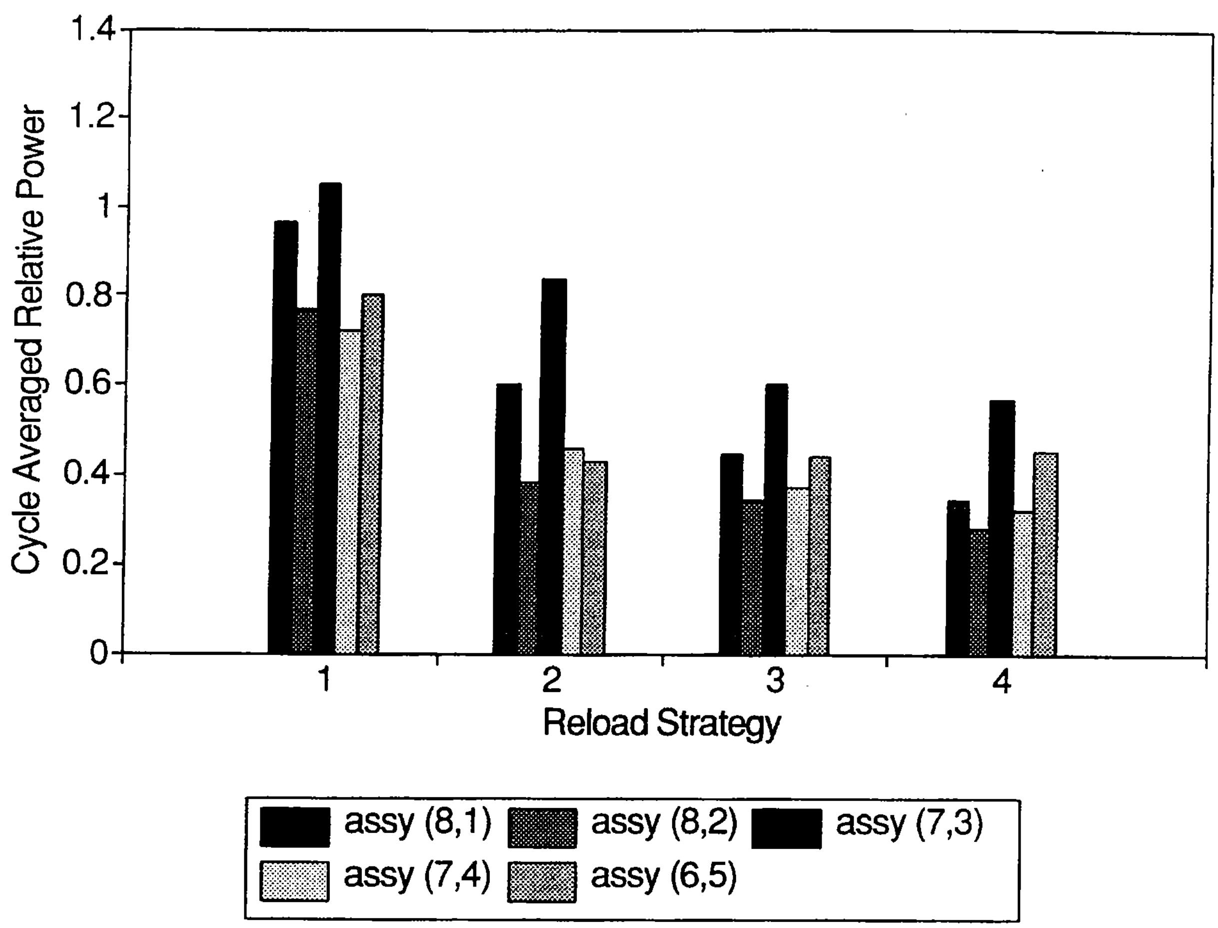

Figure 1. Peripheral Assemblies Power

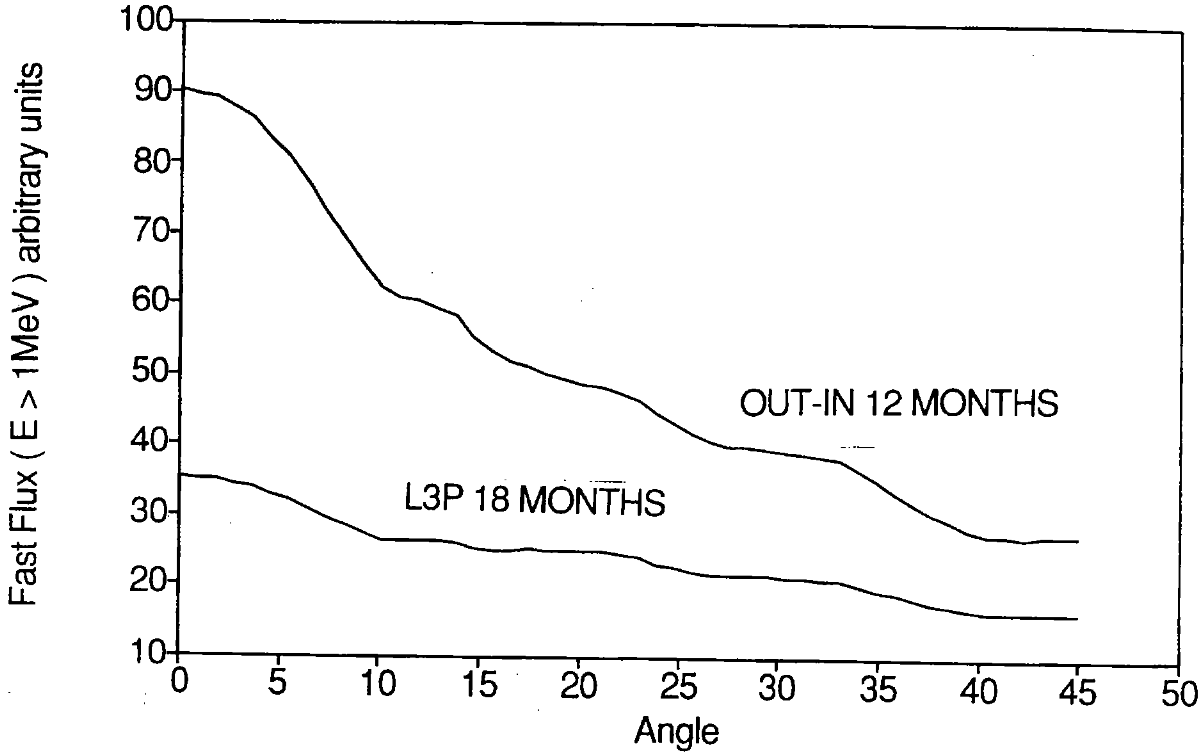

Figure 2. Azimuthal Vessel Fluence

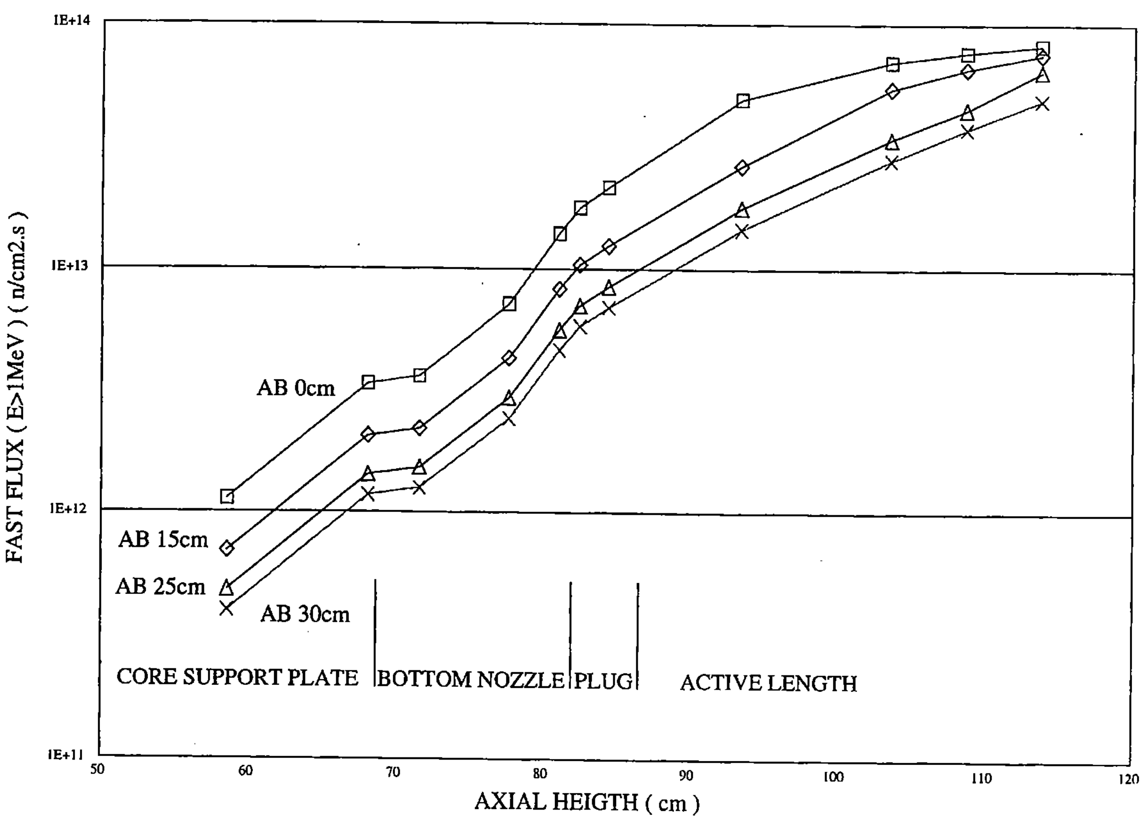

Figure 3. Axial variation of neutron fluence under active length en (W3L-XL)

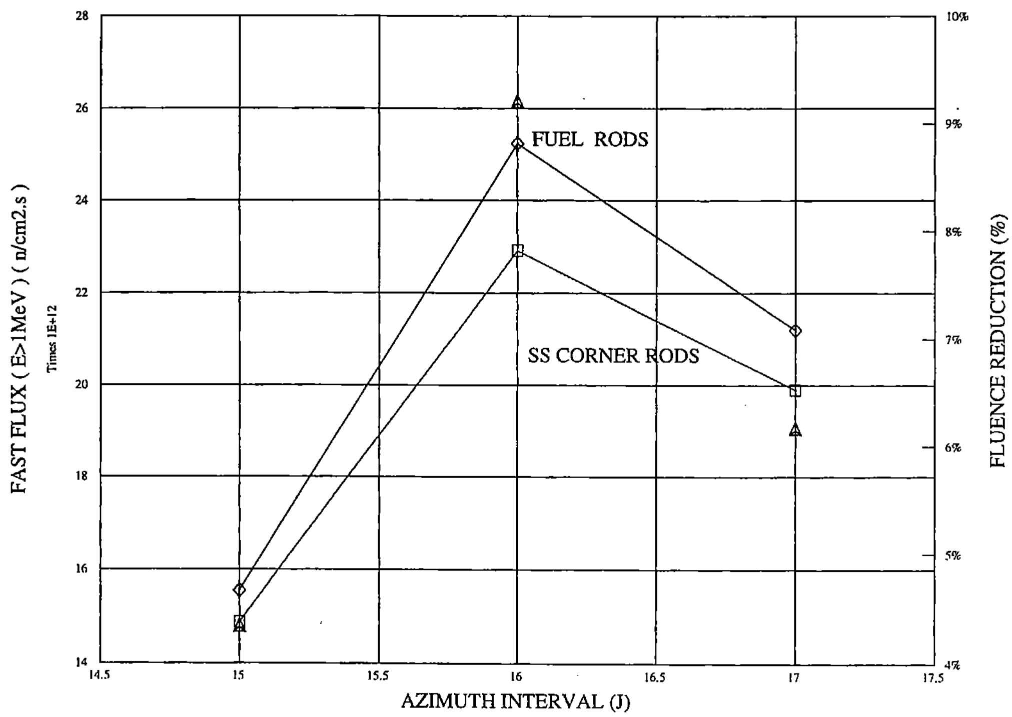

Figure 4. Fluence reduction in former plate bolt with SS corner rods

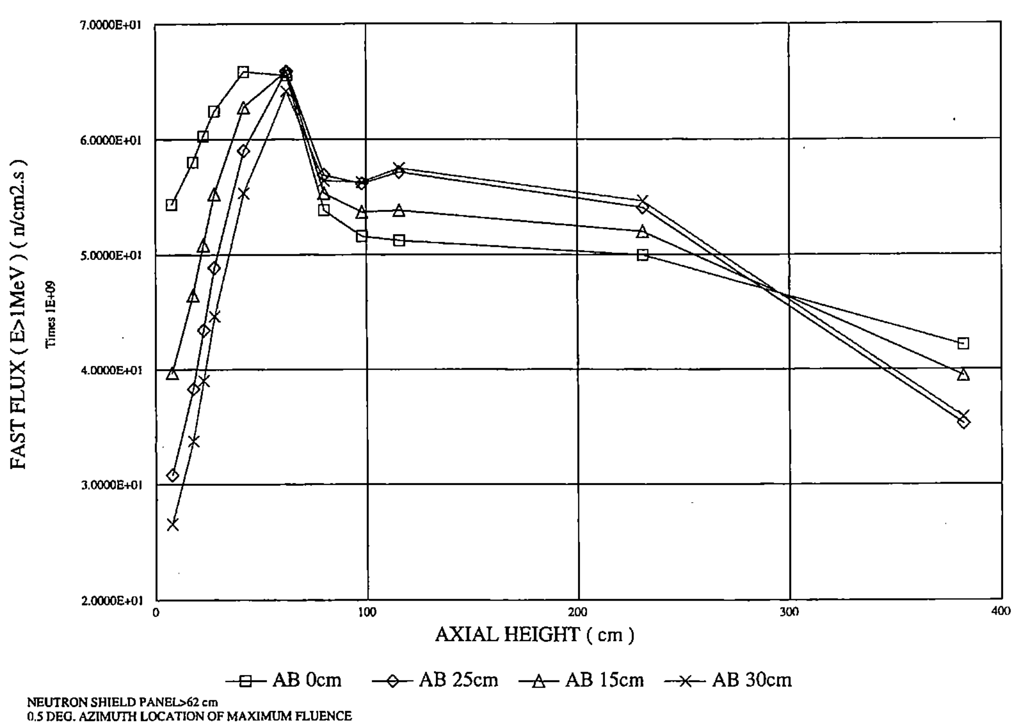

Figure 5. Axial variation of Vessel Neutron Fluence for different axial blanket lenghts (W3L-XL)

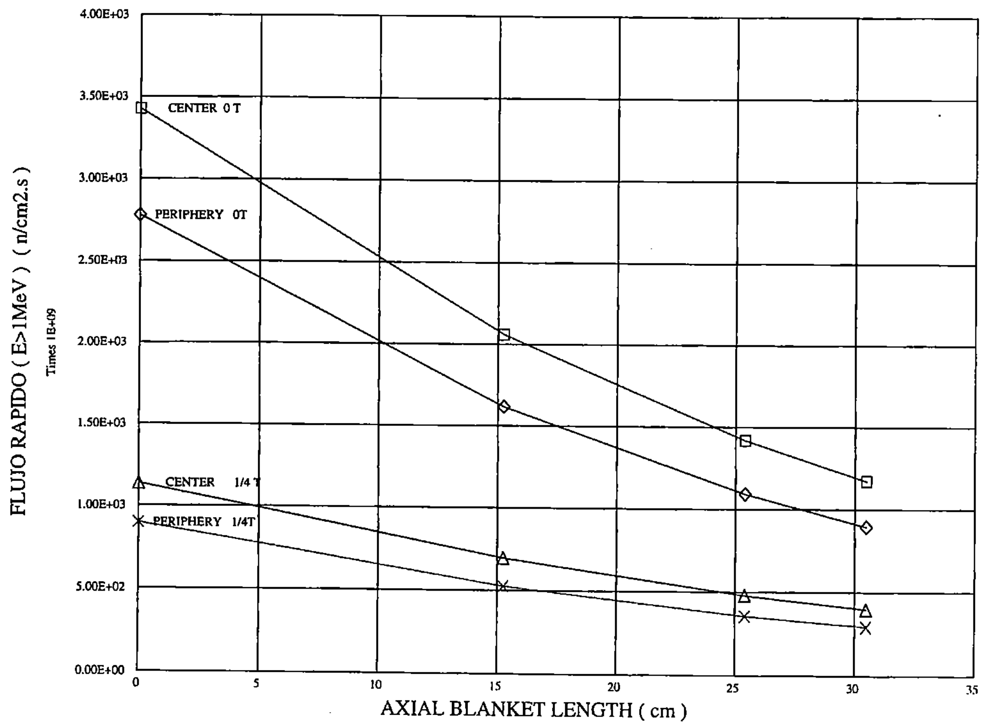

Figure 6. Variation of neutron fluence for different axial blanket lentgth.
At support core plate Surface & 1/4 Thickness (W3L-XL)

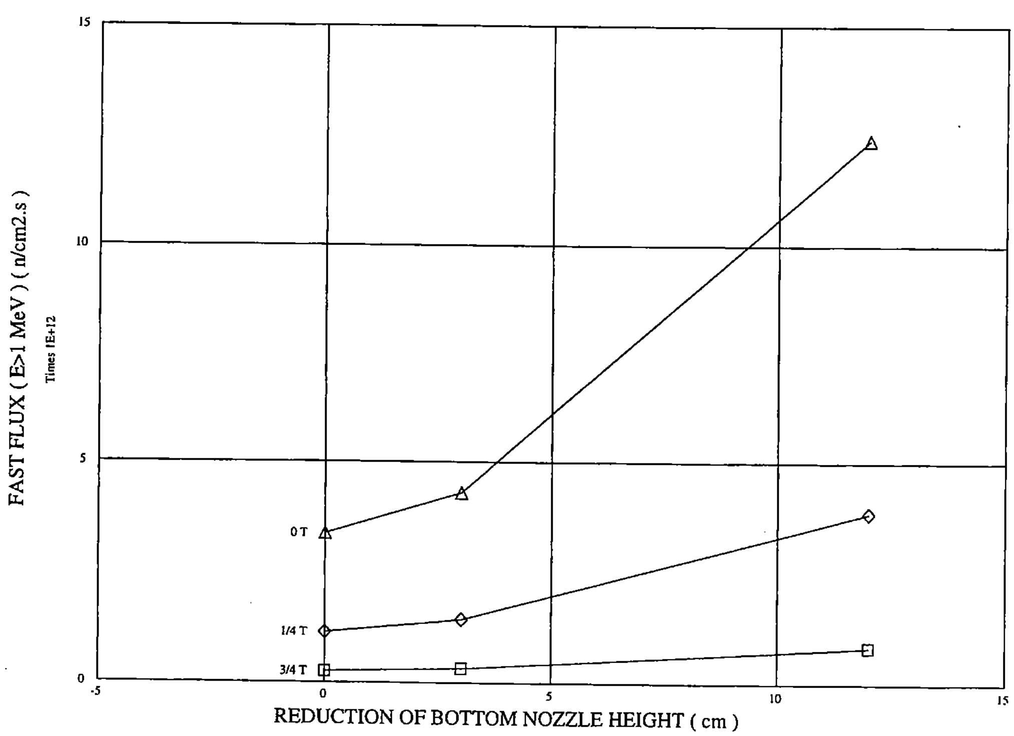

Figure 7. Variation of neutron fluence at support plate.
Surface, 1/4 & 3/4 Thickness

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