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PROCEEDINGS

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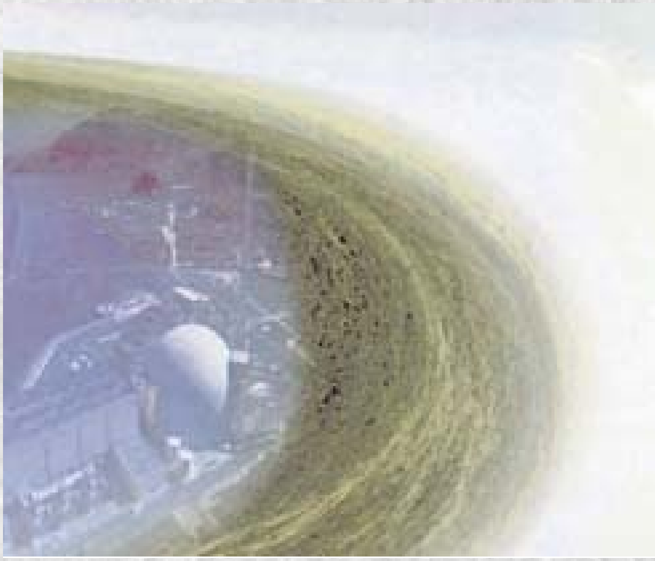
Nuclear power plant life management



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Session 1

OPENING ADDRESSES

NUCLEAR POWER: AN EVOLVING SCENARIO

Budapest, Hungary
International Symposium on Nuclear Power Plant Life Management
4 November 2002

Mohamed ElBaradei
Director General

INTERNATIONAL ATOMIC ENERGY AGENCY

NUCLEAR POWER: AN EVOLVING SCENARIO

In this opening session, I would like to discuss the evolving global scenario for nuclear power and the IAEA's role in meeting key related challenges. I will give specific attention to issues related to life cycle management.

THE CURRENT SCENARIO

Nuclear Power Operation

The rapid expansion in global energy demand — and the growing awareness of the need for sustainable development — has also increased the focus on the environmental consequences of burning fossil fuels. Nuclear power is the principal alternative that can in the foreseeable future provide electricity on a large scale with practically no greenhouse gas emissions.

In 2001, nuclear power supplied 16.2% of the world's electricity, up from 15.9% in 2000. In addition to two new nuclear units in the Russian Federation and Japan, this increase was mainly due to the continuing improvements in plant availability as a result of effective management. For nuclear plants worldwide, average energy availability in 2001 was 83.4%, an improvement equivalent to having 33 more 1000 MW(e) power plants than in 1990.

Nuclear power remains, however, mainly in a holding position. While its environmental merits are increasingly recognized, concerns remain for the public at large and for key decision makers. As I have emphasized for a number of years, the future of nuclear power will depend heavily on its success in meeting a number of challenges: namely, sustaining a strong safety record, improving economic competitiveness, demonstrating waste management solutions and continuing technological development — each of which will contribute to meeting the challenge of regaining public support. I am pleased to note that we have recently seen positive developments in many of these areas.

New Construction

With regard to the construction of new plants, Asia and Eastern Europe remain the centres of expansion. At the beginning of 2002, 17 of the 32 nuclear power plants under construction globally were in four States — China, India, Japan and the Republic of Korea — with Russia and Eastern Europe accounting for an additional 10 units.

Some important developments have also taken place recently in Western Europe and North America. The US Government has committed to work with the nuclear industry to have a new nuclear plant operating in the USA before the end of the decade — a move that would mark the first start of construction of a new US nuclear plant since the late 1970s. Similarly, Finland's decision to build a fifth nuclear power plant is the first such move in Western Europe in 15 years, and contrasts with past decisions in Belgium, Germany and Sweden to phase out nuclear energy. Finally, it is worth noting that the European Commission, in July, indicated that where developing countries have opted or are opting for nuclear energy, and where this is consistent with a national strategy promoting sustainable development, and sufficient safeguards exist, the European Union may provide technical assistance for establishing and implementing the necessary regulatory framework and institutional capacity to manage nuclear energy safely.

Status of New Reactors Under Development

In the light of these developments, a key challenge for the industry will be to prove that available new designs address the often-expressed concerns about nuclear power. Around the world, work is being carried out on advanced light and heavy water cooled reactors, modular high temperature gas cooled reactors, and liquid metal cooled fast reactors. Research is also under way in the Republic of Korea, Russia, the USA and eight countries of the European Union on accelerator driven systems. Each of these new designs aims to produce electricity at an enhanced level of safety, and some seek to serve additional aims, such as producing hydrogen as a clean fuel source, producing potable water at minimal cost, incinerating long lived radioactive waste and reducing plutonium stockpiles.

International Initiatives for Innovative Reactors and Fuel Cycles

Despite the positive results achieved to date on new designs, nuclear energy, like all other technologies, must continue to innovate if it is to play a significant long term role commensurate with its potential. The Agency encourages collaborative innovation between developed and developing countries, including both suppliers and users, to ensure that innovation proceed in a manner that will address future needs. The chief vehicle for these Agency activities is its International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO). Currently, 12 countries and the European Commission support INPRO by their active engagement, financial assistance and expertise. The INPRO International Co-ordinating Group is now working to define user requirements related to economics, waste, safety, non-proliferation and other issues.

Waste Management

The management and disposal of spent fuel and high level radioactive waste continues to be a major point of public concern with respect to nuclear power. Some important progress has taken place in the past twelve months. In the USA the Government has approved the Yucca Mountain site as a repository for high level waste and spent fuel. And Sweden has begun geological investigation of candidate sites for a spent fuel repository. Thus — together with the decision in Finland last year to go forward with a deep disposal facility near the Olkiluoto nuclear power plant — it is likely that by the end of the next decade one or more repositories for the disposal of high level waste and spent fuel will be in operation. But clearly the most important step in gaining public confidence in this area will be the *demonstration* itself — providing concrete evidence that technologically and environmentally sound waste disposal solutions not only exist but are actually working.

Late last year the Agency launched a new initiative to assist Member States in their efforts to move forward with the disposal of high level and long lived radioactive wastes through a "Network of Centres of Excellence" for training and demonstration of disposal technologies in underground research facilities. This network, built initially around facilities made available by the Governments of Canada and Belgium, has now expanded to include underground facilities in Switzerland and the USA. To date, 19 developing Member States have indicated an interest in training scientists through the network.

IAEA Role

The IAEA takes very seriously its role in helping to ensure that nuclear power remains an energy option for those countries that choose to use it. At the most fundamental level, we assist Member States with *capacity building* — through energy system assessments and planning, infrastructure development, and the establishment of legislative and regulatory frameworks. Secondly, we serve as a *global clearinghouse for information* on a wide range of topics related to nuclear power — through training, providing access to extensive technical databases, sponsoring conferences like this one, and fostering collaborative research through projects such as INPRO or the waste research network I just mentioned. Thirdly, an important area of IAEA activities is sometimes described as "creating an *enabling environment* for nuclear development," both in the safety area — where we work to establish legally binding safety conventions, issue international safety standards, share best practices and offer a wide range of peer review safety services — and in safeguards, where our role involves the verification of nuclear non-proliferation commitments. A fourth area involves our transfer of *non-power nuclear applications* — relating to areas such as human health, food safety and crop production, water resources management, and environmental restoration — in ways that help to build nuclear scientific and technical expertise in recipient States, and in so doing can lay the groundwork for the future use of nuclear power. And we are expanding Agency efforts related to *public outreach* — both to the public at large and to key decision makers — to create a more mature awareness of the comparative risks and benefits of different energy sources, the nature and effects of radiation, the considerable range of benefits provided through nuclear applications, and the IAEA's role in ensuring these benefits are used safely and securely.

LIFE CYCLE MANAGEMENT

Licence Extension

A key area of IAEA attention is how best to serve the needs of our Member States in the area of life cycle management — in particular, licence extension and decommissioning. With about one third of worldwide installed nuclear capacity over 20 years old, licence extension is a topic of great relevance to the near-term future of nuclear power. Extending the operating life of existing nuclear plants is economically attractive because it requires relatively little capital expenditure, and reduces the short term need for new generating capacity. On the other hand, these extensions must take place in the context of careful safety analysis and monitoring of equipment ageing concerns.

The importance of a sound framework for licence extension is illustrated by the US experience. When the first licence renewal application was filed in 1998, the prognosis for broad participation in the process was uncertain; but to date, 10 US nuclear power plants have now been granted licence renewals that would increase their licensed lifetimes to 60 years, 37 additional applications have been received, and nearly all US reactor facilities are expected to apply. Last year Russia also began the process of formal licence extension, with the first extension of 5 years already granted and a number of additional plants under consideration. And throughout Eastern Europe, IAEA Member States are expressing their interest in extending plant lifetimes; for example, Hungary, our host for this conference, has attached great importance to exploring safety aspects of continued operation for VVER and RBMK reactors, and in particular to extending the service life of the Paks Nuclear Power Plant.

As this process begins to go forward in more countries, it will be vital that insights are shared on all fronts — including safety and ageing considerations, emerging technological achievements and regulatory policies. It is my hope that over the next few days, in addition to sharing such insights directly, this conference will identify opportunities for more extensive future networking.

Decommissioning

Decommissioning also remains a challenge. While the number of successfully completed projects is steadily increasing — together with confidence in the feasibility of safe decommissioning — some parts of the decommissioning process still require additional effort: improved understanding and control of costs; better defined organizational structures for decommissioning projects — which in some cases must last for decades; and planning for use of the site at the completion of decommissioning. In addition, facilities with an accident history (such as Chernobyl-4) or with particular features (e.g. graphite reactors) still pose major technological challenges.

Decommissioning of nuclear facilities is a central feature of current IAEA assistance in Central and Eastern Europe. The Agency is continuing to provide technical assistance to ongoing decommissioning projects in Bulgaria, Lithuania, Latvia, Slovakia and Ukraine, as well as making available relevant safety standards and technical guidance. We have also established a new decommissioning technical group, made up of experts from Member States, to provide technical and resource insights on decommissioning issues.

In this area as well, it is vital that we learn from experience and share insights, to optimize the use of existing decommissioning resources, to address waste storage and disposal concerns, and to enhance public acceptance of the process. Experience has also shown that by improving up front the design and operation of nuclear facilities, using simple, low cost measures, we can make their eventual decommissioning safer and less costly.

CONCLUSION

We live in an era in which the global community faces many difficult social and economic issues. Against this backdrop, the scenario for nuclear power continues to evolve — in terms of both its current status and its future potential. Life cycle management issues, related to licence extension and decommissioning, will be important areas of focus for the near term. My hope is that this conference will provide important insights on how to move forward in helping our Member States to address these issues effectively.

OPENING SPEECH ABOUT CONFERENCE CHAIRMANSHIP

Jean-Pierre HUTIN
EDF, France

I would like to take a few minutes to talk not about Nuclear Power Plant, not about pipes, vessels and cables but about… human beings. It will help us in remembering that managing ageing of industrial facilities is not only a technical issue but also (above all?) a human adventure.

Initially, it was decided that the Conference would be chaired by our friend Myrddin Davies and that is why his name appeared on preliminary information sheets. But later on, he had to face severe personal problems, in relation with his wife health. So he wisely decided to cancel his participation in the Conference in order to stay with her. At that time, Myrddin and IAEA organizers asked me to take over and become Conference Chairman. I accepted for two reasons : first, it was an honour for me to serve IAEA ; second, it was a pleasure to help my friend Myrddin. That is why my name appears on papers as the Conference Chairman (particularly in the booklet distributed to participants).

Some weeks later, Myrddin problems came to a very sad end with his wife leaving us forever. Then, I thought that, somehow, it would be better for him to come back to business, to focus again his mind on professional activities. So I proposed to withdraw from the Conference Chairmanship and to give it back to him. It was important for me, because it was like a gift I was making to a friend, a gift of life.

Myrddin accepted and we got IAEA organizers informed. Of course, they were quite annoyed because papers were already printed with my name, administrative affairs were settled down and it was complicated to change things again. So, I do apologize, Karen and Slava, for this disturbance.

But really, I am sure I made the right decision by withdrawing from the Conference chairmanship and reassigning it to Myrddin. And I am sure that you will approve it. Somehow, if somebody does not understand my decision, I dare say that he will likely not understand Plant Life Management either! Because it's all a question of human attitude, a question of human beings. Because behind nuclear plant life management, behind pumps and valves and cables ageing, behind data collection and core vessel embrittlement, there is human beings. Not only "beings", but also "human".

And I am proud to participate in a Conference which is chaired by an authentic "human being", one who is unique, as we all are, but one who is also a top level engineer, one who is a recognized expert in plant life management, one who is a true humanist… and one who is a friend, Myrddin Davies.

OPENING SESSION

L. M. DAVIES. Chairman of the IAEA TWG-LMNPP and Symposium Chairman

It is good to be here at this Symposium and it is good to be here in Budapest.

During the next few days we will be looking at and discussing various aspects of Nuclear Power Plant Life Management. But at the outset let me stress that in our consideration Nuclear Power Plant Safety is of paramount importance. Without safe nuclear power plants we have no nuclear power.

Also we will be discussing Nuclear Power Plant Life Management. From the various papers and what has been written there still appears to be confusion in the terminology. NPP life is indeed long – covering the period of conception until we again have a “green-field” site.

What we usually talk about when we discuss this subject is operational life and “operational life extension”. - The operational life is when the plant makes money from the sale of electricity. But it should not be forgotten that we are considering the whole plant cycle from “pre-conception to the grave”.

But even now the Term Plant Operational Life Management seems to have subsumed some of these aspects to do with “normal operation”.

In the operation of NPP the primary consideration – after making the safety requirements is that, as a minimum the plant operates as designed. This is a minimum requirement. There has to be Plant Operational Life Assurance. There are examples where these criteria have not been met – where plant have shut down prematurely – with large financial loss.

Plant life has developed from the objectives of Plant Operational Life Assurance towards the optimization of operational life in a way that reduces the cost of electricity and increases the cost of competitiveness of electricity from NPP.

There is new and increasing understanding for the need for nuclear electricity and this will contribute to the sustenance and improvement in world living standards and in the environment for the future.

The timing of this meeting is opportune and it is heatening to see this strong interest. This, I think, is a reflection of the increased emphasis in protecting the existing capital tied up in NPP and the provision of a “breathing space” for future plant requirements.

I wish you successful and progressive meeting. An important outcome will be the further consideration made by you and the IAEA of issues on this topic.

Thank you.

Session 2
PLANT LIFE MANAGEMENT (PLIM)

Sub-Session 2.1

Chairpersons

S. Azeez
Canada

J.P. Hutin
France

INTEGRATION OF PLANT LIFE MANAGEMENT IN OPERATION AND MAINTENANCE

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Abstract:

Electricité de France is now operating 58 PWR nuclear power plants, which produce 75 % of electricity in France. Besides maintaining safety and availability on a routine basis, it is outmost important to protect the investment. That is the reason why EDF is devoting important resources to implement proactive aging management as an integral part of operation and maintenance programs (for example through appropriate data collection, specific repair and replacement projects and important anticipation efforts, taking in account the high level of standardization of the units).

A particular organisation has been set up to continuously observe and analyse all activities so as to make sure that aging concern is correctly taken in account in strategies and that no decisions are susceptible to impair plant lifetime. This "lifetime program" is paying attention to technical issues associated with main components but is also dealing with issues related to economics, staff and industry situation.

1. INTRODUCTION

Electricité de France is now operating 58 PWR nuclear power plants, which produce 75 % of electricity in France. Besides maintaining safety and availability on a routine basis, it is outmost important to protect the investment. Indeed, such an asset is a tremendous advantage just as the company is going to face the new European electricity market. That is the reason why EDF is devoting important efforts to implement effective aging management in plants which are still quite "young" (the eldest one is "only" 25!).

Of course, it must be recognized that NPP life time is not threatened only by component-related problems: other non-technical issues must be seriously considered like industrial support, information system, staff and skilled people, public acceptance, etc. It must also be noted that managing ageing and remaining lifetime of an industrial facility is a concern, which must be taken in account as soon as possible in daily activities. As a consequence, EDF pays a great attention to implement proactive ageing management as an integral part of operation and maintenance and examples of such an attitude will be given.

2. LIFETIME LIMITS AND MANAGEMENT POLICY

There are many possible definitions for the "lifetime limits".

For some components, lifetime is given by the manufacturer as a guaranteed value. But in most cases, manufacturers give only indications about what they consider as an "average technical lifetime", without any contractual commitment.

The pressure boundaries of the nuclear steam supply system represent a special case. Numbers of expected operating transients are input in the fatigue analysis required by pressurized component design codes. So that the time to undergo these transients may be considered as a "regulatory lifetime" of the NSSS (although the rate of transient consumption may be changed by adapting operating practices - see paragraph 3).

An other lifetime value is the one postulated to calculate amortization of the investment.

And of course, in many countries, licensing period may be considered as a kind of "lifetime" although it is generally not related to any technical limitations. In France, such a limited licensing period does not exist. The idea is that safety is permanently under scrutiny with a complete review and reassessment every ten years but without formal authorization process. However, experience shows that this ten-year safety review which is put under control of safety authorities may represent a "re-investment" as big as a formal licence renewal process.

More generally, the EDF life management policy for the nuclear power plants is based on four steps :

- looking for excellence in daily operation and maintenance activities, with a permanent improvement loop for safety and cost-effectiveness, using an effective experience feedback organisation and taking advantage of the high level of standardization of the units.

- "exceptional maintenance" programs to implement large repair and replacement activities with sufficient anticipation so as to eliminate the risk to be obliged to perform the job on all the units at the same time.

- every ten years, a complete review of each facility with a new set of references and an upgrading of its safety level through appropriate modifications while maintaining unit standardization in all the fleet,

- a Life Management Program, at corporate level, which permanently scrutinizes operation and maintenance activities to identify decisions which could impair plant lifetime and which surveys research and development programs related to ageing phenomenon understanding.

For the "less technical issues", the Life Management Program is particularly expected to detect those which appear not to be correctly handled and to alert managers so they can make appropriate decisions (implement a permanent process or develop a specific project).

3. PROACTIVE INTEGRATION OF LIFETIME CONCERN IN O&M ACTIVITIES

As it has been said, it is outmost important to proactively take in account the lifetime concern in daily operation and maintenance activities. And this must be done as early as possible in plant life.

It generally requires some engineering analysis because this was not fully taken in account in original operation and maintenance procedures (see following examples about fatigue monitoring or core vessel fluence). But many good ideas may also arise from plant staff. Moreover, people will apply optimised procedures more certainly if they understand the intent. So that increasing lifetime awareness of plant people is also an important feature of EDF strategy.

In many cases, evidences of phenomenon which impact component lifetime may be inferred from operating experience if properly analysed. So it is very important to have an effective experience feedback organisation. In EDF, events and data are first analysed by site engineers. Then, the most important information are provided to the corporate level where specialized departments perform a "second level" analysis, particularly to detect any generic aspect. As a consequence, they eventually propose relevant measures to be taken on other units and/or initiate appropriate updating of operating and maintenance procedures.

The following paragraphs give examples of how the lifetime concern is proactively integrated in plant operation and maintenance: fatigue life monitoring of primary components, core vessel fluence reduction, long term repair/replacement strategies, collecting relevant data, surveying industrial capacities.

3.1. Fatigue Life Monitoring of Primary Components

When assessing residual fatigue lifetime of components, past data about operation are needed.

A good example is the fatigue assessment of the NSSS. During design, a list of pressure and temperature transients was taken as hypothesis to estimate the risk of fatigue in each NSSS components: heat-up and cool-down, load variations, auxiliary system actuations, etc. But, in order to make a prediction about the remaining fatigue resistance of a given component, information about actual fatigue transients is needed. That is what EDF operators are looking for by bookkeeping all pressure and/or temperature transients from the very beginning. When a transient is detected (with very low thresholds), systematic comparison is made with the reference design transients to make sure that the actual fatigue loading is not more severe than expected but also, is not too conservative either.

This bookkeeping led to adapt several times the original design list of transients (in number and nature) in order to obtain a more "operational" reference list, which fits, closely to the actual transients (of course, it requires to update design analysis).

But the most interesting benefits is that it allows to identify operating procedures or practices which are detrimental to the fatigue resistance of some pieces of equipment. For example, it was found that pressurizer and steam generator operating procedures were imposing significant fatigue loading on some nozzles and branch lines. Then, it was possible to reduce the fatigue cumulative damage by adapting procedures.

After twenty or thirty years of operation, this transient monitoring process makes possible to perform NSSS fatigue assessments which are safe but not overly conservative.

Moreover, it was recognized that, at some specific locations, global temperature and pressure variations does not give an exact view of the fatigue loading due to local thermo hydraulic phenomenon like stratification. Extensive studies were performed on that "fatiguemeters" which continuously monitor local fatigue loading and permanently estimate usage factors.

3.2. Core Vessel Fluence reduction

The situation of EDF core vessels with respect to irradiation embrittlement is pretty good, thanks to an early awareness of the detrimental effect of some species in steel content.

However, EDF decided to reduce core vessel irradiation in order to increase the margins in the vessel failure assessment, which is of outmost importance for the plant life expectation. This was made possible by optimising fuel assembly management in such a way that the fluence is significantly reduced. The result is a reduction of up to 10 °C of the end-of-life reference transition temperature (RTndt), which is one of the main parameters of the vessel failure risk assessment.

Moreover, it was decided to extend the embrittlement surveillance program. The original program was based on steel samples installed in the vessel itself and which are regularly extracted and tested. New steel specimens are now put in capsules when locations become available. These new samples are made of original steel pieces which had been happily put aside in due time. Such an extension of the embrittlement surveillance program will be very helpful in assessing the effect of irradiation on core vessel steel during the next twenty or thirty years of operation as compared to what happened in the first twenty years, taking in account the fact that fuel characteristics and management may change several times in plant life.

3.3. Repair and replacement strategies

Because all the units are quite identical, large resources may be devoted to problem solving so that a solution found for one unit may be implemented on all the other units. But this high level of standardization could become a weak point if problems are not sufficiently anticipated. It is absolutely necessary to eliminate the risk of a generic problem, which would affect all units at once and oblige to shut them down at the same time. For that, EDF must have a very prospective view of all major degradations, which could impair component reliability or integrity with appropriate repair/replacement strategies. These strategies must cover periods of time consistent with the capacity of nuclear industry to develop methods and build components for all the fleet (between five and fifteen years).

The "Exceptional Maintenance Program" is dedicated to periodic review of design, fabrication and experience feedback of the 30 or 40 most sensitive components in order to identify possible future problems, to estimate potential consequences and to propose appropriate measures to be taken: simple surveillance, development of repair methods, early procurement of spare parts and, possibly, replacement in anticipation if the risk of having all the plants affected at the same time appears too high.

Decisions are made after performing a cost / benefit analysis of various strategies. Probabilistic approaches are sometimes used. But it must be recognized that indirect consequences of some catastrophic scenarios are sometimes difficult to assess.

Of course, consequences of the "anticipation / no anticipation" choice must be integrated on the whole plant lifetime.

Finally, the Exceptional Maintenance Program allows to identify large maintenance works which will most likely be performed and to take measures such that, at the proper time, impact on EDF fleet performances is minimized.

3.4. Collecting relevant data

When assessing plant lifetime expectancy, past data about operation and maintenance are needed.

Example of fatigue data bookkeeping was already described for NSSS components. For other plant components and other types of damage, monitoring programs (beyond regular in service inspection) depend on the ageing probability and consequences. For example, a very sophisticated and extensive approach has been undertaken for cast austeno-ferritic steel elbows: samples were put in laboratory facility where they are submitted to actual or slightly accelerated operating conditions; and actual micro-specimens were taken out from operating plant elbows for analysis and testing.

More generally, the operating experience feedback organisation, which asks for plant staff to report all significant events, is used to feed a history data bank. These data are mainly kept in three formats:

- event description and its causes and potential consequences as they are identified by plant engineering staff, are kept on file. For most interesting events, the file is completed by corporate analysis conclusions.

- yearly synthesis are made by corporate departments on selected subjects.

- system and component reliability data are separately recorded in order to be easily used as input data in probabilistic type of analysis like Safety Probabilistic Assessment or Reliability Centered Maintenance Approach.

It must be emphasized that the process of collecting appropriate data always begins with plant personnel and that, in order to correctly perform the job, they need to understand the purpose of it and the importance of doing it well. This means a "plant life management awareness" which has to be permanently maintained.

When saying that collecting data must be done as early as possible, it can be added that "early" is never "early enough”. Indeed, it was often experienced that problems arising during plant operating life were related to events which had occurred during pre-operational tests and for which most of the information were lost.

3.5. Surveying Industrial capacities

To build 58 units in twenty years, France had to create specific industrial capacities and to adapt existing facilities. At the same time, many countries were doing the same. Now, the future of NPP construction is not very clear. Even in France, there is no real need for new plants before 2010 or 2015: this is clearly a drawback of the good life management of existing plants!

Some design and construction companies were able to turn to maintenance activities. But even maintaining the equipment they built does not give as much work as construction. And there is a strong competition with companies the core business of which has always been maintenance. And last but not the least, there is companies, which designed and built so reliable components that there is almost no maintenance to perform!

For all these reasons, there is a high risk for some of these "nuclear" companies to go out of business. And the question is: if some components need large repair, revamping or replacement, will the utility find appropriate industrial support to do the job and provide the spare parts.

To cope with this issue, EDF set up a "world industrial capacity observatory" which permanently surveys international situation and try to identify "product / company" pairs which are "critical" and "sensitive”:

- "critical" because the company has no sufficient activity perspective so that it is likely to disappear or to stop providing the required product or service (this is particularly true with companies which have been totally dedicated to specific nuclear component design and manufacturing)

- "sensitive" because their disappearance would put EDF in a difficult situation

When a "critical and sensitive" case is identified, EDF brings the matter up for discussion with the concerned companies in order to look for possible solutions.

4. TEN YEAR REVIEW AND PLANT LIFE MANAGEMENT PROGRAM

As it has been explained, the best way to correctly manage plant lifetime is, first, to seek quality in everyday operation and maintenance. However, because of the importance of the issue, EDF decided to give itself complementary guarantees: the Ten-Year Reviews and the Plant Life Management Program.

- *The Ten Year Reviews*

Before starting the ten-year outages of a series, a new set of safety references is defined and discussed with Safety Authorities. component/system/structure situations and operating conditions are reviewed, taking in account experience feedback. The results of these reviews are compared with new references and a modification and upgrading program is defined.

During the ten-year outage of each plant of the series, these modifications are implemented. Extensive inspections, verifications and large maintenance works are performed, including an important set of anticipation measures (see paragraph 3.3). A complementary investigation program is also implemented, specifically to look for unexpected ageing phenomenon.

- *The plant Life Management Program*

Beyond that, a particular organisation has been set up at corporate level to continuously observe and analyse all activities so as to make sure that ageing concern is correctly taken in account, that everything is done to reach expected lifetime goals and that nothing is done which could impair plant lifetime. This Plant Life Management Program is paying attention to technical issues associated with main components but is also dealing with issues related to economics, staff skills, regulation evolutions, industry situation, etc.

The Plant Life Management Program also supports research and development works related to plant life and ageing concern (for example, development of methods and tools to correlate operating conditions and lifetime expectancy or research on degradation mechanisms and kinetics). It has close relationships with other companies and international organisations sharing the same concern (particularly IAEA).

The Plant Life Management Program periodically reviews EDF and world situation and reports its conclusions to EDF General Manager, then to Safety Authorities.

5. AGEING SITUATION OF EDF NPP COMPONENTS

Component ageing is certainly not the only factor impacting plant lifetime... but it is however an important one ! With that respect, EDF has a rather satisfactory situation, partly due to an early awareness.

Core vessel embrittlement is not really an issue: due to low level of residual elements, the end-of-life Rtndt should not exceed 100 ° C. Moreover, optimised fuel arrangements are now implemented to reduce even further the level of vessel irradiation. And the original embrittlement monitoring program is to be extended by putting new specimens in place when old ones are removed for testing.

As far as fatigue of reactor coolant system is concerned, actual usage factors are less than expected in most locations and this can be demonstrated with the results of the transient monitoring and bookkeeping program. However, it must be recognized that some locations require specific attention due to unexpected thermo hydraulic phenomenon. Fatiguemeters are now under development to address this issue and, in any case, repair is possible.

The 900 MW plant containment buildings are not a concern. But 1300 MW plant ones will certainly require some repair because of insufficient tightness of their inner wall.

Cables do not raise severe questions but need some monitoring.

The instrumentation and control life expectancy has been very carefully analysed through a specific project, looking both at technical and industrial aspects. For 900 MW plants, almost nothing is to be done before the third ten-year outages, except for some revamping of a few systems and for in-advance spare parts procurement (we are just performing the second ten-year outages on oldest plants). For 1300 MW plants, the same assessment is under progress.

Pressurizer is not an issue but replacement feasibility has been studied. For other components, repairs and/or replacements are always possible and cost-effective, as long as appropriate anticipation precludes situation where all plants would have to be treated at the same time.

So that, if other than technical aspects are correctly handled and if current operation remains safe and cost-effective, EDF nuclear power plants should easily last more than forty years.

6. CONCLUSIONS

As it has been shown, plant life management is a very important issue for any utility, and particularly for EDF. And to correctly handle this issue, appropriate measures have to be taken immediately, without waiting for the last years. Many of these measures are of technical and/or industrial nature but the importance of personnel and manager awareness should not be neglected.

Ultimately, it must be recognized that many industrial facilities are obliged to shut down for reasons which are not directly related to equipment situation or utility activity but which are the consequences of external changes: country energy policy, regulation evolution, public acceptance, etc.

Of course, utility must keep a constant watch over these external changes, which may impact NPP's survival, with enough anticipation. But the fact that, in such evolving context, a utility defends its own interest will be well accepted only if it has a high level of credibility. And this credibility may only arise from excellence in safe and cost-effective operation: in any way, such an excellence is the best guarantee for long, long, very long nuclear power plant life!

APPLICATION OF RISK BASED INSPECTION AS A PART OF LIFE MANAGEMENT OF NUCLEAR POWER PLANTS

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Abstract:

In the ageing and life management of nuclear power plants, several aspects like safety, availability, costs, radiation exposure etc,. has to be considered for planning plant activities. Reliability methods are suitable for integrating such aspects at the plant or system level. Risk Informed In-Service Inspection aims at prioritising the components for inspection within the permissible risk level thereby avoiding unnecessary inspections. Various methods have been evolved for prioritization of components for In-Service Inspection incorporating risk information. This paper presents two approaches that can be employed to prioritise components with an appropriate case study.

1. RISK INFORMED IN-SERVICE INSPECTION AS A PART OF LIFE MANAGEMENT

In-Service Inspection (ISI) of Nuclear Power Plants is essential for ensuring reliable performance, structural integrity and containment, i.e., leak tightness of all critical components, through non-destructive evaluation (NDE) of defects, stresses, corrosion, dimensional changes and micro-structural degradation in components during their service life, due to exposure to radiation, high temperature, pressure, loads and hostile media. Current ISI programs are based on past experience and engineering judgement through deterministic analysis. Service experience has indicated that failures are dominated by corrosion or fatigue mechanisms. The probable areas of failure are determined by deterministic analysis and are included in ISI programs. Studies are underway for development of alternate methodologies for suggesting an ISI program in compliance with plant safety level.

Research is being conducted in order to establish the suitability in applying the Risk informed technology for suggesting an in-service inspection plan. The process of using Probabilistic Safety Assessment information and insights to support the in-service inspection is termed as Risk Informed In-Service Inspection (RI-ISI). The goal of Risk Informed In-Service Inspection is to advance the development of risk technologies and implement these technologies to establish effective structural integrity management programs, reduce plant down time, industry and regulatory burdens, and continue to maintain plant safety.

In conducting ISIs it is important to have an inspection plan that is optimised to provide effective inspections at the right location with a proper inspection frequency. Probabilistic Safety Assessment (PSA) methodology provides a technical basis for inspection plans and also to ensure that plant operates in the safe domain within the prescribed risk levels. Using this Risk informed approach [1,2], it has been demonstrated that this method can identify and prioritise the most risk important systems for inspection. Various methodologies for ranking the components using Risk Informed Approach viz. Importance measures, matrix definition etc. have been evolved [3,4].

The various steps in carrying out Risk Informed In-Service Inspection can be summarised as below:

- Identification of systems and boundaries using information from a plant PSA
- Ranking of components applying the risk measures to determine the categories that are then reviewed to add deterministic insights in making final selection of where to focus ISI resources.
- Determination of effective ISI programs that define when and how to appropriately inspect or test the two categories of high safety significant and low-safety significant components
- Performing the ISI to verify structural integrity of component and then updating the risk ranking based on inspection and test results

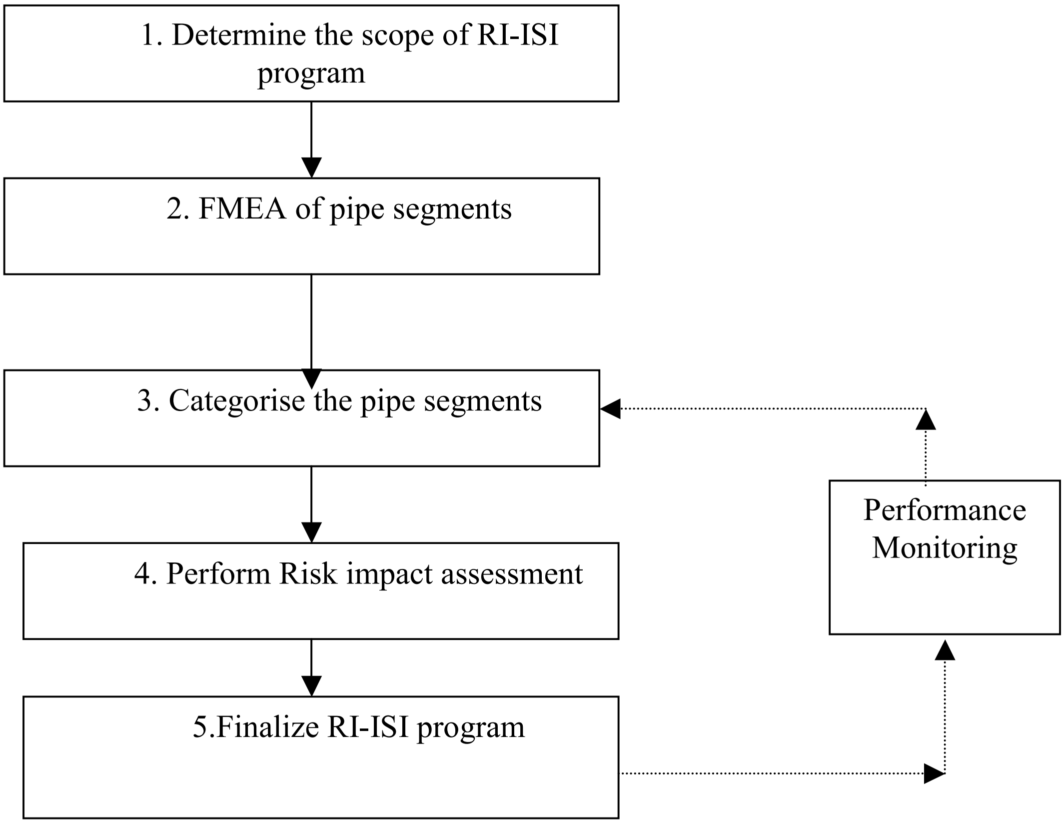

Figure 1. Flow chart on RI –ISI program.

2. COMPONENT IDENTIFICATION FOR PRIORITISATION

2.1. Using Importance Measures

There are two principal factors that determine the importance of the component in a system: (i) the function of the system, i.e., frequency of the challenge and availability of the backup trains and (ii) the reliability or unreliability of the components. Because nuclear power plants are designed according to the defense-in-depth, one single failure of a component or other basic event will probably not result in a large accident. More likely, a large accident will be the result of failure of multiple basic events. The PSA methodology determines important combinations or in other words, cutsets that could result in a large accident.

A Risk importance measure gives an indication of the contribution of a certain component to the total risk. Various methods are available for measuring the importance of components. A new parameter called inspection importance measure has been developed in order to prioritise the systems for ISI. System level ranking based on Inspection Importance Measure (I^w). Inspection Importance (I^W) of a component is defined as the product of the Birnbaum Importance (I^B) times the failure probability. Birbaum Importance of a component is defined as

$$I^B_{x_i} = CDF_{x_i=1} - CDF_{x_i=0} \qquad (1)$$

Birnbaum Importance of the system is the sum of the Birnbaum Importance of the components in the system.

$$I^B_{sys} = \sum_{x_i} I^B_{x_i}$$

x_i = i^{th} component in system

$$I^W_{sys} = I^B_{sys} * P_{f_{sys}} \qquad (2)$$

$P_{f_{sys}}$ = System failure probability due to structural integrity failures

3.2 Using Risk Matrix

Risk matrix: Decision matrix that is used to categorise the pipe segments into high, medium and low importance, based on degradation mechanism and consequence of its failure. By examining the service data, a basis has been established for ranking pipe segment rupture potential as High, Medium or Low simply by understanding the type of degradation mechanism present. Consequence can be quantified through the estimation of Conditional Core Damage Probability (CCDP).

$$CCDP_i = [CDF_{(\lambda i = 1)} - CDF_{(BASE)}] * T_E \qquad (3)$$

Where,

$CDF_{(\lambda i = 1)}$ = CDF given the component failure

$CDF_{(BASE)}$ = BASE CDF

λ_i = Pipe break frequency

T_E = Exposure Time (Detection time +AOT)

Measure of Risk due to pipe break: $\mathbf{CDF_i = \lambda_i * CCDP_I}$

CONSEQUENCE CATEGORY

	N one < 10^{-8}	**Low** 10^{-8} <CCDP <10^{-6}	**Medium** 10^{-6} <CCDP <10^{-4}	**High** >10^{-4}
High (>$10^{-4)}$	L ow 7	Medium 5	High -3 CDF = 10^{-10} – 10^{-4}	High -1 CDF >10^{-8}
Medium (10^{-7}<F<10^{-4})	L ow 7	Low 7	Medium 5	High - 2 CDF = 10^{-11} –10^{-4}
Low (<10^{-7}) No deg-mech	L ow 7	Low 7	Low 7	Medium 5

Figure 2 . Risk Matrix.

4. ESIIMATION OF PIPING FAILURE PARAMETER

Estimation of piping failure probability is a crucial task since this form the basis for further analyses. Three methods have been suggested for piping failure probability estimation: (i) Structural Reliability Analysis (ii) Service Data Analysis and (iii) Expert Opinion. The degree to which one relies on one method or another is predicted on the availability of data from service experience, experts or structural reliability or risk models.

4.1. Service Data Analysis

Databases are an important source of information that can support the estimation. Various Studies has been conducted by WASH 1400 and IAEA, through which they are able to bring out some estimates. Statistical estimates of pipe failure frequencies are derived as a key factors associated with pipe failure mechanisms (degradation mechanisms and loading conditions).

$$\lambda(F) = \quad \lambda_j(f) \qquad (4)$$

$\lambda(F)$ = failure frequency piping segment I

$\lambda_j(F)$ = failure frequency piping segment i due to failure mechanism j

$$= \frac{n_j \quad (F)}{T}$$, T reactor years

$n_j(F)$ = Number of pipe failure due to failure mechanism j

For finding the rupture frequency

$$\boldsymbol{\lambda_j(R) = \lambda_j(F)P_j \{R|F\}}$$

$\lambda_j(R)$ = Pipe rupture frequency due to failure mechanism j

P_j {R|F}= the conditional probability that a pipe failure due to failure mechanism j will be a rupture

Using Baye's theorem, the prior values can be updated with plant experience, to obtain the uncertainty distributions. Another advantage of these databases are in identification of degradation mechanism prevalent in the structure. This forms a realistic basis for establishing the proper inspection programme.

4.2 Structural Reliability Analysis (SRA)

SRA employs the use of probabilistic fracture mechanics techniques to calculate the failure probability as a function of time, including the effects of inspection frequency, probability of detection (POD) & degradation mechanism. Through Monte Carlo sampling, the results of tracking a very large number of crack simulations can be used to determine what fraction of cracks will not be detected and repaired before failure results. This methodology provides models for determining the crack growth for different degradation mechanisms also. Computer codes like PC-PRAISE etc. are available for carrying out such analysis.

These models are Computationally intensive. The results of these analyses are often driven by uncertainities in defining crack size distribution, stress history, detection probability and reference flaw size. Moreover, these estimates are too small & yet to be reconciled against service experience.

4.3. Ageing Mechanism Identification & Life time prediction

Identification of ageing mechanism is a crucial task since it forms an input in determining the type of inspection programme to be adapted for the structure. Plant operating experience and expert elicitation can provide vital insights into the degradation mechanisms prevalent in a structure. Past Inspection results can predict the progression of various degradation mechanism in a structure, which gives an indication of its remaining life.

Table I shows the classification of degradation mechanism made on the basis of data collected from plant experience, which is currently employed for Risk Matrix approach in prioritizing components for In-Service Inspection.

Table I. Classification of Degradation Mechanism

Potential	Degradation Mechanism
High	Flow Accelerated Corrosion, Vibration Fatigue, Water Hammer
Medium	Thermal Fatigue, Corrosion Fatigue, Stress Corrosion Cracking, Pitting, Erosion Corrosion
Low	No degradation mechanism

Various analytical and semi empirical models have evolved in predicting the degradation rate from typical degradation mechanism. Structural Reliability methods employ various models to estimate the remaining life as a part of life management programme. The properties of the materials, the ambient water chemistry and thermal-hydraulic conditions and mechanical load on the component must be evaluated in order to assess the type of degradation as well as the rate of degradation progression. The limit on the service life is reached when

1. the maximum allowable stress in the pressure retaining boundary is reached
2. the maximum allowable utilization factor is reached with respect to material fatigue
3. the toughness of the material drops below the required values

5. CASE STUDY ON COMPONENT PRIORITISATION

To discuss the importance of ranking of piping segments from process and safety systems, we consider the simple case of an initiating event and safety system. The initiating event is assumed to be Medium Loss Of Cooling Accident (MLOCA). This initiating event can happen if there is any failure in piping resulting in loss of inventory. The piping failure frequency includes the contribution failure frequency from main Primary Heat Transport (PHT) line and feeders. When this initiating event happens, the safety system, Emergency Core Cooling System (ECCS) will act as the mitigating system. Risk from this event can be defined as the frequency of occurrence of this initiating event coupled with the unavailability of the Emergency Core Cooling System (ECCS).

The Risk Metric R is computed as the Core Damage resulting from Medium LOCA incident coupled with loss of availability of ECCS. Accident sequences from other initiating events are not considered in this case study.

$$CDF_{MLOCA} = f_{MLOCA}\ q_{ECCS} \tag{5}$$

Where f_{MLOCA} is the Medium Loss Of Cooling Accident frequency and q_{ECCS} is the ECCS unavailability

5.1. Initiating Event Frequency of Medium LOCA

The Primary Heat Transport system consists of two inlet headers and two outlet headers. The outlet header is connected to Steam Generator, which in turn to Primary Coolant Pump. The Primary Coolant Pump is connected to inlet header. Table II presents the various piping segments considered for process system Primary Heat Transport system.

Assumptions

(i) System boundary is considered at feeders and channels are not included for ranking
(ii) Medium LOCA can occur if there is pipe failure in any of these components.
(iii) Failure frequency of basic event includes failure frequencies of welds, elbows, joints etc. From the point of view of ISI, basic events should be analysed in weld level. Since the paper aims at highlighting the application of importance measures, basic events are considered in a broad sense.
(iv) When the leak rate of coolant from the system is higher than 50gpm, the piping is termed as failed.
(v) Failure frequencies are representative figures to emphasis the suitability of importance measure and do not represent the actual figures.

Since all the piping components are assumed to be in series model, the Medium LOCA frequency is found to be 1.14e-03/yr. Eventhough individually all component failures are Medium LOCA, their combined effect is considered in this case study.

5.2. Reliability Analysis of ECCS

Emergency Core Cooling System consists of 4 accumulators, where each accumulator holds 60 m^3 of cold water and 10 m^3 of nitrogen at 5 MPa (50 kg/cm^2) pressure. The accumulators are connected through check valve and rupture disc to main header from each line. Main header is connected to the channels through feeders. Each channel consists of 8 perforated water tubes, through which cold water ejects directly on the fuel pins. These water tubes are arranged in a manner to ensure adequate cooling of pins during LOCA. For the estimation of ECCS unavailability, the contribution comes from actuation logic, accumulator system, piping etc. For the purpose of our studies, we have broadly classified unavailability contribution as from piping and non-piping components. In order to analyse the aspect of In-Service Inspection, emphasis has been given on piping failure contribution from ECCS lines and header.

Assumptions

(i) System boundary is considered at accumulator and individual header. Other components such as inlet valves, nitrogen supply etc are not considered for ranking
(ii) System success criterion is “water from any three out of four lines reaching the common header”.
(iii) Failure probability of basic event “Piping” includes failure probabilities of welds, elbows, joints etc.
(iv) Basic event probabilities are representative figures to emphasis the suitability of importance measure and do not represent the actual figures.
(v) EICCF1, the common cause component, represents the failure of ECCS resulting from failure in common actuation signal.

The Fault Tree for ECCS failure is shown in Figure 3. The basic event failure probabilities are given in Table II. The fault tree has been analysed using PSAPACK 4.2 and minimal cutsets are obtained. Since the cutset truncation probability was taken as 1e-10, 79 cutsets were obtained for system unavailability calculation. The system unavailability was found to be 5.5e-03.

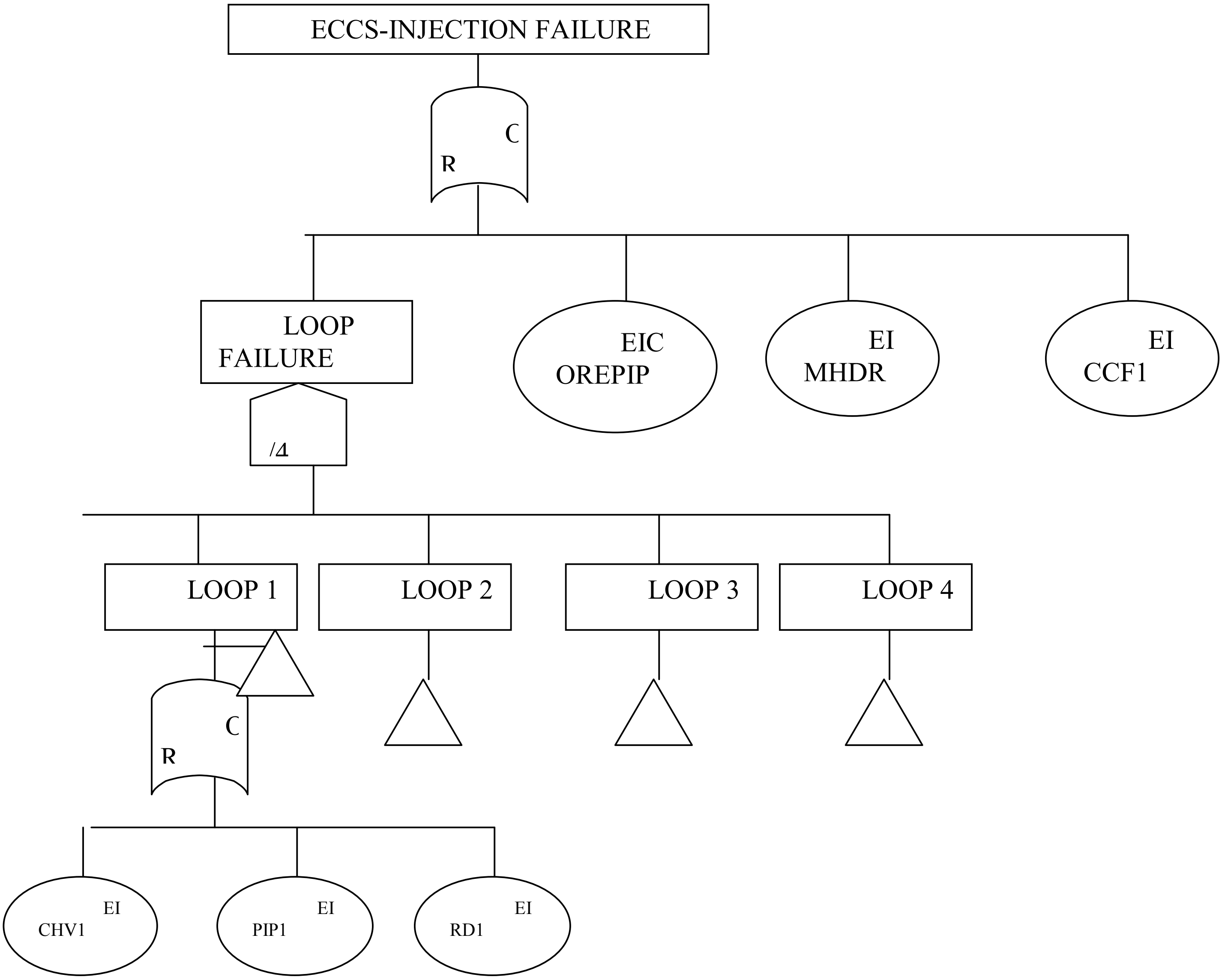

Figure 3. Fault tree for ECCS- high pressure injection.

Table II. List of Piping Segments to be ranked and their data

List of Piping components in MLOCA			List of Piping components in ECCS		
Sl. No.	Description	Frequency (/yr)	Sl. No.	Description	Unavailability
1	Feeders - Header1	9.2e-05	1	EICOREPIP	2.43e-03
2	Feeders – Header2	8.9e-05	2	EICHV1	1.0e-04
3	Feeders – Header3	6.5e-05	3	EIPIP1	8.85e-04
4	Feeders – Header4	9.8e-05	4	EIRD1	2.21e-03
5	Header-Steam Generator	1e-06	5	EICHV2	1.0e-04
6	Steam Generator –Pump	4e-06	6	EIPIP2	1.11e-03
7	Pump- Header	6e-06	7	EIRD2	2.21e-03
8	Coolant Channel	1.77e-05	8	EICHV3	1.0e-04
9	Header1	1.91e-04	9	EIPIP3	1.6e-03
10	Header2	1.91e-04	10	EIRD3	2.21e-03
11	Header3	1.91e-04	11	EICHV4	1.0e-04
12	Header4	1.91e-04	12	EIPIP4	1.14e-03
			13	EIRD4	2.21e-03
			14	EICCF1	1.0e-04
			15	EIMHDR	3.0e-03

The components in Emergency Core cooling System has been selected as a case for ranking and prioritization. They are ranked based on Importance Measure approach as well as Risk Matrix approach as shown in Table III. The Risk category will decide the frequency of inspection, scope of inspection and the type of inspection application for the particular component.

Table III. Component Ranking & Prioritisation for Risk based Inspection

Sl. No.	**Comp.Name**	**Inspection Importance Measure**	**Risk Category**
1	EICOREPIP	0.00243 (2)	1
2	EICHV1	3.60374e-009 (12)	5
3	EIPIP1	3.44908e-008 (11)	5
4	EIRD1	9.60683e-008 (4)	3
5	EICHV2	3.35077e-009 (13)	5
6	EIPIP2	3.98642e-008 (10)	5
7	EIRD2	7.09202e-008 (7)	5
8	EICHV3	2.97888e-009 (15)	5
9	EIPIP3	5.36666e-008 (8)	5
10	EIRD3	7.44685e-008 (6)	5
11	EICHV4	3.18209e-009 (14)	5
12	EIPIP4	4.19082e-008 (9)	5
13	EIRD4	8.11933e-008 (5)	5
14	EICCF1	0.0001 (3)	1
15	EIMHDR	0.003 (1)	1

6. CONCLUSION

In-Service Inspection programme includes various components from process as well as safety systems. In order to increase the availability of Plants, the ISI programme will be scheduled during plant shutdowns during which critical components are inspected. Mostly components are scheduled based on some deterministic ranking procedures. Risk Informed ISI provides a consequence criteria for ranking components. As a pre-requisite to the systematic life management programme, a methodical compilation of ageing relevant data on a plant wide basis is essentially required for successful implementation of the methodology.

REFERENCES

1] Syed Ali, Bagchi, "Risk Informed In-Service Inspection", Nuclear Engineering Design; Vol. 181: pp. 221-224, 1998.

2] K. R. Balkey, Nancy B. Closky, Jerry H. Phillips, "Developments on US NRC-Approved WOG/ASME Research Risk Informed In-Service Inspection Methodology, An internal report.

3] T. V. Vo, Bryan. F. Gore, Elizabeth J. B, Frederic A. Simonen, "Probabilistic Risk Assessment based guidance for piping In-Service Inspection", Nuclear Technology, Vol. 88: p. 13-20, 1989.

4] W. E. Vesely, "Principles of resource effectiveness and regulatory effectiveness for risk informed applications: Reducing burdens by improving effectiveness", Reliability Engineering and System Safety, 1999; 63: p. 283-292.

RUSSIAN FEDERATION NORMATIVE APPROACH TO THE QUESTION OF MANAGEMENT OF LIFETIME NPP

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Abstract:

In Russia, the designated service life of a nuclear power plant (NPP) is 30 years. During the period 2001-2010, 15 Russian NPP units will reach the end of their service life (see Table. I). In accordance with 'The program of Atomic Energy Development in the Russian Federation for 1998-2005 and up to 2010' adopted by Decree No. 815 of the Government of the Russian Federation on 21 July 1998, priority is placed on the preparation of NPPs for extension of service life and on ensuring safety in the extended operating period. General structure of normative base using in the Russia is shown on the fig 1. The 'Basic Provisions of NPP Safety Assurance', OPB-88/97, Item 5.1.14, provide for a possible extension of NPP operation beyond the designated service life. For such an extension, the NPP operating organization must apply for a license renewal to Gosatomnadzor, which needs to specify the relevant requirements. GAN is developing regulatory documents to provide a basis for NPP license renewal/extension of NPP operation, which would benefit from international experience. The normative document “Rules of construction and safety operation of equipment and piping of NPP” provide for such possibility too. Item 2.1.11 of this document say: “Life time of ” equipment and piping can be extended on the more time, then shown in the passport, on base of technical decision, which prepare by utility of NPP, design organization and manufacture. Strength calculations for conformation of life time extension possibility and acts of metal inspection have to add technical decision.

Table I. RUSSIAN NPP

NAME OF NPP	UNIT No.	TIPE OF REACTOR	POWER MWt	GENERATION	TIME OF COMMISSION	DESIGN LIFE TIME
BELOYARSKAYA	3	BN–600	600	I	1980	2010
BILIBINSKAYA	1	EGP–6	12	I	1974	2004
	2	EGP–6	12	I	1974	2004
	3	EGP–6	12	I	1975	2005
	4	EGP–6	12	I	1976	2006
BALAKOVSKAYA	1	VVER–1000	1000	II	1985	2015
	2	VVER–1000	1000	II	1987	2017
	3	VVER–1000	1000	II	1988	2018
	4	VVER–1000	1000	II	1993	2023
KALININSKAYA	1	VVER–1000	1000	II	1984	2014
	2	VVER–1000	1000	II	1986	2016
KOLSKAYA	1	VVER–440	440	I	1973	2003
	2	VVER–440	440	I	1974	2004
	3	VVER–440	440	I	1981	2011
	4	VVER–440	440	I	1984	2014
KURSKAYA	1	RBMK–1000	1000	I	1976	2006
	2	RBMK–1000	1000	I	1979	2009
	3	RBMK–1000	1000	II	1983	2013
	4	RBMK–1000	1000	II	1985	2015
LENINGRADSKAYA	1	RBMK–1000	1000	I	1973	2003
	2	RBMK–1000	1000	I	1975	2005
	3	RBMK–1000	1000	II	1979	2009
	4	RBMK–1000	1000	II	1981	2011
				I		
NOVO-	3	VVER–440	417	I	1971	2001
VORONEGSKAYA	4	VVER–440	417	II	1972	2002
	5	VVER–440	1000		1980	2010
	1	RBMK–1000	1000	II	1982	2012
SMOLENSKAYA	2	RBMK–1000	1000	II	1985	2015
	3	RBMK–1000	1000	II	1990	2020

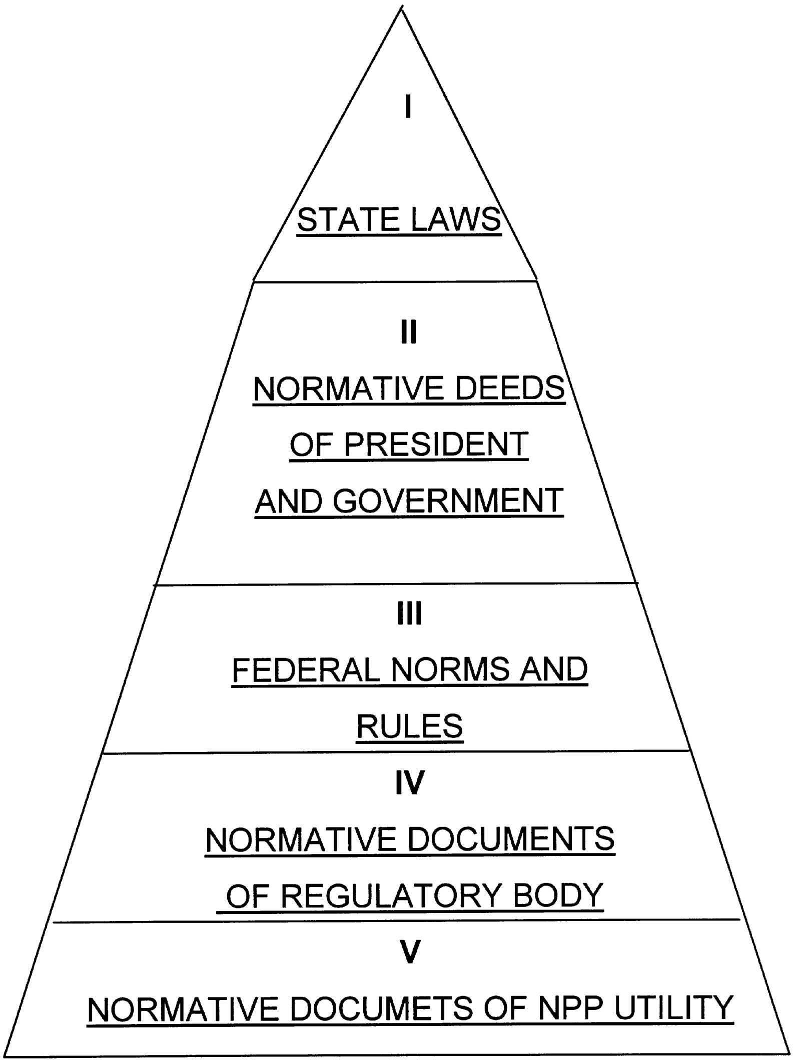

Fig 1. Pyramid of normative documents of Russia.

LIST OF BASE NORMATIVE LAWS AND NORMATIVE DOCUMENTS USING BY GOSATOMNADZOR OF RUSSIA FOR REGULATORY OF SAFETY IN FIELD OF ATOMIC ENERGY USING

This document contains the next main parts:

Laws in field of atomic energy using:

i. -international laws - 6
ii. -laws of Russia-9
iii. -orders of presedent of Russia -7
iv. -orders ofgoverment of Russia - 28

II. Structures and complexes with experemental and research nuclear reactors, ships with nuclear reactors, space ships and other transport devices – 26
III. Nuclear power plants - 120
IV. Structures, complexces, installations for production, using, reproduction, transportation of nuclear fuel and nuclear materials. Storage of nuclear materials and radiation waste - 50
V. Radiation sources, radio-active matters and radiation waste –57
VI. Guarding of nuclear installations, storages of nuclear materials and radioactive matters. Registration and inspection of nuclear materials –10

REQUIREMENTS TO STRUCTURE AND CONTENT OF DOCUMENTS WHICH BACKGROUND SAFETY UNDER LIFE TIME EXTENSION OF NPP UNIT

I. requirements to structure of documents which background safety under lifetime extension of NPP unit.
II. requirements to content of documents which background safety under life time extension of NPP unit.

REQUIREMENTS TO STRUCTURE OF DOCUMENTS

1. report with results of complex inspection of NPP unit.
2. report about deep assessment of NPP unite safety.
3. programme of preparing of NPP unite to lifetime extension.
4. summarized report about assessment of NPP unite safety for all period its operation.
5. list of deviations from requirements of actual normative documents by nuclear and radiation safety.
6. programme of insurance of work quality for lifetime extension of NRR unit.
7. report about background of safety during additional time of operation of NPP unit.
8. report with results of psa.
9. passport of NPP unite reactor.
10. act by results of fulfilled works for preparing of NPP unite to operation during additional time.
11. technological time limit of operation of NPP unit.
12. instruction for liquidation of accident on the NPP unit.
13. guidance for management of hard accident.
14. plan of measures for defense of staff in case of accident.
15. programme of diagnostic of technical state of NPP unit.
16. certificate about system of selection, training, qualification and admission to the work of staff of NPP unite.
17. programme of supervision for state of buildings and constructions of NPP unit.
18. documents which define order and procedures of application of indication devices to the nuclear materials.
19. time limit of maintenance and repair of equipments of systems which are important for safety.
20. certificate about organization of registration and inspection system of nuclear materials and about results of last stocktaking of nuclear materials.
21. certificate about insurance of guard of NPP unit.
22. act of commission which check organization of guard of NPP.
23. certificate about organization of registration and inspection system of radiation materials and radiation wastes.
24. instruction for fulfillment of inspections and tests of safety systems.
25.
26. instruction, programmes and schedules of maintenance and tests of important for safety systems.
27. instruction for insurance of nuclear safety in the time of storage, transportation and overloading of nuclear fuel.
28. schedules of inspection capacity for work of safety systems, control and management systems and ISI systems.

BASIC REQUIREMENTS TO THE LIFE TIME EXTENSION OF NPP UNITS

1. requirements to a complex inspection.
2. criterions of a possibility NPP unit operation during additional time.
3. requirements to preparation of the NPP unite to additional time of operation.

CRITERIONS OF A POSSIBILITY OF LIFE TIME EXTENSION OF THE NPP UNITE DURING ADDITIONAL TIME

- The operation of NPP unite over life time is possible in the case if necessary technical and organizational measures necessary for reduction of NPP unite in the correspondence with criterions and requirements of the acting norms and rules in the field of use of atomic energy are accepted.
- The technical state of the NPP unite during additional time of operation should satisfy to requirements technical (design, plant) documentation.
- During additional time of operation of NPP unit the activity should be performed for increase of safety in accordance with requirements of the acting normative documents in the field of nuclear, radiation, technical, fire, environment safety
- The residual safe lifetime of nonrestorable elements (equipment, buildings, structures and building constructions of the NPP unit), important for safety, should be justified and is sufficient during additional term of operation of the NPP unite, and also for decommision of NPP unit (after ending additional term of operation), if the elements of the NPP unit should be used in this period.
- The management of reliability of life time of the equipment, buildings, structures and building constructions of the NPP unite should be executed, for what should be developed and to be executed the program of management of the safe life time of these elements.
- The efficiency of methods and means of monitoring of technical state of the equipment, buildings, structures and building constructions of the NPP unit, important for safety, should be sufficient for identification and warning of design beginning events.

STATEMENT FOR MANAGEMENT OF LIFE TIME CHARACTERISTICS OF NPP UNIT ELEMENTS

1. methodology of management of life time characteristics of NPP unit elements;
2. requirements to development of special list of NPP elements for performance of works for management of life time characteristics;
3. procedure of management of NPP elements life time characteristics;
4. requirements to quality of works for management of life time characteristics; and
5. documentation for management of life time characteritics.

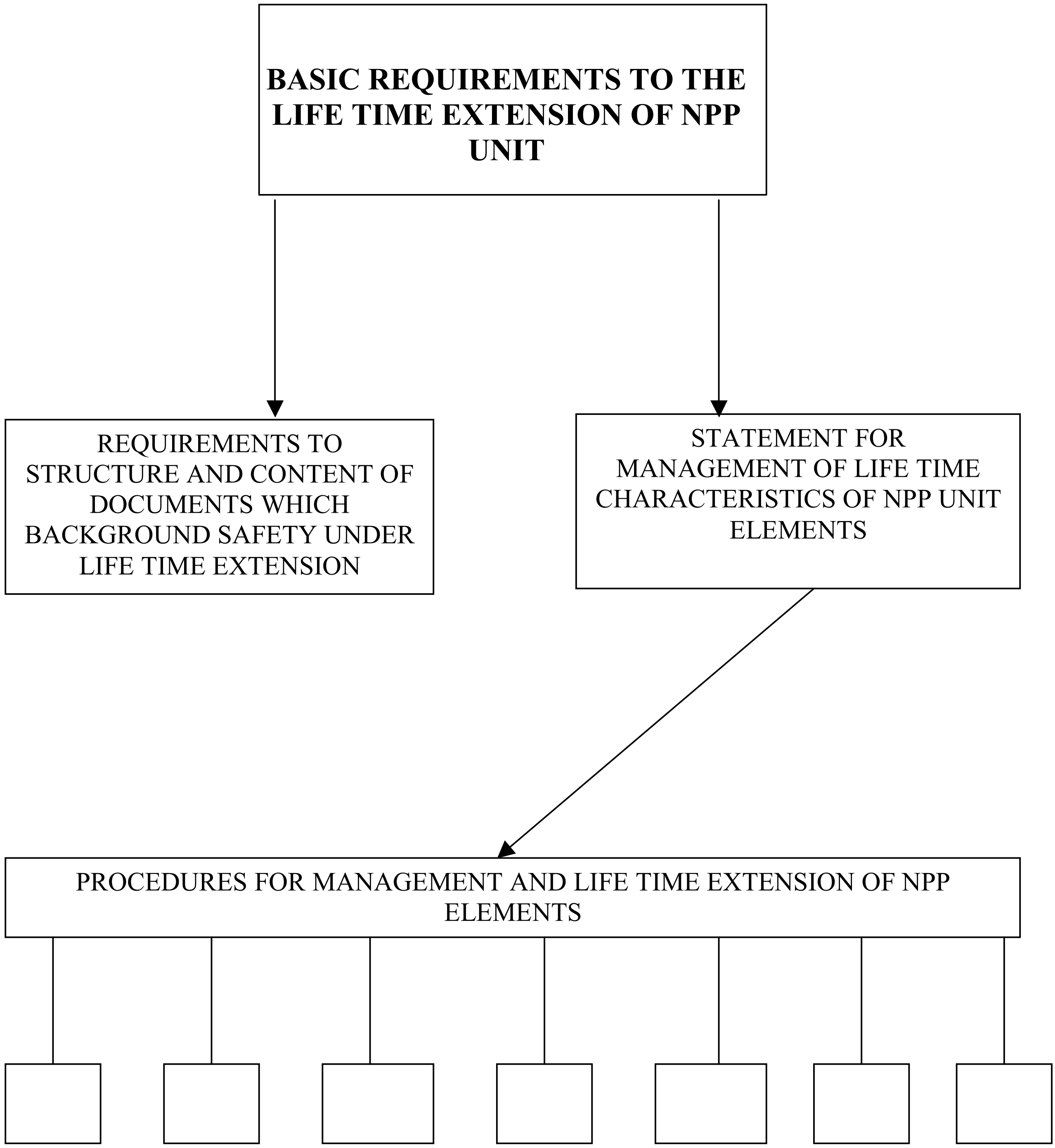

Fig. 2 Scheme of normative documents for life time extension and management of life time.

CONCLUSION

At this moment Russia have completed normative documents for lifetime extension and management of lifetime NPP.

SWISS REGULATORY USE OF DATABANKS FOR NUCLEAR POWER PLANT LIFE MANAGEMENT, SURVEILLANCE AND SAFETY ANALYSES

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Abstract.

As operational time is accumulated, the overall safety and performance of nuclear power plants (NPPs) will tend to be characterized by those areas in which structures, systems and components (SSCs) have not performed as well, or as reliably, as expected. The reasons for non-availability, or failure on demand either in use or found during periodic testing of equipment, leading to reportable events or even accidents, are varied and they must be analysed in order to obtain the root causes. Once root causes are identified, corresponding measures can be applied in order to improve SSC reliability and therefore increase availability and safety. The event information obtained, if brought into a user-friendly databank, can be used to follow NPP performance trends, to check whether a repair or replacement has been effective, to focus regulatory attention and NPP own surveillance on known weak-spots and to serve as an advance indicator where problems may be expected to arise in future. Using the databank approach, similar occurrences of failures or problems in other NPPs can be identified and generic issues recognised early on and preventive action taken. The following describes the Swiss Federal Nuclear Safety Inspectorate's databank for keeping track of NPP reportable, classified events and safety and associated lifetime management issues. The probabilistic safety analysis databank is also briefly described. Examples of the use and application of NPP reportable event data are discussed. The value of record keeping and databanks in NPP life management and supervision of safety performance is demonstrated.

1. INTRODUCTION

Switzerland has 5 NPPs in operation, namely Mühleberg (G.E. Boiling water reactor, 355 MW, in operation since 1972), Beznau 1 and 2 (Westinghouse, Pressurized water reactor, each 365 MW, in operation since 1969 and 1971 respectively), Gösgen (KWU). Pressurized water reactor, 970 MW, in operation since 1979) and Leibstadt (G.E. Boiling water reactor, 1145 MW, in operation since 1984).

It can thus be seen that the Swiss NPPs are varied in type and differ considerably in age. These facts present special challenges concerning not only the supervision and oversight of safety but also in documenting operational data in terms of reportable events and their root causes. The Swiss Federal Nuclear Safety Inspectorate (HSK) recognized early on that a systematic collection of event data would facilitate an improved oversight of how the NPPs were performing with regard to safety. Accordingly, a reportable event databank (REDB) was created, whereby all relevant details concerning classified events could be collected. For completeness, the REDB was also conceived to allow the event characteristics to be coded according to the IAEA/NEA reporting guidelines: "Incident Reporting System" (IRS) [1]. Another feature is the inclusion of the IAEA/NEA International Nuclear Event Scale (INES) [2] number of the event. As a further development, the REDB has also been integrated with additional information necessary for the probabilistic safety analyses (PSAs) carried out on the NPPs (the PSADB). In this way both deterministic and probabilistic approaches are brought together in one databank. This provides an integrated approach to documentation of operational events and behaviour of SSCs in NPPs.

The following focuses mainly on the Inspectorate's REDB, and describes the process of data collection, evaluation and inclusion in the REDB. Practical use of the data is explained. Some further comments are given with respect to the PSADB and the data collected so far.

2. FEATURES OF THE REPORTABLE EVENT DATABANK (REDB)

2.1. Reporting and data collection

The Swiss NPPs are in possession of the HSK Guideline R-15: "Reporting Guideline Concerning the Operation of Nuclear Power Plants" [3], and this ensures, already at an early stage, the correct categorization and initial assessment as to the type of event and its impact on safety. The advantage of using such a guideline is that the reportable and classifiable events occurring in the Swiss NPPs are defined and the criteria for their assessment on safety impact are given. In this manner, a standard approach for reporting on events and data collection is facilitated.

It should be noted that the reportable event scales of severity of impact on nuclear safety, as given in HSK Guideline R-15, have no direct relation to those given in INES. However, as a rough guide, a B-Event in R-15 (detailed below) is generally equivalent to Level 0 on the INES scale. In any case, events rating 2 or higher on the INES scale are subject to informing the IAEA according to the IAEA "Event Rating Form (ERF)." Events classified as S in R-15 (see below) are also notified to the IAEA through the ERF. The R-15 is a Guideline to enable the NPP licensees and the HSK to categorize in detail and to assess reportable events in the Swiss NPPs. The 5 event type categories in R-15 are called S, A, B, U and Ö. They are, in decreasing importance to nuclear safety, as follows:

Event type S: Events that cause a danger to the nuclear installation or personnel, or those that have a significant radiological effect on the environment.

Event type A: Events of safety significance having, however, only minor or no radiological consequences to the environment.

Event type B: Events of minor safety significance. Such events are registered and evaluated by both the licensee and the Inspectorate in order to facilitate possible early identification of weak spots.

Event type U: Events of interest concerning safety. Category U-events do not fulfil any of the criteria for events in the S, A or B category. Reports on U events serve only to inform the Inspectorate.

Event type Ö: Events of public interest, which are not covered by category S or A, but requiring a faster notification than B or U.

The Inspectorate's REDB contains mostly R-15-B events, since only one R-15-A event has occurred and no R-15-S event has ever taken place in any Swiss NPP, to date. The B, A or S events are reportable and classifiable events, whilst the R-15-U events are reportable, but not classifiable, serving only to inform the HSK. Some of the more interesting or unusual R-15-U events are, however, also entered in the REDB. It is planned to extend these entries in the next time in order to aid and support ongoing PSA analyses. The R-15-Ö events cover minor events, such as a sudden release of steam, which do, however, cause public interest.

When an event occurs, the NPP concerned sends a quick report, based on known facts, and an initial event-type assessment according to Guideline R-15, to the Inspectorate. This covers such details as the time and date, NPP status when the event occurred, a short description of the event, the affected SSCs, the measures taken to control or mitigate the event, the probable cause, the effect on safety, the NPP's own assessment of the event type and also the initial estimate of the INES classification, where applicable. A final report from the NPP eventually follows, and this may differ in detail to the initial one, since time has been available for the NPP to make a thorough analysis of the event. In particular, the NPP's own root cause analysis and consequences and measures to be done in order to avoid a recurrence, if possible, are laid down in the final report. Once the HSK's experts have all the necessary information, they analyse the features of the event, coming to their own conclusions. These conclusions are compared to those given by the NPP, and agreement, or not, may be present. If serious differences exist with respect to the root cause and level of impact on nuclear safety, letters are exchanged between the Inspectorate and the NPP licensee to clear up discrepancies. In any case, the final assessment of the HSK, in the form of an official letter to the NPP concerned, is the binding one. Assuming the guideline has been used correctly, and no serious discrepancies in assessment of root cause and impact on safety exist, which is mostly the case, the event data is entered into the REDB.

2.2. REDB Data entry

The REDB is based on ACCESS software, and has a user-friendly mode by which the relevant data concerning events may be entered. An overview of the structure of the REDB is provided in this section.

The following data and information is entered into the REDB: 1. The NPP involved and the event identification through a running number. 2. The full title of the event. 3. The date of the event. 4. A short description of the event 5. The IAEA/NEA Incident Reporting System (IRS) coding. 6. The root cause of the event 7. The changes or modifications made to the NPP or operational procedures and 8. A brief comment and identification of lessons learned. The event type, as described in the HSK Guideline R-15, is also included, as well as the approximate level according to the International Nuclear Event Scale (INES). This latter information enables another perspective to be obtained concerning the event severity.

The condensed description of the event is mainly derived from an abstract from the NPP's final report text. The Inspectorate, where necessary, also adds comments that have arisen from its own final assessment report. This ensures a short, informative and precise representation of the reportable event.

3. EXAMPLES OF DATABANK

3.1. Reportable event data bank (REDB)

An event registered in the REDB is entered into the fields generated by the software. The fields are self-descriptive and allow convenient separation of the various aspects of the event. The data may be filtered, and a search tool, using a keyword or other attribute, can be activated to find similarities in reportable events between the NPPs. The relative frequency of, for example, diesel-related events, can be followed for each NPP. More generally, the trending of event classification and event frequency with time gives information whether a specific NPP is 1) experiencing reportable events at decreasing periods of time, 2) events remain scarce or show a rate of occurrence typical of previous years or 3) shows decreasing numbers of reportable events with time. Such information is helpful to assess whether a NPP is showing signs of degradation, or even undergoing continual improvement with respect to the occurrence of reportable events. Depending on the trends, the Inspectorate is in a better position to supervise known or developing weak spots in NPP safety, or to indicate where inspection intervals should be adapted. It is emphasized that the NPPs have anyway their standard inspection and maintenance schedules and tests to perform at prescribed intervals.

The REDB presently contains 300 "B" events, as classified under HSK guideline R-15 ("B" events are roughly equal to INES 0). These have occurred over an accumulated total period of about 100 years of actual NPP operation in Switzerland. This gives an expected average number of R-15-B events of 3/year/NPP, assuming all other things remain equal (type, age etc.). But such a simple approach for analysing "B" events is not suitable. For example, since one NPP will have 3 emergency diesel generators and another 5, it may be expected that the latter will have more chance to fail. However, in this case, the probable importance of each diesel is somewhat less than when only three are present. Furthermore, average numbers alone may not be sensitive enough to detect gradual underlying trends. It has been found that a continual and plant-specific trending and documentation are more able to detect drifts in "B" event frequency. Analysing the root causes for the event will further indicate whether ageing in SSCs or even human factors are increasing in significance. Knowing the facts allows the regulator to focus resources for oversight accordingly, and the licensee will also be able to better plan inspections or to raise questions concerning human factors, as necessary.

Other analysis possibilities on reportable event occurrence may be extended to examine the first 10 years of a given NPP's operation (here, for convenience, designated the "learning phase"), and comparing this with the following 10 years, for example. Theoretically, it is reasonable to assume that a "bathtub" function of number of events versus operational time may be followed i.e. more events at the running-in and learning phase, followed by a period of stable behaviour and then, eventually, increasing frequency of events as ageing of SSCs or personnel (human factors, complacency etc.) takes place. Of course, preventive maintenance, replacement of SSCs, personnel training and awareness, and other factors can all work to delay the theoretically expected onset of the increase in event frequency. This will then show if there is an effect of ageing management strategies. The REDB, with appropriate treatment of the data, is capable of detecting whether such trends are present.

Studying the evolution of reportable events with time can provide some interesting results. An example is that, in general, diesel generators have been shown to fail during monthly functionality tests. Failure on demand in a real situation is, by contrast, a comparatively rare occurrence; thus there is a relatively poor statistical base to work with. The reasons for diesel generator failures on test are varied (connecting rod break, electrical fault on switches, mechanical binding in valves, etc.), and random occurrences of failure during tests are main causes of reportable events concerning diesels. This indicates that lessons concerning random failures are being learned continually, and that recurrent events may be increasingly avoided as the knowledge base expands. It is difficult to directly address such random phenomena. Lessons may still be learned, however, and areas where extra attention is necessary are identified.

Other SSCs that feature in reportable events are associated with electrical components, usually as a result of ageing causing changes in properties (resistance, magnetic etc.). The NPP licensee accrues experience with those SSCs that are prone to unsatisfactory performance and makes the servicing, revisions and replacements accordingly. The REDB helps the regulatory authority to control whether this is done in an optimum manner.

3.2. Probabilistic safety analysis databank (PSADB)

Data included in the PSADB have been collected over many years from various sources such as NPP event reports, monthly and annual reports and specific PSA studies. Events that are of interest and importance for PSA encompass a far wider range than the reportable events entered in the REDB.

It should be noted that REDB and PSADB were originally two separate databanks, and some of the events now described in the REDB were originally already part of the PSADB. Therefore, it was decided to merge the two data sets and now REDB and PSADB are each part of the same databank. Where applicable, the description of the event in the REDB has been completed with additional information from the PSADB. In order to help the user find specific data, the merged version of the databank, which is presently at the development stage and undergoing improvements, will be supplied with extended search functions allowing full-text search over all fields, or over a combination of defined fields.

The PSA data are entered into the PSADB a similar way to those of the reportable events. A typical PSADB event provides such information as a short description of the failure, the component type and number, the system, failure-mode (fails to start, fails to close etc.), failure-type (single, common cause etc,), the date and time involved and literature references. The latter are the original sources of the information. The database allows various controls to be performed, such as how good revision and maintenance are carried out. Recurrent events can be detected easily. The PSADB allows a systematic search to be carried out for important components in order to check the statistic of the licensee's PSA. Furthermore, the databank is the electronic memory of all events relevant for the regulatory body. This aspect will take on increasing significance as more persons go into retirement, and their knowledge of the events have to be recorded in a systematic manner.

4. DISCUSSION

A databank, such as that described here, requires that capacity is available to assure data accumulation in an efficient manner. But, the value of any databank depends on the accuracy and therefore quality of the data it contains. Only a quality-assured (QA) system, having the ability to identify, categorize and ascribe appropriate attributes to the event, will fulfil these requirements. To this end, the HSK has in place a Guideline, R-15, which allows NPPs to report on events in a standardized way. After performing the necessary control on the NPP's own assessment of the event, by comparing and cross-checking with the Inspectorate's own analysis, a satisfactory level of QA on the data is achieved. In addition to the Inspectorate's safety impact rating, according to the Guideline R-15, the IAEA/NEA Incident Reporting System (IRS) coding and the INES event rating is also included in the databank entry for the event. In this way it is possible to perform some broad comparisons with similar international reports on SSCs and event types and their severity, root causes and mitigation.

Since the QA of the data must be high, the iterative processes to achieve this take time. However, the advantage is the certainty that any analyses performed will be based on facts and results will have an inherently high level of significance. This favours objectivity in analysing events and any indicated trends in event frequency and for given SSCs.

Both the REDB and PSADB are living databanks, being added to as more data becomes available. The information is thus a precise, continuing chronology of each NPP and the SSCs that have caused reportable events. The repair or replacement of SSCs is recorded, and it can be followed up as to whether the performed actions have reduced the occurrence of reportable events. Lessons learned provide the foundation for creating vigilant and questioning attitudes, which are essential for maintaining effective oversight on NPP safety. The REDB registers where weak spots or recurring problems occur, and thus is an aid for more focussed inspections. In this way, a more risk-informed approach may also be addressed.

There are many ways in which the Inspectorate's databanks can be used. The data can be analysed for trends, statistical treatments can be applied, for example, estimating probabilities of failure, identifying event distribution-types (Gauss, skew, Poisson) and following of NPP performance before and after modifications on SSCs etc. A significant part of the work remains in obtaining the QA data and using it in an appropriate manner.

From a regulatory aspect, the safe operation of NPPs is mainly related to maintaining functionality and integrity of SSCs and ensuring that technical specifications (TS) are adhered to. From an owner's point of view, who in every case carries the responsibility for safety, optimum plant performance is seen as a function of electricity produced and sold, and this will be dependent on the absence of unplanned outages, scrams and other aspects affecting the safety and economics of operation. Knowing which SSCs are most liable to contribute to unplanned outages and reportable classified events will be advantageous to NPP licensees, since inspection, replacement and maintenance programmes may all be optimized for both safety and production considerations.

Although not yet an issue in Switzerland, it is recognized that deregulation of electricity markets may bring with it special issues and concerns. The need to save money and increase competitiveness may lead NPP licensees to make reductions in personnel and to try and optimize inspections and preventative maintenance. The impact of such economically driven measures may not become immediately apparent, but unfavourable trends that may develop can be recognized and counteracted if meticulous record keeping is done and databank analyses carried out. This indicates that the potential value of record keeping will increase in today's global movement towards deregulated electricity markets.

5. CONCLUSIONS

Databanks are effective tools to follow trends concerning reportable event evolution with time. This enables identification of those SSCs liable to be involved in reportable, classified events.

The value of record keeping and databanks in safety analyses has been shown. Accordingly, the Inspectorate's REDB/PSADB approaches are playing an increasingly important and integral role for the recording and analysis of reportable events. It is recognized that, due to the general loss of personnel through retirement, it is essential to record data and, in a broader sense, know-how. Databanks serve as an electronic memory, which may be used to analyse and compare present events with former ones. An updated databank is a tool to enable a nuclear regulator to follow the evolution of types of event, and what SSCs are showing degraded or less reliable performance. The REDB thus enables a more focussed approach to inspections on SSCs and the time intervals that are used.

An important task of the regulator is to be able to verify whether NPP licensees are conscientiously following their liability to ensure safe operation. Safe operation of NPPs has the highest priority, and this will also be the case in a liberalized electricity market. When SSCs are well maintained, and all operational practices in the NPPs are continually optimized, reliability and reductions in the frequency of reportable events and forced outages will all be favoured.

In cases where liberalized electricity markets are present, it will be necessary to monitor whether there has been a negative effect on safety performance. Databanks and record keeping will provide an insight on whether the root causes of reportable events and failures of SSCs are traceable to factors concerned with any changes in NPP organizational and operational practices.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY AND NUCLEAR ENERGY AGENCY OECD, International Reporting System (IRS) Reporting Guidelines, Feedback from Safety Related Operating Experience for Nuclear Power Plants, Vienna (1983).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY AND NUCLEAR ENERGY AGENCY OECD, International Nuclear Event Scale (INES), Vienna (1992).

[3] SWISS FEDERAL NUCLEAR SAFETY INSPECTORATE (HSK), Reporting Guideline Number R-15 Concerning the Operation of Nuclear Power Plants, HSK, Villigen-HSK, Switzerland (1999).

AGING MANAGEMENT REVIEW FOR LICENSE RENEWAL AND PLANT LIFE MANAGEMENT

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Abstract:

In 1995, the U.S. Nuclear Regulatory Commission (NRC) published 10 CFR Part 54 that provided the requirements for an operating nuclear plant to seek license renewal. At present, the NRC has approved renewal of operating licenses for ten nuclear units and has applications under review for 16 more units. The approval of a license renewal application by the NRC gives the utility an option to continue operation for another 20 years provided the plant is willing to commit to the agreed upon aging management programs. These aging management programs form the bases for a comprehensive Plant Life Management Program that will manage the material condition of the plant, optimize the operating life of plant systems, structures, and components, and maximize plant value.

1. BACKGROUND

United States nuclear power plants are initially licensed for a period of 40-years. The 40-year term, which was established by the Atomic Energy Commission in the 1950s, is believed to be based on engineering judgement and is consistent with the typical amortization schedule for purchasing fossil power plants. Under 10 CFR Part 54, the license renewal rule, additional terms of 20-years may be obtained through the preparation of a license renewal application that must be reviewed and approved by the Nuclear Regulatory Commission (NRC). The license renewal rule requires that applicants perform aging management reviews on passive long-lived structures and components to demonstrate aging will be managed during the period of extended operation (i.e., additional 20 years of operation). Aging of active components, which are excluded from 10 CFR Part 54, is accomplished through the Maintenance Rule, 10 CFR Part 65, using performance-based monitoring.

The license renewal rule, 10 CFR Part 54, was initially published in 1991. After significant interaction with the nuclear industry from 1991 through 1994, the NRC revised the rule in 1995 to focus on passive long-lived structures and components. In 1998, the first two applications for license renewal were submitted to the NRC by Baltimore Gas & Electric for the two-unit Calvert Cliffs nuclear power plant and by Duke Energy for the three-unit Oconee nuclear power plant. In March 2000, the NRC approved the application for the two-unit Calvert Cliffs nuclear power plant for an additional 20 years. Two months later, the NRC approved the renewal of the operating licenses for the three-unit Oconee nuclear station. The NRC completed these reviews in a timely, predictable, and stable manner.

At present, the NRC has approved renewal of operating licenses for ten nuclear units and has applications under review for 16 more units. Twelve additional companies have notified the NRC of their intention to seek license renewal for 27 nuclear units by 2005. It is anticipated that over 90% of the 103 operating nuclear plants in the United States will pursue license renewal and seek an additional 20 years of operation. Some plants may pursue operation to 80 years or longer since the license renewal rule does not limit the operating life of a nuclear power plant.

The estimated cost to prepare and process a license renewal application is approximately $10M to $15M, which includes NRC review fees. The NRC review for license renewal is strictly a safety review and plant economics is not a consideration. However, economics will drive the decision to pursue license renewal for U.S. nuclear power plants. For nuclear units with strong performance records, license renewal is a good business decision when compared to the cost of building new generating capacity.

2. UNITED STATES REGULATORY REQUIREMENTS LICENSE RENEWAL-10 CFR PART 54

The requirements for renewing the operating license of a nuclear reactor in the United States are contained in NRC Regulation 10 CFR Part 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plant [1]. The license renewal rule requires that each owner prepare an application containing the following information: (1) General Information—10 CFR 54.19, (2) Technical Information—10 CFR 54.21, (3) Technical Specifications—10 CFR 54.22, and (4) Environmental Information—10 CFR 54.23.

2.1. General Information—10 CFR 54.19

The applicant must include information requested by the Attorney General for antitrust review (i.e., 10 CFR 50.33 (a) through (e), (h), and (i)), and changes to the standard indemnity agreement to account for the expiration term of the proposed renewed license.

2.2. Technical Information—10CFR 54.21

The process required to complete the technical requirements of the license renewal rule is the most labor intensive of any of the four elements and includes the integrated plant assessment (IPA) and time limited aging analyses (TLAA). The IPA process is summarized in Steps 1 through 3, and the TLAA process is discussed in Steps 4 and 5.

Step 1--Determine SSCs Within the Scope of License Renewal

This step requires that each applicant determine the systems, structures, and components (SSCs) within the scope of license renewal. The SSCs within the scope of LR are those that are required to mitigate design basis events and those SSCs that are required to mitigate the regulated events (e.g., Fire Protection, Environmental Qualification, Anticipated Transients Without Scram, Pressurized Thermal Shock, and Station Black Out). The output of the first step is a list of SSCs that are within the scope of license renewal and the associated functions that the SSCs must perform. These functions must be protected for the period of extended operation.

Step 2--Determine SCs Subject to Aging Management Review

From the list of SSCs within the scope of license renewal (Step 1), passive long-lived structures and components (SCs) must be selected for aging management review. Passive components subject to aging management review are defined as those components that perform a function, which directly supports a system function within the scope of license renewal (output of Step1), without moving parts or without a change in configuration or properties. Long-lived components are those components that are not subject to replacement based on a qualified life or specified period of time. Aging of active components (e.g., valve internals and pump motors) is managed by the Maintenance Rule, 10 CFR Part 65, which is a performance-based rule.

In general, the components to be selected for aging management review may be divided into three categories: (1) mechanical--function to protect is primarily pressure boundary, (2) electrical--function to protect is primarily electrical continuity, and (3) structural--function to protect is primarily structural integrity. Mechanical components include, but are not limited to, pipes, tanks, vessels, heat exchangers, valve bodies, and pump casings. Electrical components include, but are not limited to, cable, terminal blocks, and splices. Structures subject to aging management review support, house, and protect systems and components that are required for safe plant operation. Structures are fabricated from structural components such as beams, columns, floors, walls, and foundations.

Step 3--Aging Management Review

For those SCs selected in Step 2, an aging management review must be performed to ensure that the structure or component function will be maintained in the period of extended operation consistent with the current licensing basis. The aging management review process requires that aging effects be defined for the structures and components, and that aging management programs (AMPs) be reviewed to determine if the aging effects are managed. Examples of programs credited for aging management of mechanical components include ASME Section XI, Boric Acid Wastage Surveillance program, Flow Accelerated Corrosion, and Chemistry Control Program. The NRC has published a Generic Aging Lessons Learned Report (GALL), NUREG-1801, Volumes 1 and 2, to assist with the identification of acceptable existing aging management programs.

Step 4--Identification of Time Limited Aging Analyses

For those SSCs within the scope of license renewal, as identified in Step 1, time limited-aging analyses (TLAA) must be identified. Time limited aging analyses are those licensee calculations and analyses that form the basis for a licensee conclusion regarding the capability of the systems, structures, and components within the scope of license renewal to perform their function(s) that-

. Consider the effects of aging; and,

. Are based on explicit assumptions defined by the current operating term of the plant (i.e., forty years).

Examples of TLAA include environmental qualification (EQ), reactor vessel embrittlement (upper shelf energy and RT_{PTS}), and fatigue cumulative usage factors.

Step 5—Evaluation of TLAA

Once identified as a TLAA, three options are available for evaluation: (1) show that the analyses remain valid for the period of extended operation; (2) reevaluate or project the analyses to the end of the period of extended operation; or, (3) ensure that the aging effect is managed by a program during the period of extended operation.

2.3. Technical Specifications—10 CFR 54.22

Each application must include any technical specification changes or additions necessary to manage the effects of aging during the period of extended operation as part of the renewal application.

2. 4. Environmental Information—10 CFR 54.23

Each application must include a supplement to the environmental report that complies with the requirements of Subpart A of 10 CFR Part 51. The environmental supplement includes an evaluation to determine potential environmental impacts associated with an additional 20-years of operation. Examples of issues that must be assessed include endangered species, aquatic ecology, socioeconomic impacts, radiological impacts, and sever accident mitigation alternatives. The complete list of environmental issue that must be addressed in a license renewal application are contained in the Generic Environmental Impact Statement for License Renewal of Nuclear Plants (NUREG-1437, Volume 1). [3]

2.5. NRC Review and Approval of a License Renewal Application

After completing a license renewal application for a specific plant or group of plants (e.g. Oconee Units 1, 2, and 3), the utility must submit the application to the NRC for review and approval. The cost to prepare the application for submittal to the NRC ranges between $4M-$6M. The NRC review process takes approximately two years to complete and includes technical and environmental reviews by the NRC, site audits by the NRC, NRC Safety Evaluation Reports, an annual update to license renewal application to capture plant modifications, and updates to site-specific reports due to responses to NRC questions. NRC review costs are estimated at approximately $5M per application. Once approved, the NRC issues the plant a new operating license that extends the expiration date of the original license for a period of up to 20 years. The total cost to the utility for completing and processing a license renewal application ranges between approximately $10M to $15M.

At present, ten operating licenses have been renewed and 16 other plants have submitted applications to the NRC for review. It is estimated that the majority of remaining plants in the United States will pursue license renewal providing the option to extend operation to 60-years for each unit.

3. LICENSE RENEWAL AND PLANT LIFE MANAGEMENT

The license renewal rule focuses on aging of passive long-lived components and aging management programs that manage those structures and components. The majority of aging management programs credited in a plant license renewal application are existing programs (e.g., ASME Section XI, Chemistry Control Program, and Steam Generator Integrity). Very few new programs are required for license renewal and at this time there have been no commitments made by any utility to replace components as a direct result of license renewal. Typical examples of new programs required for license renewal include Reactor Vessel Internals AMP to manage irradiation-assisted stress corrosion cracking and neutron embrittlement, Small Bore (i.e., less than 4-inch NPS) Class 1 Piping AMP to manage cracking by fatigue, Alloy 600 AMP to manage primary water stress corrosion cracking, Buried Piping AMP to manage loss of external material due to corrosion, and Buried High Voltage Cable AMP to manage degradation of cable exposed to wetted environments.

After the NRC has approved a license renewal application, the credited aging management programs (i.e., existing and new) become commitments for the remaining plant life. These commitments may be implemented and tracked through a comprehensive plant life management program (PLIM). PLIM is the process by which nuclear power plant staff integrate equipment aging management, operations, maintenance, licensing, engineering, economic planning, and other activities to (1) manage the material condition of a system, structure, and component, (2) optimize the operating life of the system, structure, and component, and (3) maximize plant value. A comprehensive PLIM can save the operating plant a significant amount of money by effectively planning and implementing component refurbishment and replacement. Plants that do not plan ahead may incur significant losses due to obsolescence and age-related failures that may result in plant shutdown (Figure 1).

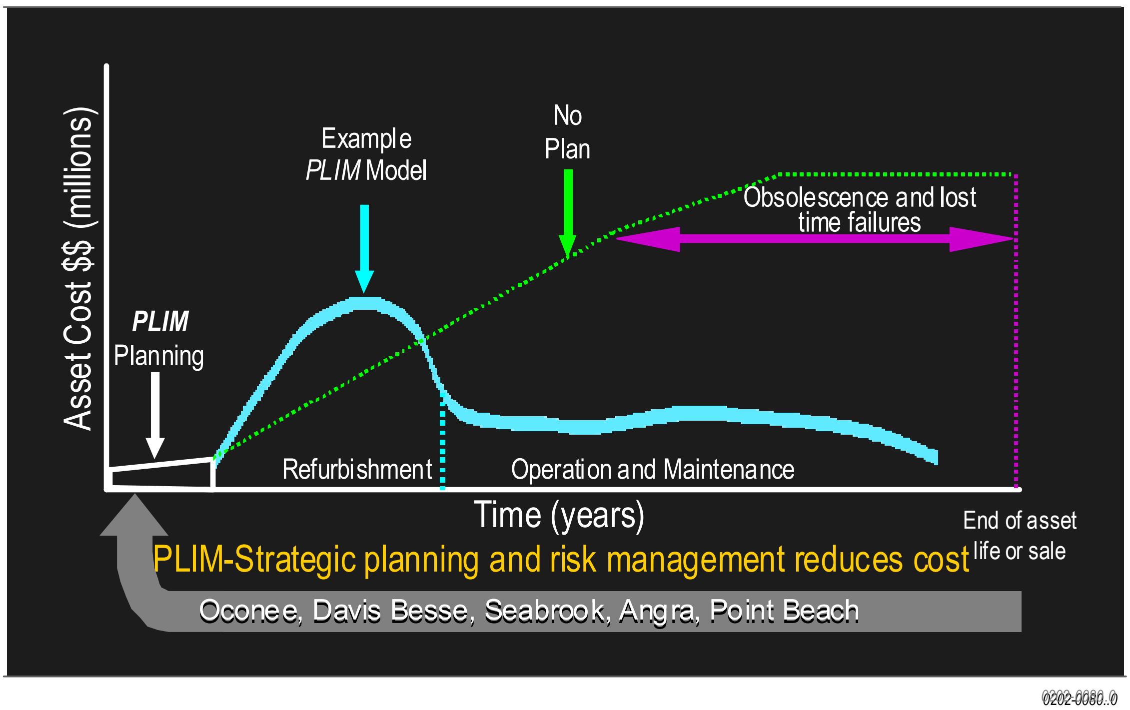

FIG.1--PLIM Strategic Planning--Time versus Asset Cost

PLIM planning includes financial modeling, long-term aging strategies, preventive maintenance optimization strategies, license renewal aging management program commitments, and obsolescence planning. Planning for PLIM includes all aspects of plant operations and maintenance (Figure 2).

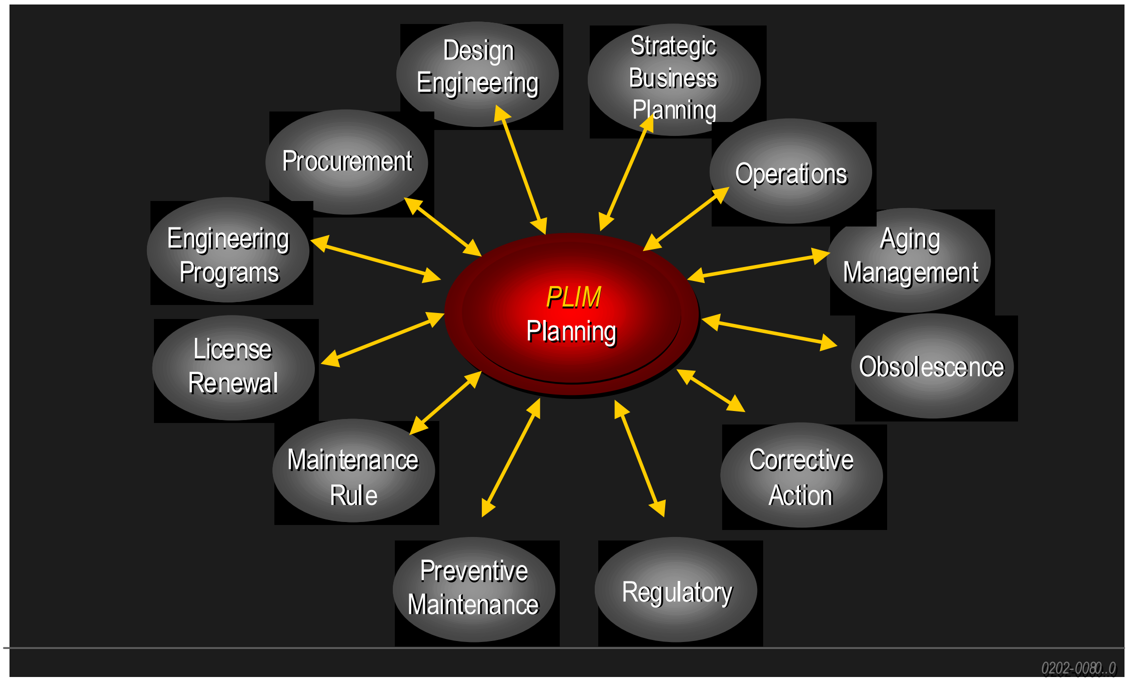

FIG. 2--PLIM Planning

PLIM provides a technical basis to support decisions regarding refurbishment and replacement, alternative aging management approaches, optimized economic evaluations, and asset management plan. PLIM provides a coordinated and consistent process to evaluate capital requests, modification decisions, operating procedure revisions to manage aging, and justification and guidance for monitoring and repair versus replacement.

US plants that have recently received renewed operating licenses and have implemented a comprehensive PLIM that includes NRC-approved aging management programs are now replacing components even though the license renewal review found existing aging management programs adequate to ensure the safety of those component (e.g., steam generators and electrical I&C cable). For example, following approval of the Oconee license renewal application in May 2000, Duke Energy announced they would replace all six once-through-steam generators, three reactor vessel heads, and significant electrical instrumentation and control cable and equipment at each of the three Oconee Units. Shortly after approval of the ANO-1 license renewal application in 2001 Entergy announced plans for steam generator replacement.

4. CONCLUSION

In summary, the regulatory requirements for the renewal of plant operating licenses in the United States are described in 10 CFR Part 54. The license renewal process has been demonstrated to be predictable and stable and provides the foundation for a comprehensive plant life management program. The approval of a license renewal application by the NRC gives the utility an option to continue operation for another 20 years provided the plant is willing to commit to the agreed upon aging management programs. The aging management programs become commitments for the remaining life of the plant, but the ultimate decision to continue operation is based on plant life management, which considers aging, safety, and economics.

REFERENCES

[1] "Requirements for Renewal of Operating Licenses For Nuclear Power Plants, "10 CFR Part 54
[2] "Generic Aging Lessons Learned Report," NUREG-1801, Volumes 1 and 2
[3] "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," NUREG-1437, Volume 1
[4] "Standard Review Plan for License Renewal," NUREG-1800

Session 2

PLANT LIFE MANAGEMENT (PLIM)

Sub-Session 2.2

Chairpersons

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USE OF PLANT SPECIFIC INFORMATION IN LIFE MANAGEMENT

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Abstract

In plant life management decisions are made on prevention and mitigation measures of ageing phenomena. In these processes, information from several sources has to be combined, and the decisions are based on data and analyses including lots of uncertainties. In order to make good decisions, the uncertainties and limitations related to both analyses and the raw data should be recognized. In this paper we present a view upon the decision making process in managing the ageing of components, systems and structures. Further, we propose practices to improve the transparency of ageing analyses and means to improve the availability and usability of plant specific information for ageing management purposes. As an example of improvement in the data collection practices we summarize the pipeline analysis and monitoring system that VTT is developing together with TVO. The system is meant to contain all up-to-date information necessary to analyze and monitor piping systems of an operating plant.

1. INTRODUCTION

The availability of plant specific information and easy access to these records are vital for the efficient ageing management. A lot of plant specific information is needed in ageing assessment. It is also very important to ensure that the information is up-to-date, e.g. possible modifications are taken into account in lifetime predictions.

Guidelines for good practices in data collection and record keeping are provided e.g. in an IAEA Safety Practice document [1]. Some of the key aspects are listed here:

– The information system should provide comprehensive and accurate information about components, including baseline information and operation and maintenance data.
– Data should be entered by maintenance and operations personnel, and the data entry should include an appropriate quality control mechanism.
– Databases distributed throughout the plant should have a common organization and format.
– The information system should provide adequate tools for data analysis, graphical display and report generation. This could include a detailed classification of age-related 'keywords' for data retrieval, and trend analysis tools.

Often the existing plant information systems do not meet all the needs of an efficient ageing management procedure. Data collection and record keeping has not been organized in an efficient way and the use of experience data is very laborious. The collection of information is not well co-ordinated, e.g. maintenance personnel may have a separate database for calibration or inspection data; and lots of information is still stored in paper archives. Failure and component coding and classification may not be suitable for all components and systems, and the coding is too general for detailed analyses. The older systems do not provide the means for displaying, graphically, component failure histories, nor for trending, which is important for quantitative ageing studies [2].

The quality of record keeping could often be improved. One way is to motivate the plant personnel to fill up more precisely the failure reports in order to avoid missing or erroneous failure descriptions. A more detailed reporting of, for instance, testing and calibration results would also be beneficial for ageing studies in many cases. Another way is to facilitate the accessibility of data by integration of information sources. The re-organization of the collection and storing of information and accounting for ageing management aspects can be done most conveniently at the time of upgrading or replacement of the plant data collection system.

In this paper, we approach the effective and proper utilization of plant specific information in ageing management from two points of view. On one hand, we want to promote the development of databases and data management systems for good record keeping and analyses. On the other hand, we want to point out the complexity of the degradation evaluation and the need of identification of major uncertainties in the analysis processes. Especially when there is a need to obtain more realistic - instead of over-conservative - estimates for remaining lifetime of components and structures, the identification and characterization of major uncertainties related to analyses becomes increasingly important.

First, we shall present our view upon the various information sources to be considered in ageing management processes. The ageing management covers not only the assessment of the technical lifetime of a component, system or structure, but also the evaluation of safety, economical, and other aspects important in the decision making. Ageing management is understood here in a wide sense, including also e.g. maintenance optimization and obsolescence. So we are not limited to the major components (pressure vessel, steam generators), but considering also replaceable equipment.

Second, our paper points out the importance of uncertainty analyses in connection with ageing management. We propose practices to improve the transparency of ageing analyses by documenting the major uncertainties. In a rational approach, the most important uncertainties should be identified and possibilities for uncertainty reduction should be evaluated in order to manage the uncertainties in a cost-effective way.

Finally, we propose practices and means to improve the availability and usability of plant specific information for ageing management purposes by describing an example of good practice in data collection and management. A database to integrate all important data, documents and models for ageing assessment of piping systems has been developed at VTT and TVO [3, 4]. When ready, this system will be composed of separate independent but interrelated databases for piping geometry, inspection results, material data, treatment of loads and analysis results as well as related documents. Various computational software packages for different aspects of lifetime estimation are connected to these databases through so called neutral files.

2. INFORMATION SOURCES AND EXPERT JUDGEMENT IN AGEING ASSESSMENT

A reliable assessment of ageing for systems, structures and components requires the combination of information from several sources. First, it should be assessed whether the available information is appropriate to be used in the specific application. Second, the weight of different evidence in the analysis process should be identified.

Figure 1 is a schematic presentation of the information used in the decision making with an emphasis on data needs and analyses for the technical life assessment of a component. On the way from the raw data to the final decision on ageing management there are several steps where engineering judgement is used or more sophisticated analyses are made. We are not considering here the process for prioritizing systems, structures and components for ageing analyses, but a description of such a process for safety important components can be found e.g. in IAEA's technical report 338 [5].

Usually the assessment of technical ageing of systems, structures and components is based on both plant specific and generic information. World-wide, generic experience from other plants is an important source of information, but it must be carefully evaluated to identify its applicability in plant specific ageing analyses. Another set of "external" information is experimental data from e.g. laboratory tests, test installations, etc. For instance material testing provides information on materials' behaviour in various conditions.

The plant specific information that can be used in ageing analyses comprises a large variety of data. We can mention design documents, inspection, monitoring and maintenance information, failure databases, plant operation data, modification records, etc. A common problem is that this information is usually spread in many different databases, and is partly in paper documents. Also the quality of plant specific information can vary a lot, as well as the quality requirements for different ageing assessments.

The analysis approaches for lifetime prediction depend on the equipment in question, and they can be categorized in many ways. For active mechanical equipment, e.g. pumps and valves, an ageing model based on the influence of all possible ageing mechanisms is often a too laborious task, since these components are so complex systems themselves. The rational (or only realistic) approach to analyze such components is often the statistical approach by identifying trends in failure occurrence, i.e. the use of reliability data instead of detailed phenomenological models. For passive components, e.g. piping, the approach is the structural analysis or modelling of a degradation phenomenon. The use of the analysis tools require usually good expertise, and lot of judgement is made during the analysis process, e.g. in the selection of model parameters.

After a technical judgement on ageing has been made, one must make the final decision on ageing management actions. In this decision making situation, additional information is needed. It is important to consider various criteria related to the decision, e.g. safety importance and costs of possible ageing prevention or mitigation solutions. For the safety assessment, the plant-specific probabilistic safety assessment (PSA) can be used. With an advanced PSA model, the effects of e.g. increasing failure rates due to component ageing can be simulated and the safety importance evaluated. However, the applicability of the PSA model should be judged case by case, and it should be taken into account in the decision-making.

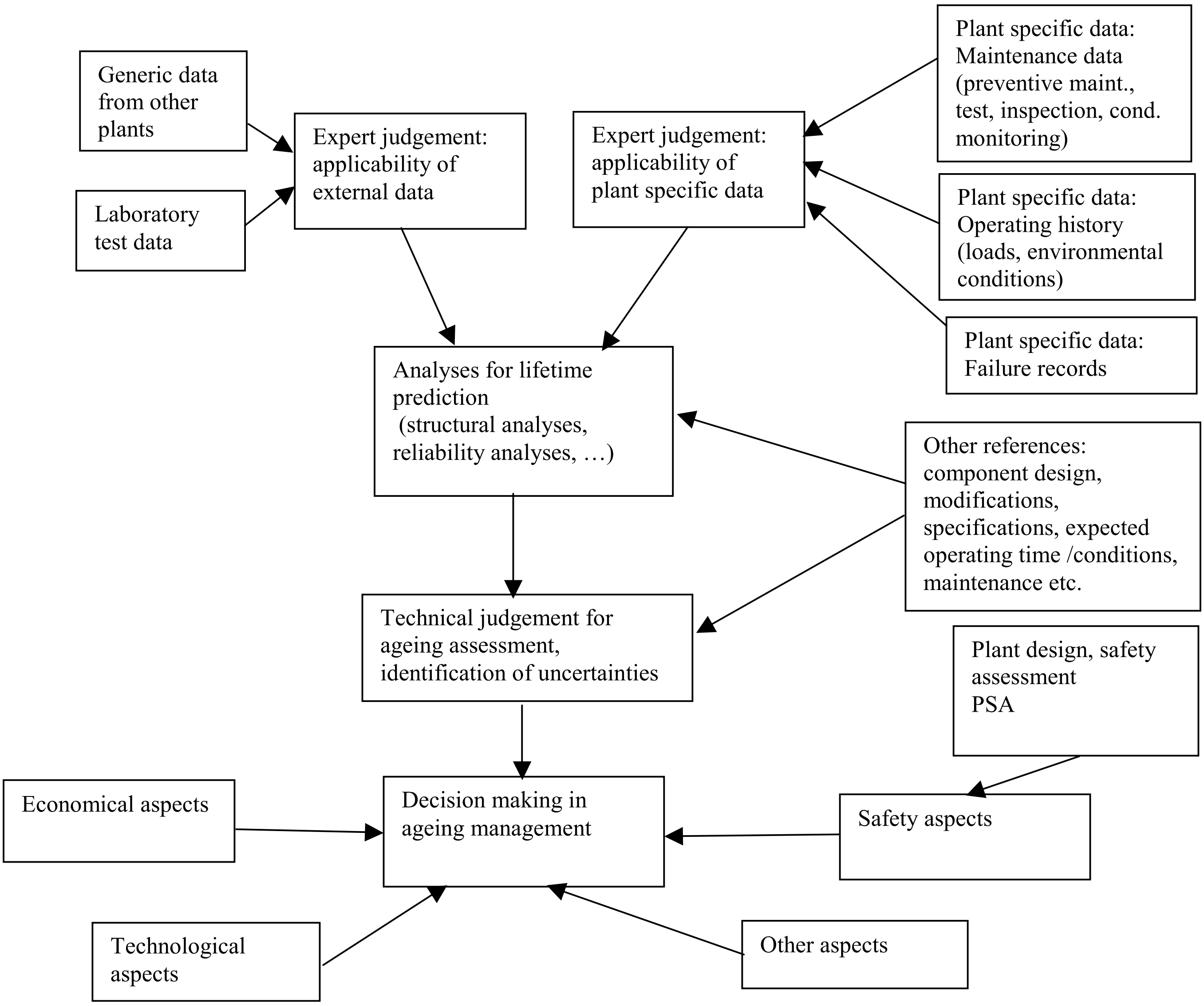

FIG. 1. Schematic presentation of the use of various information sources and expert judgement in ageing assessment.

3. UNCETAINTY MANAGEMENT IN AGEING STUDIES

Besides the quality and accessibility of raw data, the justification and documentation of the procedure for deriving residual lifetime distributions from the available data is important for the decision making in ageing management. The ageing assessment includes several uncertainties; some are related to the ageing phenomenon itself, some to the modelling of the ageing process.

The identification and interpretation of the nature of uncertainty is not always straightforward and may depend on the individual, e.g. the analyst, decision maker, or other observer. Thus the coverage in uncertainty analyses and transparency in documentation are important for the communication between various parties. A full analysis and propagation of uncertainties is a difficult and costly exercise, and thus it should be done only if it is relevant to the ageing management issue. However, the need for a qualitative uncertainty analysis should be emphasized, because with rather moderate expenses, the main uncertainties can be summarized, and such a summary can serve the decision maker in identification of needs for further analyses or in selecting other most cost-effective means for uncertainty reduction.

3.1. Types of uncertainties

Categorization of uncertainties may be helpful in decomposition of the problem and it may improve the transparency of the uncertainty analyses. The distinction of various types of uncertainty can be used in a decision making situation in order to identify the most suitable measures for uncertainty reduction and for determining the needs for additional evidence. Figure 2 adapted from [6] illustrates various uncertainties related to ageing prediction.

In phenomenological ageing models, all phenomena having impact on the component's or structure's behaviour may not be included in the model. This type of uncertainty is often referred to as *incompleteness*. It can be due to intentional decisions during the analysis planning: some things have been left out of the scope of the analysis on purpose. The reason for this kind of decisions is usually lack of resources. On the other hand, the incompleteness may be due to lack of knowledge about the ageing mechanisms. Incompleteness can be seen as a special form of *model uncertainty or model inadequacy.* Model uncertainties are often related to assumptions behind the model, level of detail, and scope or domain.

Parameter uncertainties refer to the unknown parameter values of valid models. This uncertainty is present as well in probability models (e.g. failure time distributions) as in deterministic or probabilistic phenomenological models. Parameter uncertainty has been traditionally taken into account in uncertainty propagation of models.

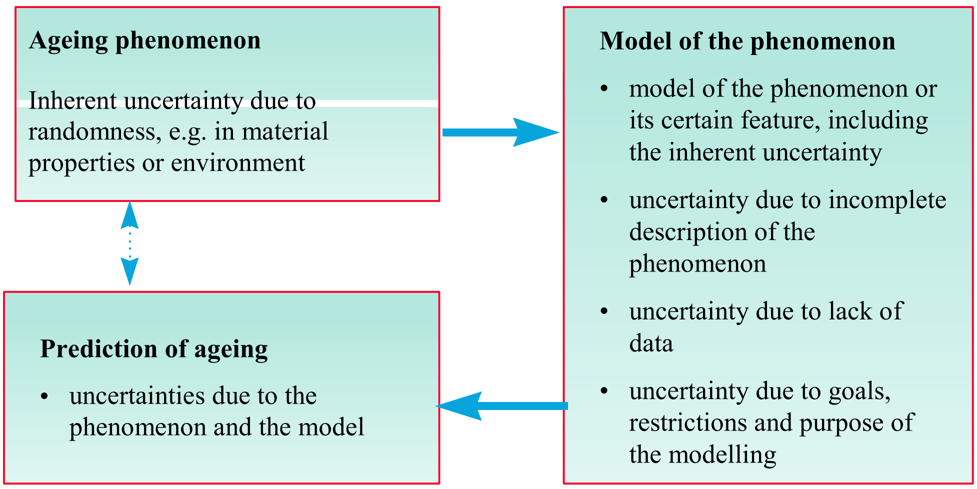

FIG. 2. Relationships between uncertainties in a phenomenological ageing model.

In addition to the classification of types of uncertainties to incompleteness, model and parameter uncertainties, a distinction in the nature of the *phenomenological uncertainty* may be made. One can speak about stochastic or *aleatory uncertainty* and knowledge or *epistemic uncertainty*. Aleatory uncertainty is sometimes called irreducible, since they cannot be made smaller without observing the real realization of the uncertain process. Epistemic uncertainty can be decreased by obtaining additional information or by making experiments. There are differing opinions whether such distinction can be made, but often they may have useful implications for the practice of modelling, e.g. in decomposition.

3.2. Documentation of uncertainties

During ageing analyses, these uncertainties are often implicitly identified by the analysts, but seldom documented in a brief and comprehensive manner. We suggest a simple approach for documenting the main uncertainties and limitations related to the analysis steps in order to improve the transparency of use of data and various analyses [6]. The approach should help the documentation of assumptions and uncertainties in ageing analyses, and thus improve the uncertainty communication and the identification of possibilities to uncertainty reduction.

The approach consists of formats or uncertainty documentation tables, in which the component and its ageing phenomenon are considered. The phenomenon is described and its significance for ageing management is evaluated qualitatively. The theoretical basis and the degree of validation of the models and tools are shortly described. Furthermore, the use and role of formal or informal expert judgement is explained, and the sensitivity and uncertainty analyses together with applied methodologies and main results made are presented. In addition to this general description each possible source of uncertainty is evaluated. In this connection both qualitative characterization and, if possible, the impact of uncertainty to the final results is evaluated in a (semi) quantitative way. In order to direct additional analyses, the possibilities to reduce the uncertainty are presented.

In the documentation format, the sources of uncertainties to be covered include:

– the inherent and knowledge uncertainties related to the phenomenon under analysis (e.g. randomness, turbulence, material properties)
– model uncertainties, including those originating from the scope of the analysis, incompleteness
– uncertainties due to input data
– uncertainties due to boundary conditions applied in the model
– uncertainties selection of initial states for calculations (e.g. initiating events, assumptions on the amount of certain substances in the system, the results from another model)
– uncertainties due to computational or numerical properties of the model (nodalization, time steps).

4. EXAMPLE OF DATA COLLECTION AND MANGEMENT: PIPELINE MONITORING SYSTEM

To make fitness, safety and lifetime related assessments for class 1 nuclear piping, the amount of necessary input data is considerable. Often it has to be collected in a very short time. At the same time it is essential that the data is reliable and up-to-date. This example outlines the contents of the database system, consisting of separate geometrical, material, loading and reference document databases, which is being developed by TVO and VTT [3,4] to facilitate the analyses of class 1 piping and generation of the associated documentation.

In existing power plants, the number of people responsible for load, structural and vibration related projects might be very limited. This means that tasks related to obtaining starting points, performing an analysis and preparing documentation might be the responsibility of just one person. This person will be asked questions like:

– We want to make a change, what are the implications?
– We had an abnormal event, what are the implications?

– We want to make a risk assessment, where do we get the starting points?
– We need up-to-date information to order a new valve, please supply?
– During the inspection we found a crack, can we run until next year's outage?

A lot of work and an adequate and up-to-date documentation is necessary to give the answer to these simple questions. The availability of necessary documentation may be a real problem for several reasons. Load and strength analyses for different systems have been done over tens of years time span. These analyses may have been performed and documented by different persons in different ways using different tools and have even been archived in different ways and at different locations. During the lifetime of the plant, there may have been a power uprate, major piping and equipment exchanges and modernization projects. The documentation may not have always been fully updated. Changes performed in the plant may not affect the as build structures, but may affect future changes. Some or all of the above reasons may have led to a difficult-to-use load and strength archive.

4.1. General information with regard to the system

The system presently under development is meant to contain all up-to-date information necessary to analyze and monitor piping systems for an existing and operating plant. For a start only the TVO OL1 and OL2 plant will be entered into the system. The system is basically an "as built" system and is not meant as a design tool although parameter studies should be possible. All data in the system will be accompanied by the necessary information with regard to dates, version and validity. It will be possible for instance to keep an "as-designed/standard" version and an "as-built/measured" version. Other versions could be kept as well.

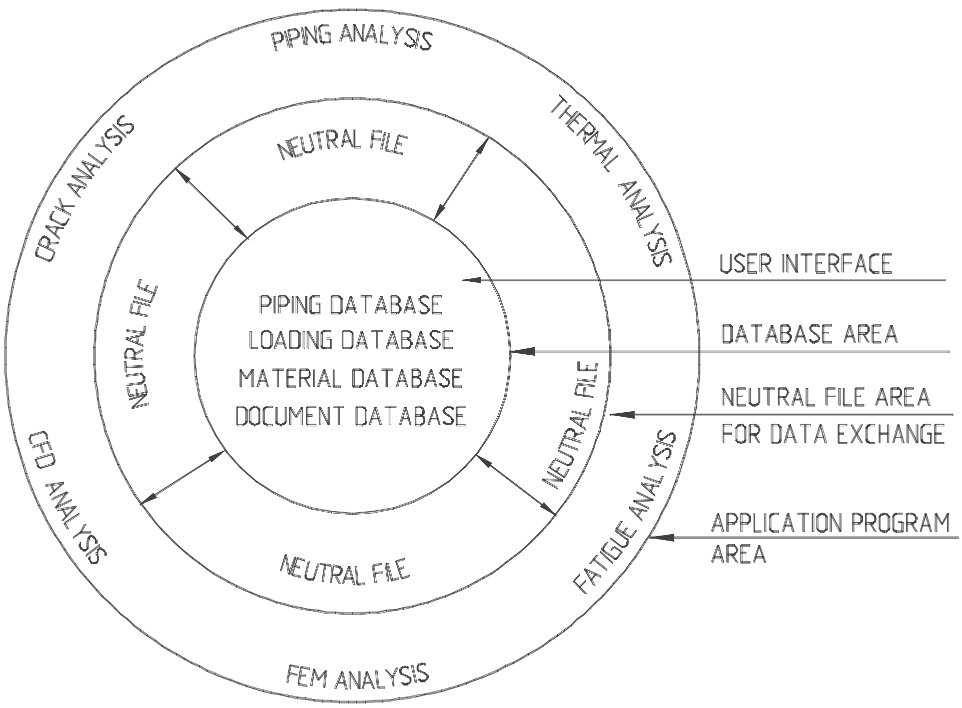

FIG.3. Structure of the pipeline analysis and monitoring system [3].

In case a load definition is changed, the system must "know" that the subsequent strength analysis and the associated results are not valid anymore. It is very important that subsequent analysis, like fatigue and fracture analysis, uses up-to-date input. In that way, the remaining lifetime can be estimated and the need for actions determined. And with a growing importance of parameter studies and probabilistic analysis it is more and more important to have the input data to the analyses in a flexible and readily available electronic format.

The system is built up of separate and stand-alone databases and program modules. Thus, different parts can be used for their own purpose without the whole system having to be completed or in use. Commercially available programs will be used as much as possible (database development, piping analysis, FEM, CFD, etc.). For special purposes, customized programs will be developed (crack growth, event monitoring etc.).

When ready, the system can be logically divided into three main areas: core, outer border and interface (see Fig. 3).

The core of the system consists of several interconnected databases and their user interfaces, and is called the database area. The contents of some of these databases are described in more detail in the next section. The different databases are edited with their own user interfaces that are designed to give visual information alongside alphanumerical. According to the present plans, control and navigation is, as far as possible, handled from this level. For the moment the database area contains the following databases:

- The piping database, containing information like geometry, material, contents, isolation, loading, boundary conditions, detected cracks etc.
- The material database, containing all information with regard to the materials referred to in the piping database. These properties may be standard properties or measured ones.
- The loading database containing all information with regard to loads, loading combinations, ASME design and service limits, design events and occurred events. This database is actually the most complex part of the system.
- The result database containing all significant information with regard to the analysis results like stresses, displacements etc. The presence of the analysis results in the database gives the possibility to perform subsequent analysis without first having to go through the stress analysis.
- The report database containing the documentation that is related to the previous items. Input made to and documentation produced by the technical databases will refer to the relevant documents from this database.

At the outer border of the system are the application programs. This is called the application program area. As far as possible these programs shall use data from the databases and run in batch mode. However, if necessary, data from external sources can be used. This may be the case when some new data for the database is obtained with special purpose programs. Basically, the application programs can be of two types: commercially available analysis programs to perform structural, flow, thermal, fatigue, fracture mechanical and/or other analyses or tailor made analysis modules to perform post processing of previously obtained results, event monitoring, fatigue monitoring, crack growth monitoring, definition of inspection intervals etc.

“Neutral” files are used as interface between the databases and the application programs. A neutral file is typically a batch-input file to control the flow of the analysis modules and to supply the input for the subsequent analysis. This means that an interface module is necessary to write the necessary data from the database into the neutral file and in the right format. In order to be independent of program developers a user definable interface module was developed. This module enables the engineer that is responsible for the subsequent analysis to define the neutral file.

Although information is divided over several databases, one of the main principles in this project is that no information that may be used as input to an analysis is allowed to occur more than once. Another main principle is that all data shall be accompanied by a source reference, a date and a validity indication.

4.2. Elements of the database system

Piping database

The piping database consists of the piping geometry and all other information necessary to perform analyses. Therefore, it also contains information on welds and equipment, boundary conditions and the materials of, in and around the piping. The organization of piping geometry in the database is similar to the organization of piping systems and related drawings at TVO. This organization is as follows:

- At the first level, the system is found with the system identification number. Examples of systems are the feed water system (system 312) and the relief system (system 314). Drawings at this level are called system isometrics.
- At the second level, the main parts of the system are found. The feed water system for instance is divided into parts called 312 BAA-1, 312 BAA-2, 312 BCA-1 and 312 BCA-2. There are no separate drawing series at this level.
- At the third level, the piping geometry is divided into isometrics and associated part lists. At this level, the drawings have a part name followed by a sequential number, like 312 BCA-2-1, 312 BCA-2-2 and 312 BCA-2-3. This is the lowest and most detailed level of piping drawings available at TVO. There is one input table for every isometric.

The above choices were made to make organization of the database easy and recognizable for all possible users at TVO. Furthermore, possible future electronic drawings will be organized in the same way and this will ease the data exchange between the systems.

Further division was made according to normal FEM convention. This means that the database contains nodes and elements with all sorts of associated properties. In this context "elements" refer to geometrical elements like straight piping parts or pipe bends and "nodes" to the points connecting these elements. This means that a new element starts whenever there is a change of any of the element properties. Separate nodes will also be appointed to welds or nodes that shall be analyzed later on. Taking these rules into account is the work of the person entering data into the system and is very important when designing and building the database model.

However, the database model is not meant to be equal to the associated finite element model. It is meant to contain an as-build representation of the actual geometry inclusive main equipment, see also Fig. 4. This means that for instance additional nodes, necessary to perform a sound dynamic analysis will not be added in the database building stage. For this case the FPIPE program [7] will be further developed. It will be enhanced with routines to automatically change the model and, by the use of an iterative solution method, come to an optimum solution.

Examples of information associated with a node are:

– The node number, co-ordinates in the plant coordinate system and isometric number (document database reference inherently including version number and date)
– Node element information like mass and associated stiffness and center of gravity
– Support information like stiffness, pipe whip restraint, stiffness matrix, gap or damping
– Reference to weld drawings inclusive weld (repair) information and dates
– Information as to what type of analysis shall be performed at the node (stress check, crack growth, fatigue etc.)

Examples of information associated with an element are:

– The element number, the nodes at the end of the elements and the isometric(s) to which the element belongs. As the isometrics can be found from the document database only a link to this database will be made.
– Cross sectional information like material designation, diameter and thickness of the pipe, the content designation and the isolation material designation. It should be noted that the designation of the material, contents and isolation is not more than a link to the material database. In this way the one of the most important rules of database design, namely "no data shall occur more than once in the database system", is again fulfilled.
– The element type information like straight pipe, bend, T-type , expansion joint etc.
– Even very specific information like detected or postulated cracks can be entered into the database, see Fig. 5. In this way the system can also be used to perform bookkeeping of all the findings made during the inspections. Furthermore, it will be immediately available to perform subsequent analysis. During the, nowadays very short, outage, speed of analysis and adequate documentation is of great importance. As all the related starting points for such a subsequent analysis will be in the system the analysis should in fact not be more than a press-on-the-button. This is however still a vision for the future.

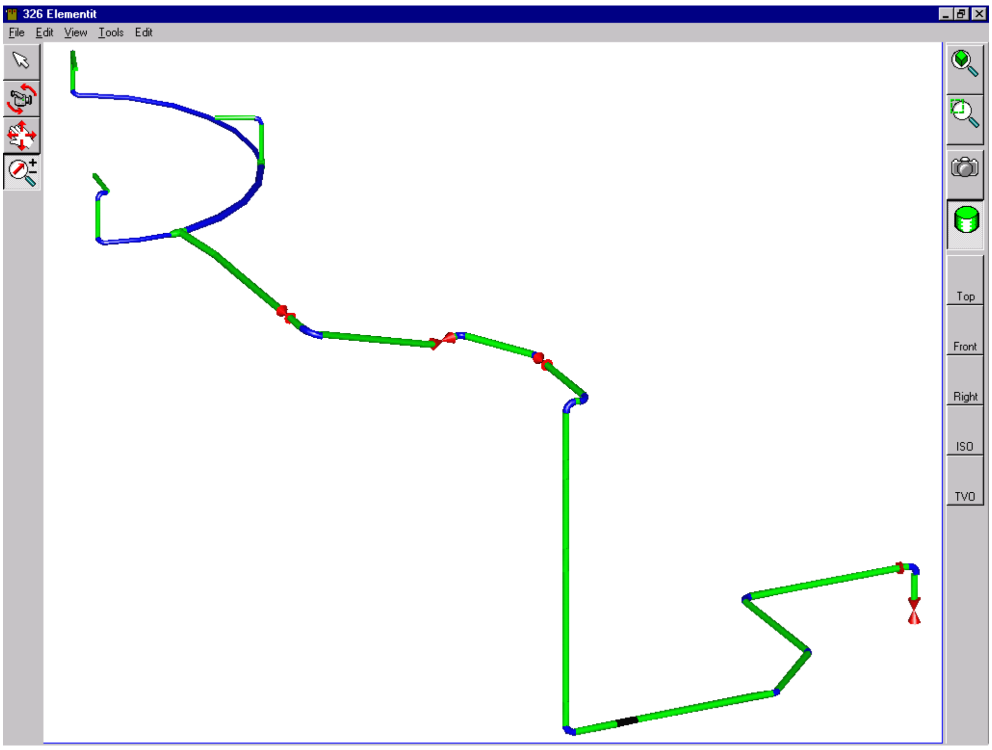

FIG. 4. Visualization of piping using the piping database user interface [3].

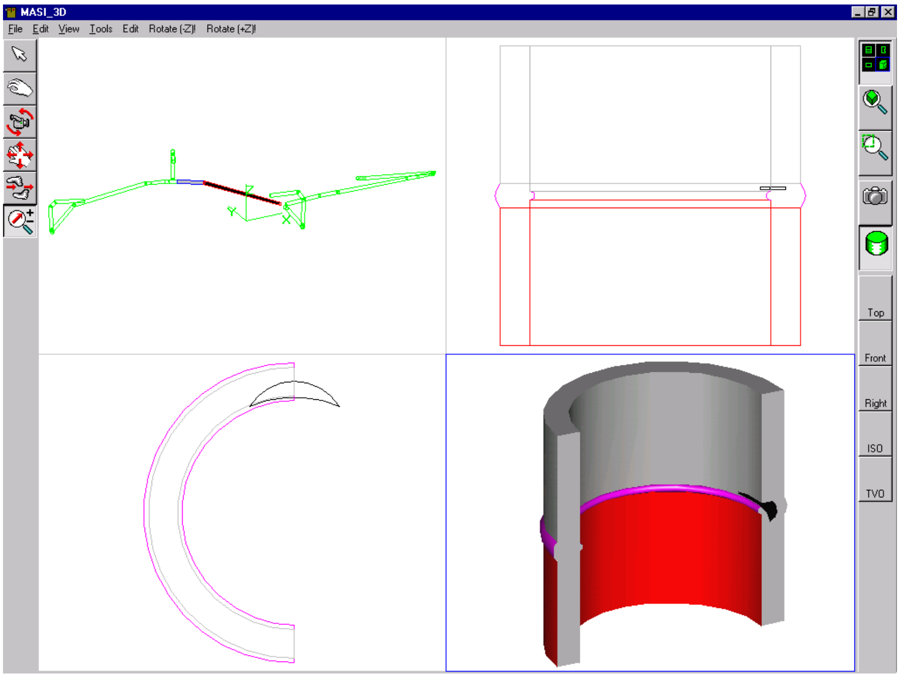

FIG. 5. Visualization of a crack [3].

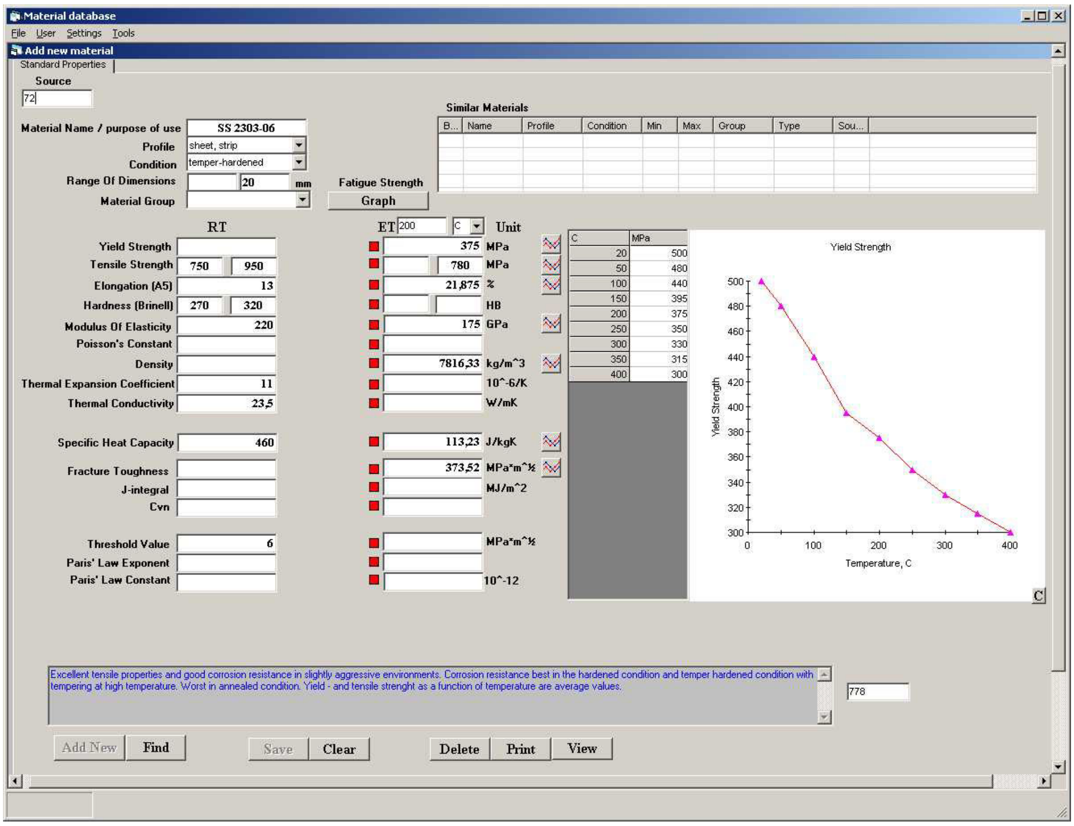

FIG. 6. A typical view of the material database program interface display [3].

The material database MATDBS

The material properties of the normally used materials at TVO are gathered in a database called MATDBS. A typical view of the program interface display is shown in Fig. 6. As can be seen, many different material properties, if necessary as function of a second variable, can be saved. In addition, the source of the information is input as a link to the document database.

Note that the material identification, the content identification and the isolation identification have to be chosen from the TVO material database and as such cannot be freely chosen. Either an existing material has to be chosen or a new material has to be first introduced into the material database. This in fact shows another basic principle in the database development project where data may only occur once and users are helped/forced to use sound data. A possibility will be incorporated to calculate either with standard values or with alternative ones, for instance measured values.

The loading database

The loads, combinations, events and everything else related to it will as far as reasonable be saved in a loading database (Fig. 7). This database is described in more detail in [4]. The load database is designed to:

– Contain and document the actually valid design load specification inclusive service limits
– Act as an input database to perform stress, flexibility, fatigue and/or crack analyses
– Monitor and document the annual cumulative thermal transient and other fatigue related events
– Perform book-keeping of the load-cases and -combinations that are valid at a time and contain the connection between old and new data
– Give the structure for the result database where the significant results of analyses are stored.

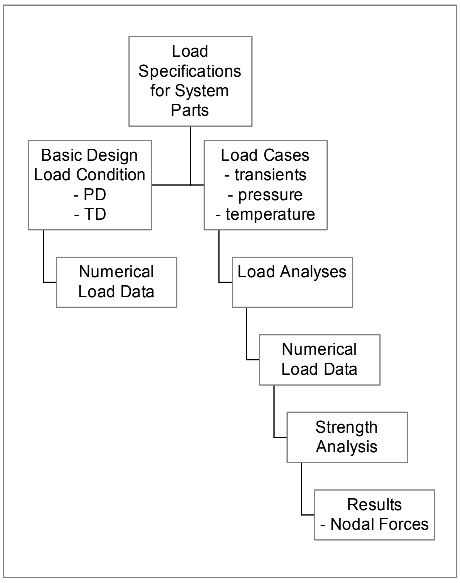

FIG. 7. Load specifications for piping sections and system parts, the hierarchy of load cases, load analyses, load data, strength analysis, and nodal results from the strength analysis [4].

The document database

In order to easily find documents, a document database was developed by TVO (see Fig. 8). All documents that are related to any of the items within the TVO pipeline analysis and monitoring system will be gathered into this database as will all other documents that are related to strength and vibration. Once a document is part of the document database, it can be logically associated to any of the other databases. For instance, a load analysis report can be coupled to the load of the system (part) that it is related to. Also input data that was retrieved from an isometric can be coupled to the applicable revision of the isometric. In this way data in the database will always be accompanied by an exact trace to the data source. In case the database document is available in electronic format even this file can be coupled to the document database and will thus be readily available from the PC. An extra option in the document database is that activities and deadlines may be associated to the documents and reports can be produced showing future activities and deadlines.

After making the associations that are described before it is possible to automatically add information with regard to the data sources to the documentation that is produced by the system. For instance the crack growth analysis that is mentioned later on is performed as a batch analysis and will automatically generate a complete report.

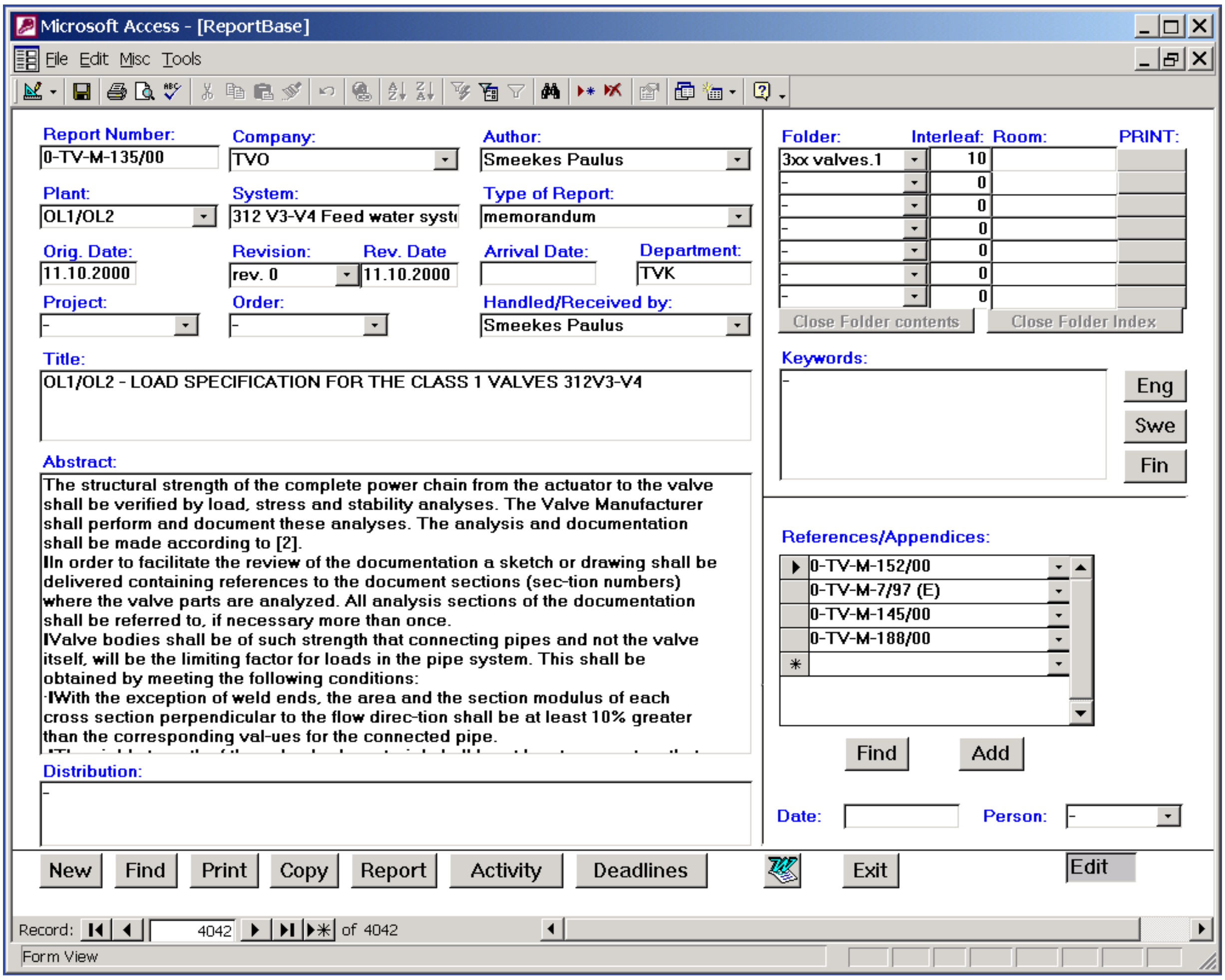

FIG. 8. An example of the main form of the document database.

The result database

The significant analysis results will be saved into the result database. Due to the fact that results are computed for loads the organization of the result database will be similar to that of the loading database. The development of the result database has not yet been started, but is "next on the list".

4.3. Analyses and application programs

The various application programs connected to the database system, as well as their interrelations and connections to the databases are depicted in Fig. 9.

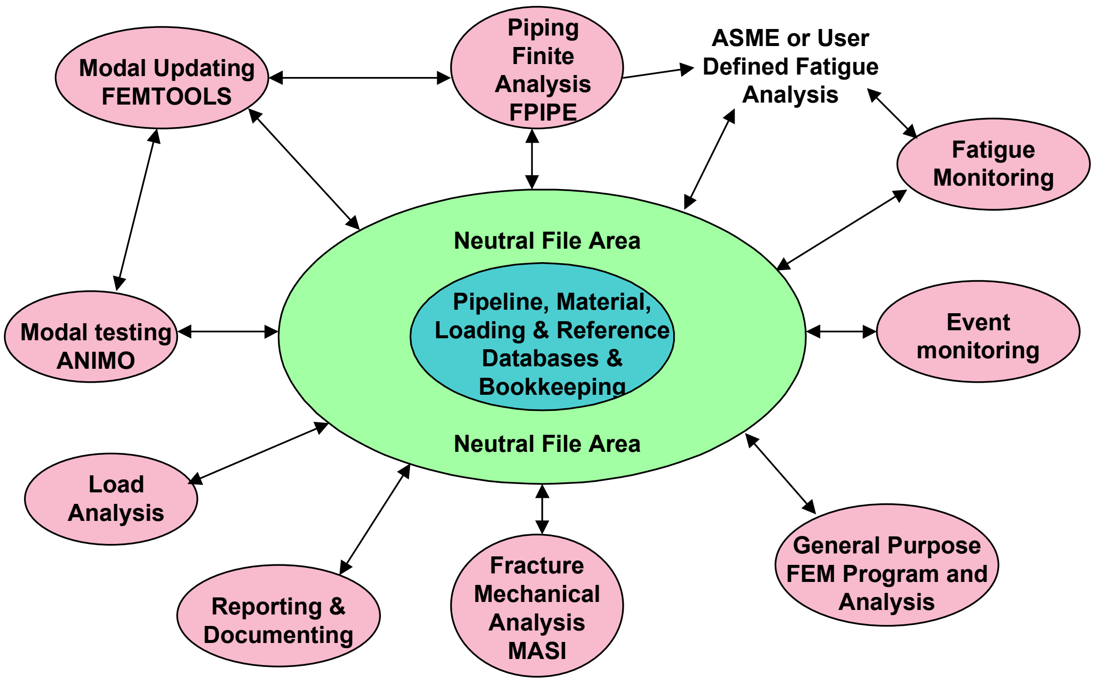

FIG. 9. Application programs connected to the database system.

Piping strength analysis

The piping strength analysis will be carried out with a commercially available piping analysis program. For the moment FPIPE [7] is chosen. It shall use the geometry, material properties and the loading as described in section 4.2. Compared to a basic piping calculation program, some additional features may be necessary:

Loading for these analysis may be temperature, mechanical or of any other nature. For the moment, two analysis types, which may be either static or dynamic, will still be handled separately. The results of linear elastic analysis for load cases or combinations will be used for documentation, post processing or written back into the result database for later post processing. In case the results are written back into the database, the structure of the result database will be very similar to the structure of the load database. In the case of non-linear analysis for load cases or combinations the results will typically be used for immediate documentation and post processing.

Fatigue analysis

A fatigue analysis may be done according to the ASME, the materials' Wöhler diagram or any other method. The system contains a program that performs fatigue analysis according to the ASME III –standard [8] for class 1 piping. The input for this program shall be taken from the input data and results of strength analysis programs. This analysis shall be performed for materials like ferritic and austenitic steel and INCONEL. Events causing loading shall be taken from either the design event database or from the event counter database. The strength analysis results for all locations analyzed must be available to perform other analysis like for instance crack sensitivity analysis. In the design phase of the program the feed water system and two directly connected systems, systems 312/321/327, will be used as a pilot analysis entity.

Fracture analysis

When performing fracture analysis, several crack growth mechanisms have to be considered, like crack growth due to cyclic mechanical loading or IGSCC. As these mechanisms are dependent upon the material and the environment these method(s) can be chosen automatically. These analyses may be performed according to the simplified ASME method, using the VTT developed MASI-PIPE program [9] or other programs using batch type input.

A conservative way to estimate the usage status of the piping may be made through either the assumption of a postulated initial crack equal to maximum non-detectable crack at the least favorable location or the worst detected crack. This crack shall, while including an appropriate safety factor, not grow to a critical crack during lifetime or until the next planned repair outage. These cracks are obviously assumed to grow from the last periodical check. The crack growth is estimated using the actual events at the station. Thus the worst possible crack growth can be predicted and necessary actions can be specified in due time.

Organization of data exchange

The data exchange between the databases and the application programs will be done via neutral files. For the time being the neutral files are equal to the batch input and standard output files of the application programs. So they are not yet true neutral files. As this project is a stand-alone project, it is not worthwhile to develop true neutral files. It is however important to follow the international development in this field. True neutral files would be a huge step forward as the same file could be used to perform analysis with different application programs and again the results would be readable with only one tool.

4.4. Bookkeeping and validation

As the database will be quite complex, a good design and bookkeeping is very important. Records shall be kept for piping, equipment and other significant parts. The records shall contain such information as date of installation and possible exchange, as-build geometry and properties, welding, inspection and repair. Also the validity of the data shall be indicated. Thus analysis can be performed based on reliable and up-to-date information.

All information comprised in the database shall be accompanied by significant information related to date of installation and reference documents. The date is important as for instance thermal cyclic loading that has occurred before a part was replaced shall be ignored with regard to the fatigue of the replaced part. Reference documents are important, as, in order to be significant, input data to analyses shall be traceable. Reports that are produced with help of the database shall contain references to the source of the information contained.

5. CONCLUSIONS

The role of up-to-date plant specific data in ageing management is crucial. Reliable estimates on the remaining lifetime of specific components can be made only by using relevant data. The diversity of necessary data types and the huge amount of data - especially when also monitoring results are utilized - set high demands on the management of data. Integration of data sources and record keeping on modifications become an essential part of effective data management. At a later stage these features will be realized in the data base system which is being developed for the ageing management of the primary piping systems at OL1 and OL2 plants. The computational methods and models connected to the system can easily be verified by the related documents.

Besides the improvements in plant specific data collection and management practices, we emphasize a rational analysis of major uncertainties in ageing analyses. There is a recognized need to move from deterministic analyses towards probabilistic life assessment methods to reduce the over-conservatism and to enable risk-informed decision making. In connection with probabilistic ageing analyses, the identification and quantification of uncertainties plays an important role. The classification of uncertainties is needed in the decomposition of the problem and it helps in the identification of means for uncertainty reduction. A broad qualitative uncertainty analysis forms a basis to determine requirements for quantitative uncertainty analyses. Further, an enhanced documentation serves to improve the uncertainty communication between different experts and the decision-maker, and between utilities and safety authorities.

ACKNOWLEDGEMENTS

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REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Data Collection and Record Keeping for the Management of Nuclear Power Plant Ageing, A Safety Practice, Safety Series No. 50-P-3, IAEA, Vienna (1991).

[2] SIMOLA, K., Reliability Methods in Nuclear Power Plant Ageing Management. VTT Publications : 379. Technical Research Centre of Finland, Espoo, Finland (1999).

[3] SMEEKES, P., LIPPONEN, A., RAIKO, H. & TALJA, H., "The TVO pipeline analysis and monitoring system", SMiRT 16, Paper 1868.

[4] RAIKO, H., LIPPONEN, A., SMEEKES, P. & TALJA, H., "Load-case and –combination database", SMiRT 16, Paper 1869.

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Methodology for the Management of Ageing of Nuclear Power Plant Components Important to Safety, Technical Report Series No. 338, IAEA, Vienna (1992).

[6] PULKKINEN, U. & SIMOLA, K., Identification and Communication of Uncertainties of Phenomenological Models in PSA, Rep. NKS-45, Nordic Nuclear Safety Research, Roskilde, Denmark (2001).
[7] FPIPE, a finite element method (FEM) based piping analysis program developed at FEMdata Oy, Finland
[8] ASME BOILER AND PRESSURE VESSEL CODE, SECTION III, Nuclear Power Plant Components, Division 1, Subsection NB, Class 1 Components.
[9] MASI-PIPE, a fracture mechanics analysis program to make quick crack growth and residual lifetime assessments for cracked piping, developed by VTT Manufacturing Technology, Finland

FANP CONCEPT FOR PLANT LIFE MANAGEMENT AND RECENT EXPERIENCE

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Abstract

The deregulation of the power generation industry has resulted in increased competitive pressure and is forcing operators to improve plant operating economy while maintaining high levels of plant safety. A key factor to meet this challenge is to apply a comprehensive plant life management (PLIM) approach. The PLIM strategy addresses relevant ageing and degradation mechanisms, the safety concept and availability requirements. In addition, it affects the management of plant personnel, consumables, operations management systems, administrative control procedures and component documentation.

Framatome ANP GmbH has developed an integrated PLIM concept and associated software tools applicable for both new and operating plants. The concept includes procedures and strategies regarding mechanical, electrical and I&C components as well as civil structures.

The majority of components and structures in a well-kept power plant will experience a technical service life, which is far above the intended design life. In most cases, only a small percentage of mechanical components is subject to significant degradation which may effect the integrity or the function of the component. The intention of an effective PLIM concept is to focus maintenance activities on safety and availability relevant components, were a degradation potential exists. The PLIM concept utilizes a combination of strategies to select components in a power plant which are relevant to life management and to identify surveillance methods to be applied:

- An integrated safety review identifies components essential to safety, providing a classification of the associated safety levels
- Assessment concerning the availability relevance of components is conduced
- Components identified to be important to safety and availability are subject to a screening process for further grouping with respect to their degradation potential
- For components with an existing degradation potential, cost effective measures are provided for specific degradation mechanism identified.

The selection process provides reasonable prioritization of ageing relevant components and ensures that efforts are devoted to elements, where ageing is a relevant concern. For the selected components special life management (LM) strategies are established to efficiently monitor the as-is condition. In order to efficiently support the PLIM processes, advanced IT tools have been developed.

1. INTRODUCTION, CONCEPT FOR PLANT LIFE MANAGEMENT (PLIM)

The purpose of a systematic ageing and plant life management strategy is to allow the lifetime of plant components to be projected and to indicate when a component has reached the end of its effective lifetime before it fails. The systematic application of such a strategy results in an increase of power plant availability and enables the implementation of a targeted maintenance strategy in terms of its economic and technical effect.

Framatome ANP GmbH has developed an integrated PLIM concept applicable for both new and operating plants and addressing all ageing phenomena regarding plant safety concept, mechanical, electrical and I&C components as well as civil structures, plant documentation, plant personnel, operations management system (OMS), consumables and administrative controls.

The PLIM concept utilizes a combination of strategies to select systems or elements ("components") in a power plant which are relevant to life management and to identify surveillance methods to be applied:

- An integrated safety review identifies components essential to safety, providing a classification of the associated safety levels (see Chap. 2.1)
- Assessment concerning the availability relevance of components is conduced (see Chap. 2.2)
- Components identified to be important to safety and availability are subject to a screening process for further grouping with respect to their degradation potential (see Chap. 2.3)
- For components with an existing degradation potential, cost effective measures are provided for specific degradation mechanism identified (see Chap. 3).

The selection process provides reasonable prioritization of ageing relevant components and ensures that efforts are devoted to components, where ageing is a relevant concern. For the selected components special life management (LM) strategies are established to efficiently monitor the as-is condition, see Fig. 1.

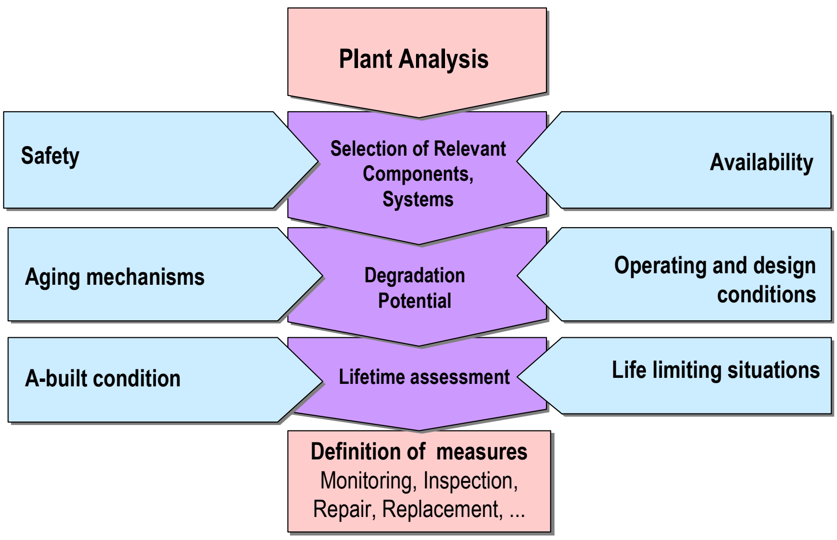

FIG. 1. Framatome ANP plant life management concept.

An example for a plant-wide application of the PLIM strategy covering a variety of screening and assessment methods applied for mechanical, electrical and I&C components as well as civil structures is described in Chap. 4.

Activities in the field of plant life management generally require the systematic processing of large amounts of data, which has to be continuously updated to describe the current ageing status of the plant. With the use of specifically tailored software tools for PLIM, this process can be efficiently streamlined. For this purpose the PLIM-Software family has been developed by Framatome ANP to provide software tools for ageing and plant life management (PLIM) and plant life extension (PLEX) activities on the fields of mechanical components, electrical equipment and civil structures (see Chap. 5).

2. SELECTION OF RELEVANT COMPONENTS

A power plant consists of a great number of systems and components. In order to streamline the plant life management process, activities have to be focused on a limited number of ageing relevant elements. In order to select components relevant to plant life (components for LM), an integrated approach is applied which considers the

1. safety relevance
2. relevance to plant availability
3. and the existence of a degradation potential

of plant components, see Fig. 2.

2.1. Safety relevance

The safety concept of a plant is specifically assessed during the periodic safety review (PSR) which commonly is to be performed every 10 years. Within the PSR the main tools for assessing the safety concept are

- deterministic safety status analysis (SSA),
- probabilistic safety analysis (PSA), and
- evaluation of operational experience.

System functions and associated components/structures important to safety are identified on the basis of stipulations in the licensing procedure. The safety status analysis provides precise information on the relative importance of safety functions with respect to preventing or controlling accidents. Depending on importance to safety, a categorization is performed.

Another option, in case a PSR does not exist, is to utilize an existing safety classification system provided by the utility. Framatome ANP also provides the service to perform PSR according to international standards which has been performed already for different plant types.

2.2. Relevance to plant availability

System functions and components/structures important for economic reasons - particularly for plant availability or due to high costs for replacement - are identified on the basis of operating experience, reliability analyses and expert judgement. A categorization depending on the importance to availability and component/structure costs is to be performed.

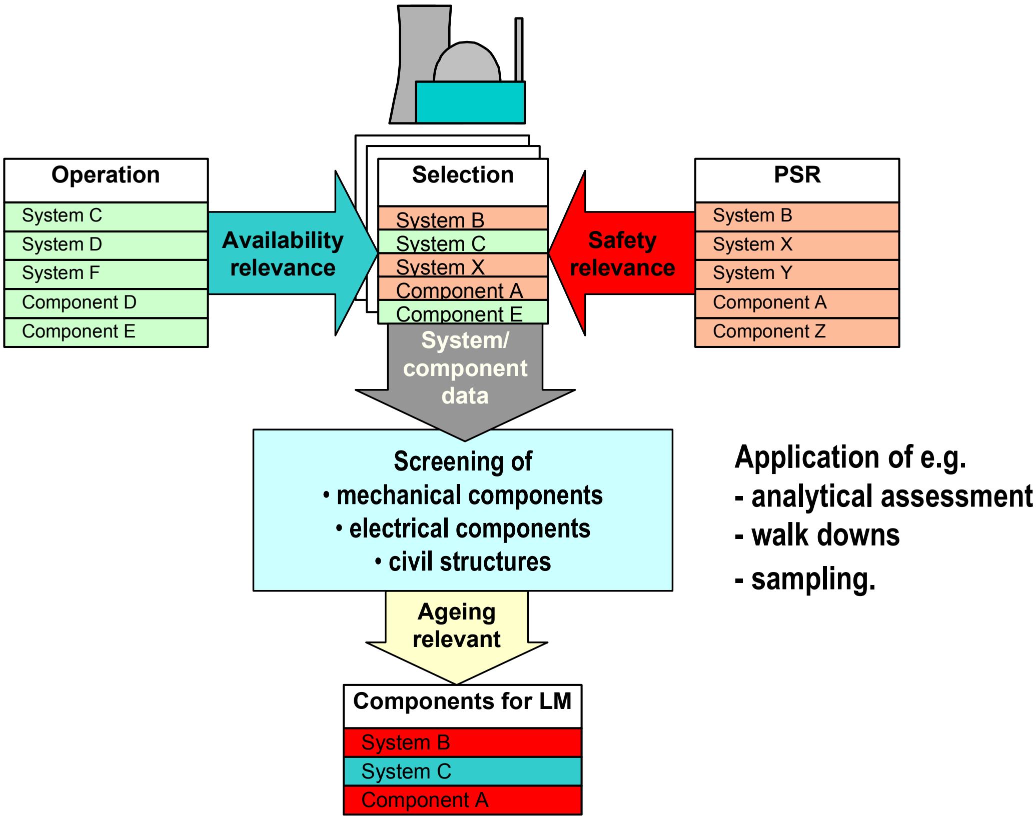

FIG. 2. Selection of relevant components for LM.

2.3. Degradation potential

In the next step an evaluation is performed for safety and availability relevant plant components in order to identify potential degradation mechanisms, which may affect the function or integrity of components concerned. This step requires a detailed understanding of the relevant degradation mechanisms to be expected for specific components. A screening process is applied to reliably identify the degradation mechanism to be considered in each case. Depending on the type of component in question - mechanical components, electrical equipment or civil structures – methods applied may cover analytical assessment, walk-downs by experts or sampling of materials.

In order to efficiently screen the e.g. mechanical components of a plant unit for systems potentially affected by degradation mechanisms, operating and design data has to be systematically accumulated on a cross-system basis. It requires the knowledge of system operation conditions (e.g. temperature, pressure, flow, steam quality), water chemical conditions (e.g. pH value, oxygen concentration) and material properties (e.g. alloy content, allowable stress, yield stress, toughness).

For screening the mechanical components of a plant unit, the Framatome ANP software can be utilized. Based on the heat balance diagram data the program establishes a basic virtual power plant model and allows for an analysis of the water chemistry cycle to be conducted based on the thermal-hydraulic parameters (see Fig. 3). Taking into consideration the representative materials used in each case, the system areas are then studied with respect to the potential risk posed by degradation mechanisms.

The result of this study provides a matrix indicating for which power plant systems a degradation potential exists. The IT-supported screening process guarantees an economical state-of-the-art application strategy. Based on degradation priorities and results from safety and availability considerations, systems, which require further assessment, can be identified. System with no degradation potential existing, can be disregarded for further evaluations.

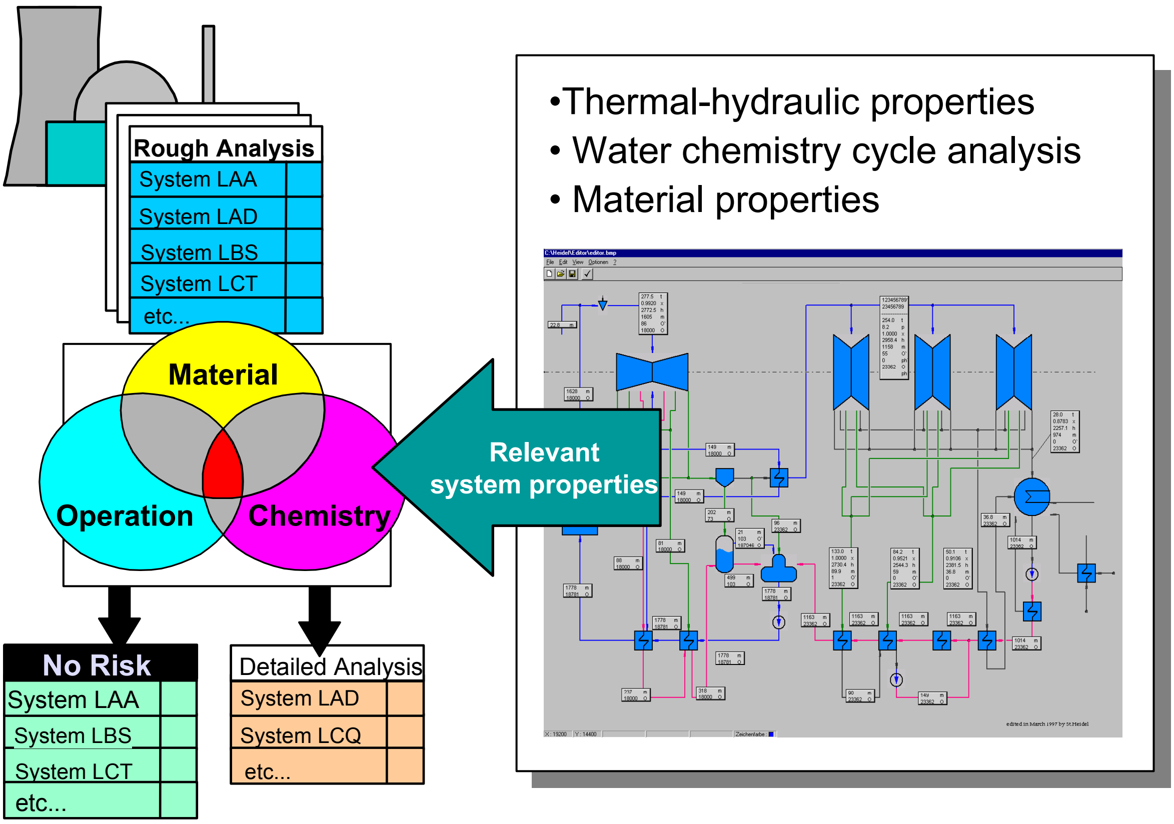

FIG. 3. Plant screening for degradation potential.

Depending on the type of degradation mechanisms identified, specific countermeasures against degradation may be considered. Depending on the type of degradation and the specified category of safety and availability requirements, a graded cost effective lifetime management (LM) strategy for components/structures can be adopted.

3. THE LM PROCESS FOR SELECTED COMPONENTS

For safety and availability relevant components, where a degradation potential exists, the progress of degradation may have to be observed in order to avoid loss of integrity or function. Before conducing specific measures, assessment should be performed to quantify the degradation behavior of the component considered. However, unfortunately not all degradation mechanisms experienced in power plants can be quantified based on available data.

Depending on the nature of the degradation mechanism to be expected and the state-of-the-art of science and technology, the following categories can be established:

1. **Quantifiable degradation mechanism** - the progress of degradation can be quantified and reasonably predicted
2. **Qualifiable degradation mechanism** – the root cause is generally known and accepted, however the progress of degradation can not be reliably quantified for the individual component
3. **Unpredictable degradation** – degradation is of stochastic nature and is initiated by specific local conditions beyond the control of the plant operator.

For degradation mechanism of category 1 an deterministic analysis method is to be applied to effectively predict the current degree of degradation experienced by the component with respect to life limiting situations. Life limiting situations can be deduced from the design information of the system or component providing component integrity criteria or specific functional requirements. Base on these deterministic lifetime predictions, maintenance management and plant availability can be optimized. Hence the maintenance activities can be effectively streamlined depending on the predicted progress of degradation in respect to life limiting situations (see Chap. 3.1).

For degradation mechanism included with category 2 an assessment method is to be applied to effectively detect the current degree of degradation experienced by the component with respect to life limiting situations. Surveillance methods to be applied depend on the particular component, the location of the component where degradation is expected and the material used (see Chap. 3.2).

For degradation mechanism category 3 surveillance methods can be engaged which ether monitor the environment in order to avoid hazardous environmental conditions or which provide an early warning before component failure (see Chap. 3.3). This type of monitoring may be advisable, in case the component concerned is not designed with sufficient redundancy and the loss of function would compromise on safety aspects.

The above categorization requires analytical assessment and experts knowledge in combination with walk-downs, monitoring and inspection activities. Inspection techniques can be e.g. ultrasonic testing of the wall thickness, ultrasonic flaw detection, visual inspections. There also are a variety of monitoring methods and remedial activities that may be applied.

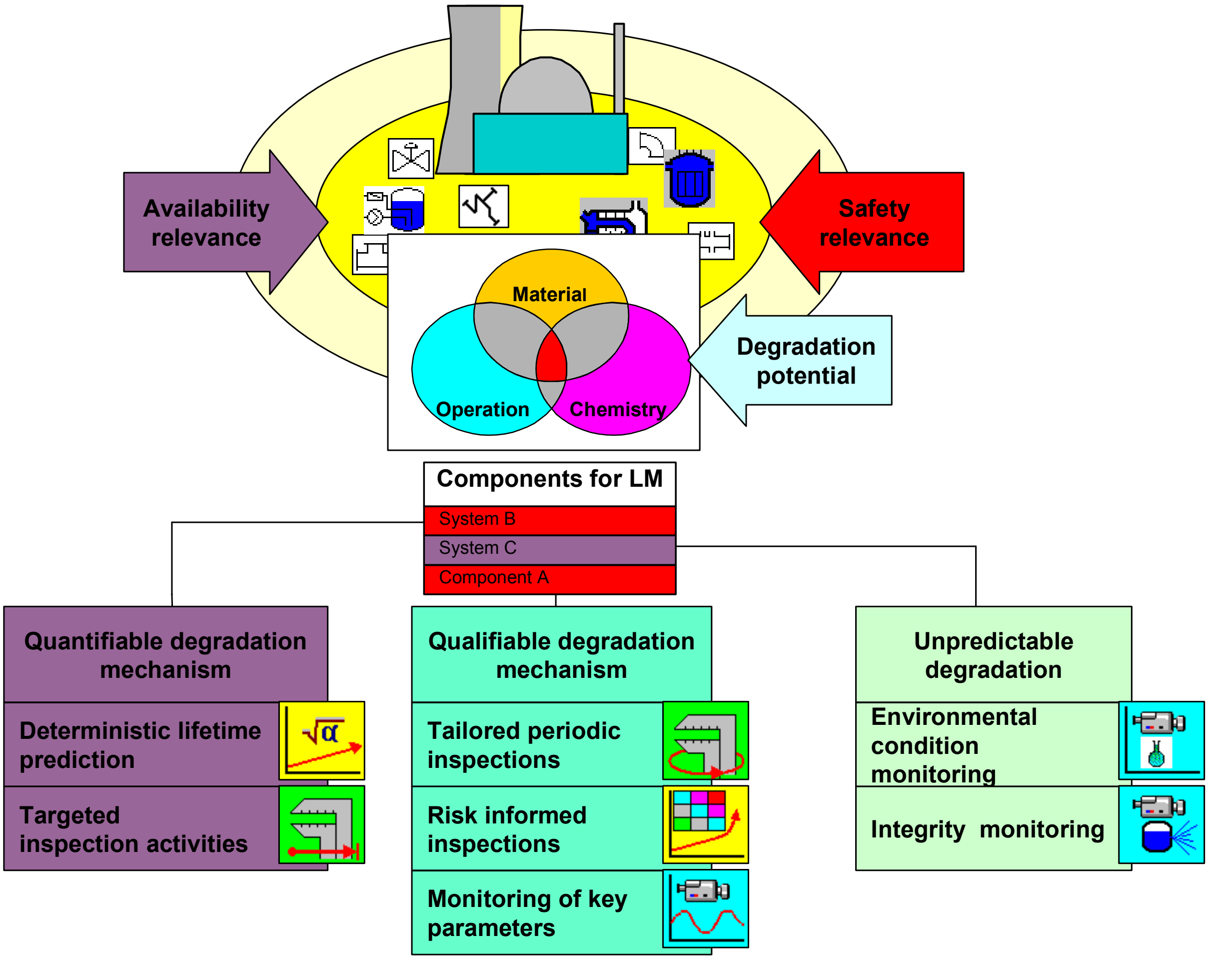

FIG. 4. Surveillance measures for mechanical components (LM components).

For components were none of the relevant degradation mechanisms apply, LM activities need not to be applied. For components where a degradation potential exists due to different degradation mechanisms, the most relevant degradation mechanism should be prioritized.

3.1. Quantifiable degradation mechanism

If the progress of degradation for a component is deterministically predictable within reasonable margins, an approach using condition oriented inspection has proven to be the most cost-effective LM strategy to be applied.

For e.g. mechanical components the Framatome software system provides service life predictions as a key function for the relevant of predictable degradation mechanisms: (e.g. strain-induced cracking, material fatigue, flow-accelerated corrosion, cavitation erosion, droplet impingement erosion). Based on design requirements the boundary conditions for life limiting situations are determined (e.g. minimum toughness or wall thickness). On the basis of service life predictions the maintenance management and plant availability can be optimized and the service life capacity can be fully utilized.

The experience shows, that the majority of mechanical components is designed to experience a technical service life, which is far above the intended design life. In most cases, only a small number of components show integrity relevant degradation. For those components targeted maintenance and inspection activities can be explicitly scheduled on a long-term basis. An efficient service life management program builds on these degradation predictions, which are validated and optimized through the performance of a minimized number of examinations at critical points.

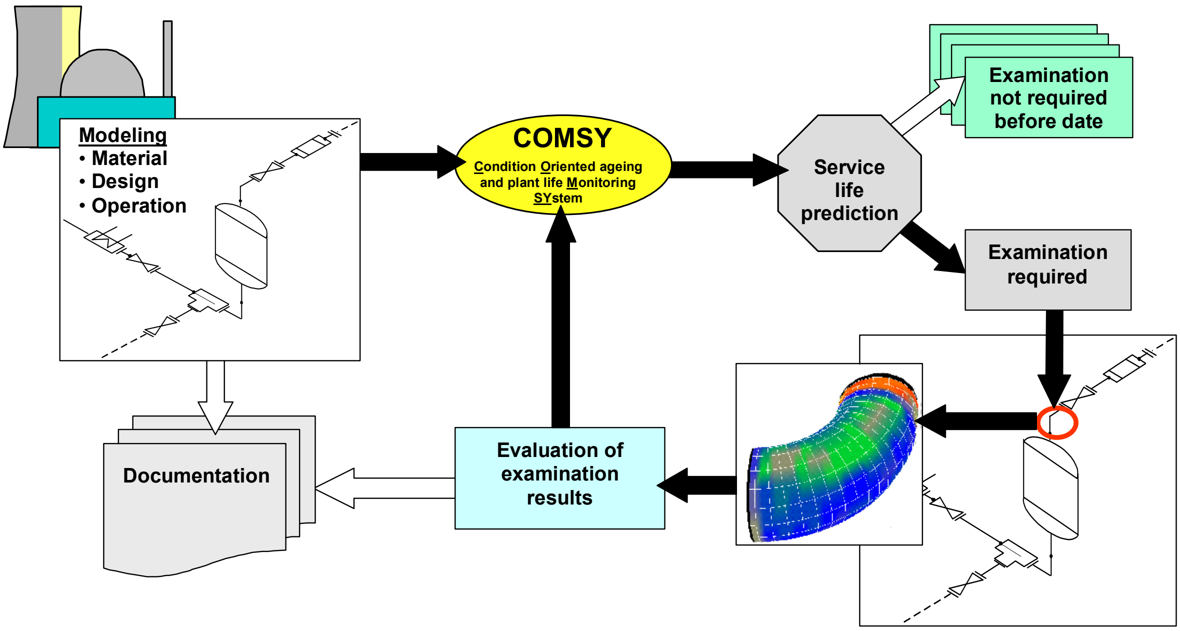

FIG.5. Closed loop process for optimized inspection scope.

The results of component examinations are fed back into the COMSY program. A calibration function serves to compare the measured degradation with the predicted degradation. This process ensures that experience gained from evaluation of examination data will be fed back into the performance of analytical service life predictions. Examination data resulting from in-service inspections are thus consistently used in the preparation of a reliable database which is kept continually up to date. Overall, this systematic, closed-loop process enables up-to-date maintenance utilizing quantifiable data characterizing the technical as-is status of the plant.

For e.g. electrical cables the capability to predict the insulation material behavior is beneficial for the scheduling of maintenance activities. The conservative prediction of cable material properties is achieved by placing of in-containment cable specimens in the vicinity of the reactor coolant piping, thus exposing them to doses and dose rates higher than in other equipment locations. Measuring and recording the changes of their material properties versus time allows for conservative lifetime prediction for similar equipment / materials in areas with lower dose rates.

3.2. Qualifiable degradation mechanism

For a number of degradation types the basic mechanism is well known and the components susceptible to the damage mechanism can be reliably distinguished by using eligibility criteria. However, the progress of degradation is not predictable as some parameters required for the prediction calculation are not known. Based on deterministic risk parameters, components can be prioritized for examination programs, however a specific inspection deadline can not be computed. Depending on the category of safety and availability requirements of components concerned, the following LM options can be considered:

- Periodic inspections of susceptible components
- Probabilistic-based assessment
- Monitoring of degradation progression.

Periodic inspection of susceptible components

Periodic inspections are commonly considered to be the most costly approach for maintaining the function or integrity of a component. In addition to the associated manpower effort and dose-rates, the delay of outage times and functional deficits resulting from the examination procedure may have to be considered.

In case periodic inspections are required for maintaining a component, the assessment effort should be focused on determining reasonably tailored inspection intervals. For e.g. mechanical components the PLIM software provides trending functions which utilize key examination results from previous inspections. In combination with e.g. a prognosis of crack growth rates for worst-case assumptions, the inspection intervals can be reasonably adjusted to local integrity requirements.

A comprehensive LM strategy is applied for e.g. the core belt line region of the reactor pressure vessel, which may be subject to irradiation embrittlement. Irradiation embrittlement causes an increase in hardness and yield strength and a decrease in fracture toughness. Periodic NDE is supplemented by RPV surveillance programs which monitor material properties of specimens.

Periodic testing is performed for e.g. electrical equipment of safety grade systems. Where possible, parameters indicative of as-is LOCA qualification can be checked periodically as well, e.g. by measuring parameters using the FANP VESPA® System or checking criteria relevant for LOCA resistance. Typical intervals between measuring the latter parameters range from 10 to 20 years, depending on the individual situation.

For civil structures it is common LM practice to perform periodic visual examination of accessible civil structures. The PLIM software provides systematic documentation of the structural condition and the associated visual information. This method provides a rapid and effective method of identifying most degradation effects.

Probabilistic-based assessment

The probabilistic assessment provides information on the probability of failure for individual components based on operational experience. In combination with detailed analysis of existing examination data, this method allows to focus inspection and maintenance effort on those components with the highest failure probability considering the failure consequence. Probabilistic software tools are provided to establish program guided inspection plans. The results of component examinations are fed back into the program system and are used to update probabilistic assessment criteria for further optimization of service intervals.

Monitoring of degradation progression

For degradation mechanisms with known characteristics of degradation progression it may be advisable to monitor the key influencing parameters, if they are not of steady nature. The application e.g. of the following monitoring systems makes the degradation progression predictable:

- In order monitor the load cycle behavior for fatigue sensitive systems (e.g. surge line or temperature stratification in piping) the fatigue monitoring system FAMOS is used to monitor temperature fluctuations. Based on the resulting load cycle record the current usage factor and the remaining life can be computed.

- The vibration monitoring system SUS is used to detect high-cycle fatigue-inducing vibrations. In addition it is used to provide timely detection of wear and fretting causes.

- A pump diagnostic system is used for pump shaft orbit monitoring motion for e.g. the reactor coolant pump.

- The valve diagnostic system DIPLUG monitors the electrical power in the drawer of the related switchgear to analyze condition of the electrical valve operator and the mechanical components of the valve.

- Main transformer monitoring: gas in oil, water in oil, humidity in expansion tank, flow velocity of air and oil, temperatures, oil level, oil pressure in HV bushings, currents, voltages, power and partial discharges.

- Main generator monitoring: temperatures (stator, rotor, bearings), vibration, current, voltage, speed, hydrogen pressure and leakage, power, excitation parameters and partial discharges.

- For civil structures the monitoring: of structural settlement is initiated during the construction phase to verify that the actual settlement is consistent with the predictions.

- Groundwater monitoring : groundwater conditions are normally observed by means of boreholes and/or wells to monitor groundwater levels, water fluctuation and chemical conditions.

3.3. Unpredictable damage

Unpredictable damage is commonly caused by unexpected situations like loose parts, chemical pollution or undetected manufacturing defects. By implementing on-line monitoring systems for specific parameters, effects of different ageing mechanisms can be detected in an early stage. Some monitoring systems are engaged to provide an early warning before component failure. Examples are:

- Also loose parts detection system (KUS) and vibration monitoring systems (SUS) can be used to provide timely detection of wear and fretting causes.

- On-line chemical monitoring system DIWA serves to survey the water chemical operating conditions on a cross-system basis. An integrated fuzzy-logic identifies functional deficits of water chemical control systems and components (e.g. the condenser). In addition the on-line measurement of redox potential can support the detection of sensitive water chemistry deviations.

4. PLIM APPLICATION ON A PLANT-WIDE BASIS

Recent experience for the application of lifetime management on a plant wide basis has been gained in the frame of a contract for the evaluation of the residual lifetime of main components and equipment for Kozloduy NPP Units 3&4.

The RLT Contract was based on earlier work performed in the frame of WANO and PHARE projects starting already back in the early 90ies. As a result, the selection of equipment and components to be considered was performed based on already available information, resulting in a remarkable reduction of the efforts for the screening related to safety and availability.

For the primary and secondary circuit components, the spent fuel storage pool, hoisting equipment, electrical equipment and civil structures the actual condition was determined by review of available information from design and operation as well as extensive walk-downs on-site. Based on the collected information, the degradation potential with respect to the different aging mechanisms was assessed and the residual lifetime of the considered equipment and components was determined. Finally, suggestions for e.g. additional analyses, inspection, monitoring, repair or replacement activities to provide for integrity and function over the operational life were given according to the different LM strategies described above. Thus, the basis for an effective future plant life management was elaborated for Kozloduy NPP Units 3&4.

5. IT-SUPPORT FOR LIFE MANGEMENT

Activities in the field of plant life management generally require the systematic processing of large amounts of data, which has to be continuously updated to describe the current ageing status of the plant. With the use of specifically tailored software tools for PLIM, this process can be efficiently streamlined.

Software tools for plant life management have to meet the following basic requirements :

- The capability to store, update and manage all relevant data which are essential for the associated field of ageing and plant life management. Next to the capabilities of a standard documentation system, it should be able to provide comprehensive numerical parameters for the application of mathematical functions for analytical assessment.
- The capability to store information concerning environmental operation conditions and component status in a service time dependent mode, in order to consider the impact of changes initiated during the service history of the plant.

In order to streamline efforts regarding PLIM, the software should provide the computational functionality to qualify and – if possible – to quantify the effect of degradation mechanisms. Probabilistic assessment functions and engineering judgement functions should be available to support assessment regarding degradation mechanisms, for which existing lifetime prediction models are not applicable. For the purpose of processing feedback on component as-is information obtained e.g. from examination activities, the software should be capable to directly access numerical information resulting from e.g. NDE testing.

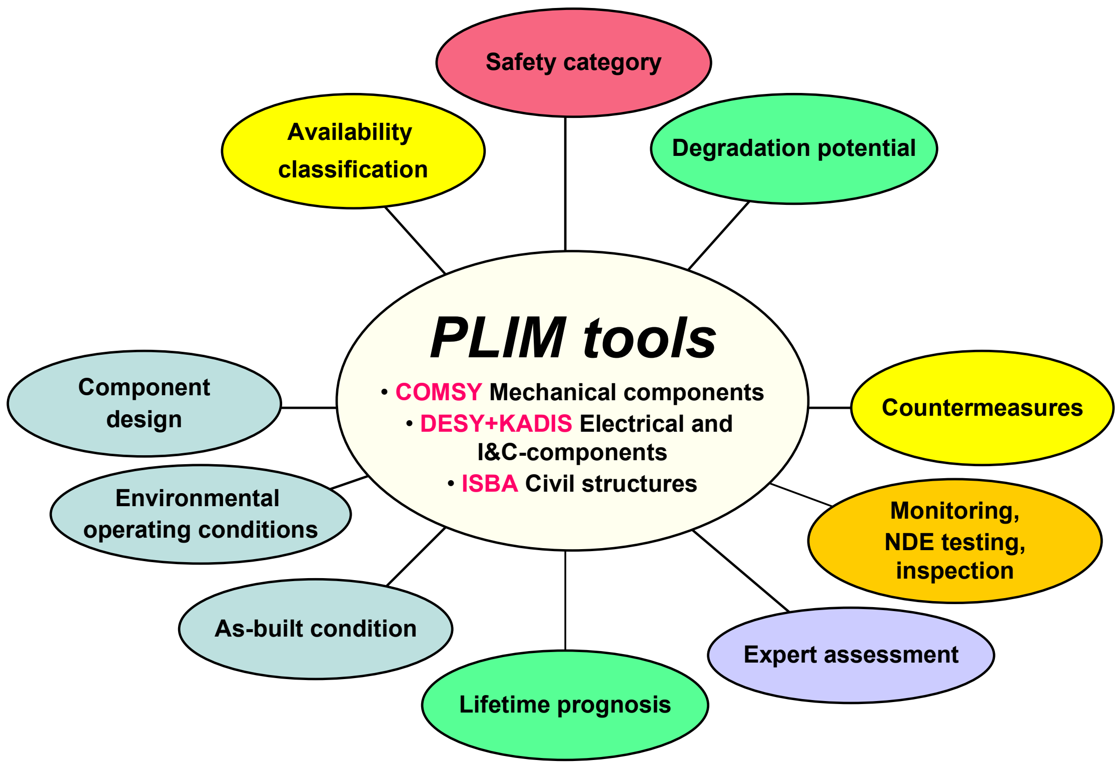

FIG.6. Framatome ANP software family for PLIM.

In the following software products providing IT-support for different fields of plant life management will be described. Due to the specific approaches and technologies applied for mechanical components, electrical components and civil structures, the corresponding software tools are to a large extent unique. The use of an uniform identification system, which is used to establish relations between different fields of engineering, allows to access information contained within the different modules.

5.1. Mechanical Components

The COMSY (Condition Oriented ageing and plant life Monitoring SYstem) program has been developed to provide a software tool for the ageing and plant life management of systems and mechanical components in power plants. The PC-based program by Framatome ANP GmbH provides the functionality to assist plant life management (PLIM) activities and innovative maintenance management methods. It is designed to support a plant-wide strategy for identifying components with an existing degradation potential and is capable to support an ageing monitoring process by providing lifetime predictions for mechanical elements, which are validated by a small number of examinations at priority locations.

In order to identify components with an degradation potential, a screening is performed on the plant in a cost-effective manner. Based on the heat balance diagram a plant systems model is established with information regarding operation conditions of all relevant plant systems and components. Based on this simplified model an analysis of the water chemistry cycle is conducted based on the thermal-hydraulic parameters (see Chap. 2).

Taking into consideration environmental operating conditions and representative materials used in each case, the system areas are studied with respect to different degradation mechanisms. The results provide a classification, indicating which power plant systems areas are expected to suffer form degradation due to their design and operating parameters. Systems areas with no indications can be excluded for further processing. Based on the table of system areas with degradation potential existing, the results of the periodic safety review are used to identify components important to safety and plant availability as components for PLIM.

For PLIM systems and components a ageing monitoring process is initiated. For this purpose the program generates component specific folders which compile all ageing relevant parameters and documents.

For the performance of lifetime prediction additional data are required, which describe the systems, lines or components in detail. As many of the system parameters required for lifetime prediction cannot be obtained from system documentation, a number of engineering tools for data preprocessing are supplied with the COMSY program system, e.g. flow calculations, stress analysis functions, water chemical calculations. A material library serves as a knowledge base for details on material properties for a large number of commonly applied steels.

Based on the calculated progress of degradation of a line or component, the program computes the minimum residual life expectancy for individual elements considering local stress conditions. The resulting service life prediction is validated and optimized through the performance of a small number of examinations at priority locations, which are indicated by the program. Based on examination results, a validation/update process leads to a reduction of conservatism included with the prediction process. This systematic closed loop process ensures the generation of a quantifiable database which is continually kept up to date with information related to the technical as-is status of the plant.

COMSY handles the storage, administration and evaluation of UT examinations and visual inspections for individual plant elements. Trending diagrams show material degradation tendencies on piping elements, action schedules support the planning of inspection campaigns. The evaluation of component examinations is supported by interactive analysis functions which greatly simplify the geometry-dependent evaluation of measurement results. A calibration function supports the comparison of the as-measured condition with the predicted progression of the degradation, while making allowance for measurement tolerances. The results of this comparison are used in order to improve the accuracy of future service life predictions

In order to consider the as-built condition of PLIM components COMSY provides methodically compiled documentation for design and operation conditions of relevant plant systems as well as NDE examination data. If required, the program also supports the storage and administration of material certificate values for mechanical components (as-built properties) and the administration of manufacturing data including the documentation of mechanical testing results for base material and welding material.

During the ageing control process this database is continuously kept up to date. Information resulting from surveillance activities, e.g. NDE results, are thus consistently used to provide the capability to systematically trace the current condition of LM components.

The capability to perform service life predictions is the key function of a software system for ageing and plant life management. On the basis of systematic, software supported approach to plant life management, maintenance management and plant availability can be optimized. This capability is particularly useful for the service life extension of systems and components. The implementation/application of COMSY in various power plants within and outside Germany (e.g. Japan, Spain, Switzerland, Finland, Bulgaria and Hungary) has confirmed that systematic plant life management makes good economic sense, and that the process can be greatly streamlined through software support.

5.2. Electrical and I&C-Components

DESY is an integrated tool for the documentation of the electrical power supply system data:

- it contains all information for the unit power supply in terms of numerical data
- all information of branches circuits can be graphically and numerically displayed, including designations and data of e.g. bus bars, circuit breakers, cables in mild environment, containment penetrations, cables in harsh environment, junction boxes, plugs and sockets and the consumers
- it provides features for general analysis and diagnostics of location related faults, branch related faults or object related faults
- it can perform specific evaluation of electrical data.

With the graphic tool **KADIS** one has a virtual, visual access to building specific and room specific information, like

- overall cable routing including all rooms which are passed and
- location of the parts of the function chain
- complemented by information like
- local temperature and irradiation dose rate
- operating temperature of the component and
- remaining qualified life time corresponding to original qualification, considering local and operational loads.

5.3. Civil Structures

The following chapter gives a brief description of ISBA, a design tool and management system for all phases of the lifetime of nuclear power plants. ISBA is interconnected as one module among others by one common user-interface in order to meet all needs of life management for a NPP.

Operation of nuclear power plants requires a large amount of information. This information consists of e.g. descriptions, licensing documents and drawings. ISBA (i.e. an acronym based on the German, meaning "Civil engineering and plant layout information system") was initially developed to provide all information related to civil engineering in one central data pool. It is based on an object-oriented database, which allows governing not only individual data but also rules and relations. This is accomplished with the implementation of a global product data model. Today the system has reached the capability to manage all information desired.

Starting with the 3D-model where all geometric data are created and complemented by attributes like materials, any information and document can be linked with each object of the model. Picking the particular object may retrieve such information. Thus a digital archive may be created that assures the availability of stored information over the entire lifetime of a plant. Moreover ISBA contains functionality to support managing the operation of the facility. These features allow keeping accurately track for all needs of an efficient ageing management. The database driven approach guarantees security for all investments (both financial and human) over generations of engineers.

The system provides generally two options for the access to the information pool. The classic way is via the 3D model of the plant represented in AutoCAD as viewer. The alternative is to communicate with the system through the Internet/Intranet. This alternative enables the user to communicate with ISBA without the need of having special software programs installed on the PC. An Internet browser is sufficient to get in contact with ISBA and work.

6. CONCLUSIONS

For many utilities a systematic and efficient ageing and plant life management system is becoming more and more important to ensure an economical power plant operation in spite of continued facility ageing.

Framatome ANP GmbH provides a strategic PLIM concept, which focuses maintenance activities on safety and availability relevant components, were a degradation potential exists. The selection process applied provides reasonable prioritization of ageing relevant components and ensures that efforts are devoted to elements, where ageing is a relevant concern. For the selected components special life management (LM) strategies are established to monitor the as-is condition in respect to life limiting situations.

In order to efficiently support the PLIM processes, advanced IT tools have been developed In this regard the Framatome PLIM software family makes a knowledge-based program system available which integrates advanced analysis tools and comprehensive functionality with a "virtual power plant data model." It enables the condition-oriented and risk-informed service life evaluation of mechanical components, electrical components and civil structures with respect to relevant degradation mechanisms. The results of component examinations are fed back into the program system, and are used for further status evaluation over the life cycle of power plant systems. Overall, this systematic process enables up-to-date maintenance utilizing quantifiable data characterizing the technical as-is status of the plant. On the basis of reliable and degradation-relevant predictions, maintenance management and plant availability can be optimized and the service life of costly systems and components extended.

The PLIM concept described has already been applied to individual plants on a plant-wide basis (see Chap. 4) of for specific tasks. The implementation/application of e.g. PLIM software tools in various power plants within and outside Germany (e.g. Japan, Spain, Switzerland, Finland, Bulgaria and Hungary) has confirmed that systematic plant life management makes good economic sense, and that the process can be greatly streamlined through software support.

REFERENCES

[1] Schwarz, W., Waas, U., Kühne, B., Überprüfung der Störfallsicherheit, Schutzzielorientierte Sicherheitsstatusanalyse am Beispiel von Neckar-1, atw 42, S. 23, Januar 1997

[2] Helmut Nopper, Manfred Erve, Wolfgang Kastner, Lars Scharf, COMSY – A Program System for Aging and Plant Life Management, Plim&Plex Meeting on Plant Life Management and Plant Life Extension in Nuclear Facilities, 3-5 November 1999, Madrid, Spain

[3] H. Nopper, B. Kastner, Cost effective plant life management, Nuclear Engineering International, P36-37, November 2000

[4] H.Nopper, W.Daeuwel, W.Kastner, Ageing and Plant Life Management Software COMSY, Ninth International Conference on Nuclear Engineering, Nice Acropolis – France, April 8-12, 2001

[5] Schillberg, H.; Drozella, H.; Objektorientierte Produktmodellierung für die Tragwerksplanung im Kraftwerksbau, Darmstädter Seminar "Multimedia im Bauwesen" Band 16. Freunde des Instituts für Massivbau der TH Darmstadt e.V.

[6] Schillberg, H.; Drozella, H.; Technische Informationssysteme für die bautechnische Bearbeitung von Kraftwerksanlagen, Bauingenieur 72 (1997) 479-487

[7] Schillberg, H.; Drozella, H.; Technical Information System for Nuclear Power Plants, Transaction of the 15th International Conference on Structural Mechanics in Reactor Technology (SMIRT-15) Seoul, Korea, 1999

[8] Schillberg, H.; Drozella, H.; ISBA, a Technical Information System for the Construction, Operation, Service, Backfitting and Ageing Management of Nuclear

Power Plants, Transaction of the 16th International Conference on Structural Mechanics in Reactor Technology (SMIRT-16), Washington, USA, 2001
[9] Hans-Juergen Golembiewski, Bela Bechtold and Karlheinz Storb IMPROVEMENT OF AVAILABILITY AND SAFETY BY ADVANCED MONITORING SYSTEMS FOR NUCLEAR POWER PLANTS, Siemens AG Power Generation Group, Erlangen 22-11-2000

RENEWAL OF THE OPERATION LICENSE FOR PAKS NPP, REGULATORY PROGRAM

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Abstract

The Paks NPP units have a 30 year design life that expires in the period 2012-2017. The Hungarian Atomic Energy Authority elaborated a set of guidelines for the licence renewal program. The US NRC approach was selected as a basis for licence renewal. It requires the separation of the tasks of Final Safety Analysis Report according to NRC Reg. Guide 1.70. and Periodic Safety Report according to IAEA draft safety guide DS-307.

1. CURRENT SITUATION

- Unit No.1 of Paks NPP was commissioned in 1982, its so called "design life" was 30 years, which expires in 2012.
- According to earlier Hungarian legislation the design life was the limit of the licence for operation. Recently this legal restriction has been cancelled, but there is an obligation: if the Licensee wants to operate a unit longer than it's design life, it shall submit an application 5 years before the expiration date. There are no more detailed guidelines on how to do this.
- The license renewal can be performed via: application – preliminary license – fulfilment of the preliminary license conditions and prescriptions – final license renewal (at the expiration of the current license).0
- Originally the FSAR was elaborated according to Soviet practice of the 70s'. It means: most of detailed strength, thermal, hydraulic, physic and other safety analyses were not provided for Hungarian site. They were elaborated and approved (we hope so) by the Soviet Regulatory Body when licensing the prototype unit (Novo- Voronezh unit 3).
- Part of missing analyses was performed in Hungary in early 90s. (AGNES project)
- We don't know the basis of the defined 30 years design lifetime, but we know that in certain manufacturers' documentation there is a warranty of reliable operation for 30 years. It is similar to American situation where the 40 years were based on antitrust low and then this term went to the manufacturers analyses (e.g. fatigue analyses, qualified lifetime of certain equipment).
- According to the new Atomic Law the FSAR has to be elaborated on the basis of the Reg. Guide 1-70. There are detailed instructions in the Reg. Guide and in its Standard Review Plan (NUREG – 0800) on contents and aspects of analyses and descriptions that can not be fulfilled using the available documentation. This information should be completed.

- There is another legal prescription: once in teen years the Licensee shall submit a Periodic Safety Review, which is the basis for extension of the operation license for next 10 years. Currently the PSR for unit No.1 is valid until 2008. It means the Licensee shall submit an application for the License Renewal (if it wants to do so) in 2007 and the next PSR in 2008.

2. PREPARATORY ACTIVITIES

- Team members have studied the IAEA TECDOC series on ageing of different major SSCs', AMAT and PDRP documents etc. Also we studied US legal documents (10 CFR 50 and 54), documents of the NPAR program and many NRC guidance and compliance documents.
- The team realised that the NRC Reg. Guide and the IAEA Guide were elaborated independently each from other. NRC hasn't considered a Periodic Safety Review and the IAEA Guide doesn't rely on an updated FSAR. The two documents differ also in the scope on related SSCs, in details of description of the tasks of both Licensees' and Regulator's etc.
- HAEA decided to use American approach for the license renewal for two reasons:
    a) There is a real experience – US NRC really made license renewal in case of some NPPs.
    b) There is detailed open information about regulatory requirements, guidelines and licensees' applications for license renewal as well.
- It is not so easy to explain in a few words what does it mean, but the main features are as follows:
    - The scope of LR contains only the passive and long lived components (passive are the ones that perform their intended function without moving parts or changes in configuration or characteristics, long lived are the ones that have no qualified life time or other life time restriction shorter than 30 years),
    - the licence renewal application consists of scoping, screening, integrated plant assessment, FSAR modification, TS modification (if necessary), and environmental impact study,
    - the licence renewal does not consist of resolving problems that should be resolved for maintaining of current operating licence conditions
    - the safety margins of operation should not be deteriorated neither at the end of the design life time nor at the end of the renewed life time
- This approach can be utilised only if we adopt the conditions of the operation licence according to 10 CFR 50 regulation. So we carefully reviewed that conditions and identified four major prerequisites to utilise the American practice.

3. LESSONS LEARNED

- The first thing was to understand the role and relation of the FSAR and PSR.
- After discussions the team agreed on understanding of the FSAR as a living document, a verification of “as is” design conditions for safe operation. In this meaning the FSAR is the basic document for giving and retaining of the operation licence. The FSAR focuses on description of the current configuration of the unit and on safety analyses that are in line with this current configuration. (A continuous comparison with current configuration is important while performing a wide safety enhancement program – which is the case in Paks - or other major modifications.)
- If the FSAR bears the responsibility for maintaining licence conditions, the PSR can concentrate on ageing issues in wider sense of the expression: ageing of SSCs, people, organisation, requirements, procedures etc.
- The PSR gives an opportunity for a complex overview of SSCs’ technical conditions, their ageing management programs, environmental conditions, changes in the state of the art of science and technology, utilisation of operational experience, etc.
- In other words the PSR can skip the description and the safety analyses of the unit and its SSCs, or just refers to FSAR. In our opinion the recent changes in new IAEA draft safety guide DS-307 can be fitted to this separation of targets.
- The second thing was to decide what to do with missing input design information. We decided to include the gaps of input information to the list of unresolved safety issues for further investigation or resolving. The meaning of this decision was that we didn’t want to require a complete “re-design” of the NPP, which has been operated (very well) for more than 20 years. The unresolved safety issues will be resolved “on demand” of current operation license conditions and the license renewal application.
- The comparison of Hungarian licensing conditions to 10CFR50 and 10CFR54 requirements resulted in identification of four major missing areas:
    a) completion of design basis information
    b) implementation of a new regulation for monitoring of maintenance effectiveness (similar to 10CFR50.65)
    c) completion of environmental qualification for electrical and I&C equipment
    d) introduction of a systematic ageing management program.
- There are three important statements to these four areas:
    a) “If the license renewal is a chair, these prerequisites are the four legs of it.” In other words: these areas are to be managed before the license renewal application.
    b) The “legs” can be (and will be) different for different countries, NPPs, regulatory systems etc.
    c) These legs should be carved out under current license conditions **even if the Licensee calls off** the license renewal.

4. ACHIEVEMENTS AND FURTHER TASKS

- The first version of guides on ageing management has been issued. These guides answer to the current structure of the Hungarian Nuclear Safety Regulations – there are guides for:
    - a) Consideration of ageing processes during design of NPP and equipment
    - b) Ageing management during operation of NPP
    - c) QA tools in ageing management program
    - d) Regulatory processes in ageing management program
- It turned out that the listing of equipment that needs ageing management is not so easy. A thorough analysis of safety related equipment was necessary for proper scoping of SSCs, degradation places and processes, tools to cop with these processes etc. The Hungarian Electric Research Institute provided a large set of ageing analyses of Paks NPP equipment.
- Three guides on equipment qualification have been issued (methodologies for design, operation and regulatory processes).
- Two draft guides on implementation of a new Maintenance Rule are under discussion with the Licensee.
- The regulatory expectations for FSAR have been included in the QA manual of the FSAR upgrading project.
- Two draft guides are elaborated on license renewal project requirements and on the content of the license renewal application.
- The further tasks were distributed in three groups:
    - a) tasks for the period of preparation of the license renewal application
    - b) tasks for the last five years of operation under design lifetime
    - c) tasks to be performed in the renewed license period.

IAEA-CN-92/19

PURPOSE-ORIENTED PROGRAM OF RUSSIAN MINATOM RESEARCH WORK IN SUPPORT OF LIFE CYCLE MANAGEMENT (LCM)

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Abstract

The life cycle management of nuclear power plants is a composite complex process associated with multifactor analysis and important decision-making. While accomplishing this work, knowledge and technologies are used, which are not only specific and particular to the given unit, but they are also of a more general nature ensuring successful life cycle management of the plant. Purpose-oriented program of the research works (RW) of the RF Minatom in support of life cycle management is specifically aimed to solve this problem. The program is designed for 8 years; it includes about 50 subject tasks and envisages development and perfection of normative and methodological documents, support databases, computational software and means for their verification. Some of the Program positions are close to have been completed.

1. INTRODUCTION

The life cycle management of nuclear power plants is a methodology and practice of ensuring profitability and safe operation of commercial energy generating plants. The ideology of the process of this management is the optimization of the profit/safety relationship. The world operation experience in nuclear power industry demonstrates that optimization of technological & economic indices and safety level tasks can be solved for each NPP unit with allowance for the whole life cycle only, starting from design of the power unit to the moment of its decommissioning.

It is scheduled to ensure preferential increase of electrical power generation in Russia by nuclear energy industry under the "Program of Nuclear Energy Development in the Russian Federation during 1998-2005 and for a Period till 2010" approved by the government of Russia. During this process, vital importance is attached to the introduction of the life cycle management technology, which Russian Minatom has announced to be the baseline conceptual approach with strategic consequences in outlook. The life cycle management is a composite complex process associated with important decision-making on the basis of multifactor analysis. The examination of the contents of basic type work under the life cycle management demonstrates that the total amount of work for a particular power unit can be represented as a sum of two components, viz. the one particular to the given unit and the general one. The former component consists in the implementation of the life cycle management process on a specific unit. While implementing this work, knowledge and technologies are being applied, which are not specific and particular for the given unit only, but they are also of a more general nature, so the set of them can be called as the base ensuring life cycle management and used during accomplishment of the general component.

The replenishment and perfection of the ensuring base requires considerable expenses and efforts, and they should be carefully planned. For these purposes, the program of research work of Russian Minatom "Development and Perfection of Normative and Methodological Bases to Ensure Implementation of Works on NPP Units Service Life Management" was developed for 2000-2007. It envisages updating and creation of the package of normative and methodological documents, databases, algorithms and computational programs indispensable for life cycle management of NPP equipment and components and for technological justification of power units' safety. The program is prepared by the Co-ordinating Committee on nuclear plants life management. The Committee is formed from the authorized representative of leading organizations and enterprises of the Minatom and cooperating industries involved in the work on life cycle management supporting base.

2. DESCRIPTION OF THE PURPOSE-ORIENTED PROGRAM

2.1. General Structure

The program consists of the following five sections:

– Development and perfection of normative and methodological bases.
– Development and perfection of support databases.
– Research work for justification of the normative and methodological documents under development.
– Development of software and databases needed for their verification, as well as benchmark control samples matrix.
– Information/methodological assistance and control of the Program.

The sections of the Program are mutually matched as to the contents and time periods.

2.2. Contents of the Program Sections

2.2.1. The implementation of the life cycle management technology requires for corrections to be introduced into the effective normative and methodological documents, as well as for a number of new ones supplementing the normative and methodological base to be developed. Accordingly, the development of documents on the following directions is included into the first section of the Program:

– Fundamentals of service life management and extension/renewal of the operation license.
– Methodology of gathering, processing and analysis of the information on operation and conditions of safety-related equipment and components.
– Estimation of veracity of non-destruction inspection results and ageing control.
– Management of the components' reliability and severe accidents of power units.

The work under this section of the Program is based on results of activities under section 3 and, accordingly, they are scheduled for a period of 2003-2006.

2.2.2. The contents of life cycle management of a power unit, as well as that of any management, consists of an "analysis - decision making - action" line and a feedback system. The basis of success here is taking of the exact decision, which requires the respective data support.

In this connection, gathering, processing, storage and analysis of the information on life cycle of each power unit and its basic components and systems, i.e. creation of the appropriate databases, is becoming a major element of the life cycle management. A continuity of information in time (along with the whole life cycle) and its entirety serve as a basis of success while solving the task of optimization of the profit/safety relationship. Here the application of modern information technologies is a demand of time.

Accordingly, the development of the general-purpose databases on the following directions is included in the second section of the Program:

- History of operation loading.
- Properties of materials.
- Diagnostics and non-destructive inspection.
- Damaging factors and degradation.

The contents of this section are coordinated with other sections of the Program ensuring their information support. The work on the section is scheduled for a period of 2000 - 2006.

2.2.3. The perfection and creation of the package of normative and methodological documents aimed to support NPP units' life cycle management requires for the appropriate R&D base in justification of the contents of these documents to be developed. Accordingly, the work on section 3 of the Program is coordinated with sections 1 and 2 and is scheduled for a period of 2000 - 2007 as to the subject and time periods.

2.2.4. The implementation of life cycle management technology is associated with the application of computational software, which must be verified and certified. Accordingly, the development of the following is included in section 4 of the Program:

- Algorithms and software.
- Databases for computational codes validation.
- Benchmark samples banks to validate the codes.

The works on this section is scheduled for a period of 2000 - 2007.

2.2.5. Practical application and implementation of the developed subjects obtained as a result of realization of sections 1, 2, and 4 of the Program requires information/methodological assistance for specialists and industry enterprises to be performed via meetings, seminars and consultations.

Program management also requires some organizational measures, viz. organization of expert analysis of performance specifications and research results; consideration, coordination and distribution of documents; organization of activities under the Program. This activity is stipulated by section 5 and, accordingly, is scheduled for a period of 2000-2007.

3. STATE OF WORKS UNDER THE PROGRAM

Program realization was commenced in the second half of the year 2000, and by now only a small part of its subjects is close to have been completed. Abstract data on the three of them is given below.

3.1. Fundamentals of Plant Life Management in Russia

The work under this subject was commenced with time advancement of the schedule. At present a preliminary version of the future document is available.

The document contains a conceptual account of fundamentals of Russian plant life management technology. This technology is being considered as a concept and process covering the whole life cycle of NPP that consists of three basic phases, viz. pre-operation; operation (including the time period of prolonged/renewed license on operation); after-operation.

The total amount of material is about 20 pages. It is structured on the following eight sections:

1. Introduction
2. General
3. Provision of conditions for preparation and realization of plant life management (PLIM) of power units at their life cycle stages
4. Ways of provision and safety enhancement at stages of PLIM realization of power units
5. Features of PLIM directions realization of operating reactor facilities and power units with different types of reactors
6. Organization of licensing at stages of power units life cycle
7. Provision of public acceptance of PLIM power units
8. Sources of development

The completion of work under the document is scheduled to 2003.

3.2. Database on Physical & Mechanical Properties of the VVER Pressure Vessels

The pressure vessel materials of VVER reactors operate under difficult operating conditions and must comply with a broad range of requirements to service properties. Therefore, problem of the reactor vessels integrity could be solved by applying a comprehensive approach only, which would allow for all diversity of factors influencing reactor service life being taken into account. A necessary component of such comprehensive approach is creation of effective data support of the whole existence cycle of a reactor vessel (design - manufacture - operation). The basis of this data support can be computer-aided information system made with the use of modern database management systems.

The work commenced in 2000 includes the following:

- Review of state-of-the-art in this field.
- Development of DB structure, composition and format.
- Finalizing of data acquisition methodology and its preparation to be further input into DB.
- Design of DB computer version.
- Approbation of DB structure.
- Gathering and preparation of information to be included into DB.
- Development of computer programs range to analyze data input into DB.
- Development of program interface for information being displayed from DB.

The database is implemented on DEC Alpha server with Compaq True 64 Unix operation system under DB Oracle 8i server control by way of a code written on DB – SQL programming language (language of structured request). The work scheduled for 2003 will include information inputting into DB and preparation for trial operation.

3.3. Database for Thermo-Hydraulic Codes Validation

The purpose of work consists in the creation of a database based on up-to-date information technologies necessary to validate thermo-hydraulic computational codes that simulate reactor facility and are used to assess its safety.

The database for system thermo-hydraulic codes validation is the vital modern tool for checking the adequacy level of computer simulation of actual thermo-hydraulic processes that take place in the analyzed reactor facility with application of the validated code.

Previously performed experiments on test facilities of ENIC, RRC Kurchatov Institute and NIKIET were considered as the first standard safety problems while implementing validation for the RBMK reactor. A list of experiments on the existing test facilities realizing other modes for experimental and design clarification of different parameters was proposed. The analysis of specific accident modes of NPP with RBMK reactor was conducted, which allowed finding out of important processes and effects determining the progress of accident modes.

A large list of phenomena specific to each reactor design was chosen and characteristics of these phenomena were given for standard validation problems for the VVER reactors. Descriptions of used experimental facilities for studying VVER particular design features were formulated and the available experimental data on standard safety problems were submitted.

The construction of database based on experimental results of thermo-hydraulic test facilities was being commenced with a data analysis. During analysis of test facility's structure, the classification of design features of the test facility elements was conducted. This analysis was conducted with the purposes of determination of exact links and optimization of the data structure. The method of CASE-simulation and the tools ensuring automation of this process, i.e. Oracle Designer 2000 version 2.1, was used for data analysis.

The created database is oriented to be used during RELAP5-3.2 code validation. It is implemented on DEC Alpha server with Compaq True64 Unix operating system under DB Oracle 8i server control by way of a code written on DB - SQL programming language (language of structured request). It is scheduled to complete the work on database inputting with available information in 2003.

4.CONCLUSION

The NPP life cycle management is an integral part of this nuclear facility operation and safety control process.

The perfection and introduction of LCM technology into the NPP operation practice is a clue to success in nuclear power industry. The development of a support base is necessary for LCM technology to have been successfully translated into reality.

Session 2

PLANT LIFE MANAGEMENT (PLIM)

Sub-Session 2.3

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REST LIFE TIME MANAGEMENT OF KOZLODUY NPP UNITS 3 AND 4

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Abstract

Unit 3 and Unit 4 of NPP Kozloduy are second-generation WWER440/230 energy reactors. Unit 3 was in operation from 1980, Unit 4 – from 1982. No surveillance specimens for monitoring of RPV metal aging are available for both KNPP Units. The reactor pressure vessels rest lifetime assessment, Plant life management and Surveillance program development for the both units are presented and analyzed in this report.

Introduction

Each economically grounded and technically effective program for the management of aging mechanisms in nuclear power plant is a part of efficient plant management. The first step in development of such a program is to adapt an approved methodology taking into account the specifics of the particular plant.

The program developed for Kozloduy NPP Units 3&4 included the following stages.

- First stage - determination of the RLT of the most important components and systems responsible for the nuclear safety.
- Second stage - creation of a computerized data base for all components and systems for the period from the start of NPP operation till the current moment including design data, data obtained during exploitation giving a special account to the influence on environment, all results from in-service inspection and repair works.
- Third stage – development of program for the aging management which will permit detection of aging processes and identification of components in which they could appear, determination and management of aging mechanisms in order to reduce their effect on degradation processes in plant components.

The components addressed are all high-safety components and representatively all other components, systems, and equipment. These are:
- All components of the primary circuit;
- Components of the secondary circuit (piping, valves and supports);
- Spent fuel storage pool;
- Hoisting equipment;
- Electrical equipment (i.e. generator, transformers (main and house), batteries, switch gears, cables and diesel generators);
- Civil structures (such as the reactor building, turbine building, ventilation stack, double channel and pump stations).

In most cases, the relevant component-specific degradation mechanisms are well known from the results of previous projects related to the residual service life of NPP Kozloduy

Units 1-4, from experience with the operation of similar plants but also plants in other countries. Those specific degradation mechanisms are addressed within the specific work program:

- Either by investigation of available samples form an RPV metal (remaining material from sampling in weld) 4 from PRV Unit 1);
- By research on possible embrittlement due to neutron radiation (RPV and RPV support structure);
- By stress and fatigue analyses of possible degradation sites (at the RPV, secondary piping, pumps and valves), or
- By providing suitable tools to monitor or calculate fatigue (FAMOS), erosion corrosion (COMSY) or radiation fluence levels;
- Also the main lifting equipment, the electrical equipment (including I&C) and building structures are included by assessing the status quo with respect to possible degradation mechanisms and sites.

The work defined will allow and provide a substantial step forward in the quantitative evaluation of the residual service life on the Units 3 and 4.

On the basis of the available documentation and information gathered by walk-downs and from discussions with KNPP staff, a quantitative assessment of the residual lifetime was performed, taking into account design, material properties, loads and operational history as far as respective information. The residual lifetime was evaluated based on the assumption of:

- Regular inspection according to the valid standards and regulations applying qualified techniques, equipment, personnel and acceptance criteria;
- Regular maintenance according to the manufacturer's requirements or valid standards and regulations;
- Timely repair of registered defects or defective parts.

In case no algorithm or accustomed methodology, or none or insufficient data has been available for quantitative evaluation of the RLT, a reliable engineering judgment was elaborated for quantitative evaluation of the residual lifetime of the components, systems and structures.

As a conclusion of RLT of the considered components, systems and structures of Kozloduy NPP Units 3&4 until their design life of 30 years is possible, taking into account certain recommendations, such as:

- Giving continuous special focus to the irradiation behavior of Weld 4 of Unit 3 Reactor Pressure Vessel;
- Collection of additional information as well as subsequent analysis e.g. for some valves;
- Experimental verification of current condition of cables, and
- Performance of repair works e.g. at specific areas of civil structures and hydrotechnical facilities.

The radiation lifetime of reactor pressure vessel (RPV) is the most important limiting factor for the term of exploitation of the whole power unit. The main degradation mechanism of RPV metal which should be managed is the neutron induced embrittlement. Processes of radiation aging running in RPV metal lead to fracture toughness decrease and to increased probability of brittle fracture of the vessel under thermal shocks. This explains the importance of RPV integrity assessment and rest lifetime management.

Base information on NPP Kozloduy Unit 3 and Unit 4 RPVs

Reactor pressure vessels of NPP Kozloduy Unit 3 (KNPP3) and Unit 4 (KNPP4) are second generation WWER440/230. Unit 3 was put in operation in 1980, Unit 4 – in 1982. Both reactors have anticorrosion cladding. The chemical composition and mechanical properties of reactor shells have been determined by the manufacturer. The content of phosphorous and copper in KNPP3 RPV weld 4 is typical for WWER440/230 model and in KNPP4 they are similar to WWER440/213 model.

Low leakage core zone charging scheme was applied in both units in order to reduce the neutron loading on RPV wall. Additionally 36 dummy cassettes were inserted in core zone periphery of KNPP3.

Recovery annealing was applied to KNPP3 RPV weld 4 metal in 1989.

Rest life time assessment

No surveillance specimens for monitoring of RPV metal aging are available for both KNPP Units. Cutting of templates from RPV wall is not recommended because of the presence of cladding. So KNPP3 and KNPP4 RPV metal current status can not be experimentally evaluated.

The existing information on impurity elements concentration and on initial critical temperature of embrittlement ensure the possibility to predict the rest life time by empirical methods provided that embrittlement and re-embrittlement laws are reliably confirmed. The experience and results gathered from other WWER440/230 RPVs should be used for this purpose.

KNPP3

The method for lifetime assessment of annealed RPVs accepted in Russian standards is based on the conservative approach of reembrittlement. The analysis of KNPP1, KNPP2, Novovoronej 3 and 4 weld metal re-irradiation data shows that the rate of re-embrittlement of WWER440/230 RPVs weld 4 metal is lower than predicted by this approach and the lateral law is valid in this case [1, 2, 3]. The lateral re-embrittlement law should be applied for KNPP3 RPV radiation lifetime assessment also because of the similar chemical composition of KNPP3 weld 4 metal.

Tkf calculations made according lateral embrittlement law show that the KNPP3 RPV integrity is proved practically up to 28th fuel cycle, i.e. to the end of design lifetime. The application of elasto-plactic fracture mechanics methods for Tka calculation permits the rest lifetime to be extended beyond designed one.

KNPP4

The prognostic calculations of Tkf shift of KNPP4 weld 4 metal during operation were performed. It was shown that the exploitation of Unit 4 will be safe beyond the design life time.

Surveillance program development

A Surveillance program for KNPP3 and KNPP4 RPV metal is under development at the moment. As no archive RPV material for specimens manufacturing is available, RPV materials of similar composition, manufactured by identical technology like Unit 3 and Unit 4 materials will be used for Charpy and tensile specimens manufacturing (Table 1).

Table 1

Unit	Material	P	Cu
KNPP3	WM	0.040	0,17
KNPP3	WM	0.035	0,17
KNPP4	WM	0.012	0,11
KNPP4	WM	0,023	0,03

Containers with surveillance specimens, arranged in chains, will be inserted in surveillance channels of NPP Rovno 1 (low flux) and 2 (high flux) for irradiation. Investigation of weld metal behavior after initial irradiation, annealing and re-irradiation is planned in KNPP3 Surveillance program. Flux effect for KNPP3 specimens will be studied. For this purpose irradiation and re-irradiation at high and low neutron flux will be performed parallelly.

KNPP4 Surveillance program foresees investigation of base and weld metal behavior under irradiation. Initial accelerated irradiation up to fluence 8.10^{19} cm^{-2} is foreseen followed by two irradiations at low flux to fluence 11.10^{19} cm^{-2} and to 14.10^{19} cm^{-2}.

REFERENCES

[1] Siemens AG, Evaluation of rate and degree of irradiation embrittlement of KNPP1 weld4 metal during re-irradiation.

[2] Siemens AG, Evaluation of rate and degree of irradiation embrittlement of KNPP2 weld4 metal during re-irradiation.

[3] Y. Korolev, A.Kryukov, Yu.Nikolaev, P.Platonov, Ya. Shtrombakh, R.Langer, C.Leitz, C.Rieg, Assessment if irradiation response of WWER-440 welds using samples taken from Novoronezh unit 3 and 4 reactor pressure vessels, Nuclear Engineering and Design, 185 (1998), 309-317.

ENHANCING PLANT PERFORMANCE IN NEWER CANDU PLANTS UTILIZING PLIM METHODOLOGIES

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Abstract

Over the past 5 years, Atomic Energy of Canada Ltd. (AECL) has been working with CANDU utilities on comprehensive and integrated CANDU Plant Life Management (PLiM) programs for successful and reliable operation through design life and beyond. Considerable progress has been made in the development of CANDU PLiM methodologies and implementation of the outcomes at the plants. The purpose of CANDU PLiM programs is to understand the aging degradation mechanisms, prevent/minimize the effects of these phenomena in the Critical Structures, Systems and Components (CSSCs), and maintain the CSSC condition in the best possible operating condition.

The CANDU PLiM program has been applied mainly to mature CANDU plants that have already seen duties of over 60%-70% of their design life such as Point Lepreau and Gentilly 2. The PLiM program has been timely in that results have been used as pivotal information in life extension decisions at these plants. This paper discusses the experiences and “lessons learned” and shows how the application of these proven PLiM methodologies to newer CANDU plants can provide significant benefits. In particular, use of the knowledge of CANDU degradation mechanisms and “stressors” gained at the older plants is important in designing inspection, monitoring and maintenance programs and strategies that can prevent similar degradation from occurring at a newer plant. Also, since not all aging effects can be prevented, plant programs can be “optimized” to detect degradation at the earliest possible time, which will allow degradation rate and component life to be assessed more accurately.

In order to assure high capacity factors during the operating life of the plant and beyond, there is also a major association of system maintenance optimization within the CANDU PLiM program. This involves a program of detailed assessment of critical systems (using Reliability Centred Maintenance based techniques), which is now moving into the implementation phase. In

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order for a plant to take full advantage of this work, ensure the appropriate transfer of system related knowledge over time, and to provide a means for controlled and continued optimization of the plant maintenance program, AECL has developed a System based Adaptive Maintenance Program (SAMP).

This paper outlines how experience gained to date with the CANDU 6 PLiM program methodologies can be used proactively at newer CANDU plants to enhance plant performance and reduce operating costs.

1. CANDU PLiM OVERVIEW

AECL, with CANDU reactor owners, has developed and implemented a comprehensive and integrated CANDU Plant Life Management (PLiM) program that will see the first generation of CANDU 6 plants successfully and reliably through to end-of-design life and possibly beyond this time frame [1]. This program has a focus on critical systems, structures, and components (CSSCs) and is being applied in three phases (as shown in Figure 1): Phase 1 - Assessments, Phase 2 - Life attainment (implementation), and Phase 3 - Plant Life Extension (PLEx) – also known as plant extended operation. The key elements of the program are a) Components and Structures Aging Management, b) Systems Maintenance Optimization, c) Integrated Safety and Performance Assessment, d) Obsolescence Mitigation, and e) Technology Watch. Given the focus on critical assets, a risk-based methodology is used to establish the CSSCs included in the program.

In addition to management of the physical plant, the CANDU PLiM program also recognizes that successful management of other factors is essential to successful performance as the station ages such as: configuration control of the plant, operational management and personnel, and the business, regulatory, and public impact.

Typically, the CANDU PLiM program will modify and enhance, but not replace, existing plant programs that address aging. Many plant programs, such as inspection, maintenance and surveillance already address equipment aging, but it has been found that there are areas where these programs can be enhanced and made more effective for managing aging. Performance measures show that a successful and fully implemented PLiM program is effective in managing aging and provide demonstrated evidence of this effectiveness. This requires a structured and managed approach to both the assessment and, most importantly, the implementation processes.

Establishing a PLiM program is particularly important for a newer plant. International experience has shown that plants that deal with aging early in the operational life will reap later benefits. They will be better positioned to detect passive degradation, and to optimize monitoring and maintenance for degradation of active components. Also the integrity of aging related plant data is a significant concern in the nuclear industry. With the passage of time, retention, integrity, and accessibility of data histories within the plant culture and organization can become issues. Plants that start early on organizing data systematically will be better able to optimize their maintenance strategies.

Many of the critical components assessed in the CANDU 6 PLiM program to date have had very good service records with little or no significant active aging degradation. However, this excellent operating record has provided a unique challenge for the CANDU PLiM program. In performing the systematic and detailed assessments, little plant-specific degradation data has been available and the current history does not provide enough feedback to help understand the potential for future degradation. Hence, a heavy reliance is placed on a thorough understanding of the applicable degradation mechanisms and the associated "stressors". Much of this understanding comes from research and development programs, integrated with knowledge from relevant field data of other plants. Appropriate diagnostic and assessment methods for each physical asset under investigation are then determined. Health prognosis approaches are also used but these are specifically "tailored" to the CSSC component characteristics [2].

A detailed understanding of aging mechanisms enables an in-depth component-specific assessment to be made of the aging risk during plant operation. Component inspection, monitoring and maintenance scope and techniques can then be focused on the specific significant degradation mechanisms and vulnerable locations, and proactive measures can then be taken to slow the aging process. New field data related to aging is regularly fed back to the updating of the reports to improve reliability of the remaining lifetime assessments.

Some examples follow that show how a comprehensive and proactive PLiM program can be used at a newer CANDU plant to enhance the effectiveness of surveillance, inspection, maintenance, and operations programs. By successfully managing aging, a utility can achieve the plant's targets for safety, reliability and production capacity during its planned operational life and preserve options for extended life.

2. PLiM FOR "OPTIMIZED" PLANT OPERATINS AND MAINTENANCE

Effective plant practices in monitoring, surveillance, inspection, maintenance, and operations are the primary means of managing aging. Systematic and comprehensive PLiM assessments provide important Critical System, Structure and Component (CSSC) aging management strategies that enable these plant programs to be "optimized". Two major types of PLiM assessments are Life Assessments and Systematic Assessments of System Maintenance, as described below. Recognizing differences in operating practices and policies between various utilities, the methodology developed by AECL focuses on providing specific guidance for integration into current plant processes and programs. The overall objective is to provide effective aging management, for whatever remaining operational plant life that the utility requires or expects.

Life Assessments facilitate program "optimization" by the use of assessment outcomes to "target" in-service inspection to age-sensitive areas of a component. The life assessment work includes: (a) a thorough review of active and plausible degradation mechanisms, (b) relevant information from operation and design that is related to degradation mechanisms, stressors, and design margins, (c) a detailed risk-based assessment of whether the degradation mechanism is significant and (d) development of what an appropriate age management approach should be. For critical components and structures, this work is at the sub-component level. Inspection results are a key information source, so effective aging management requires inspection at the "right" location, using appropriate techniques. The PLiM life assessments identify the specific areas and sub-component regions within a CSSC that should receive special age management attention.

The in-service inspection effort can then be focused to these age-sensitive locations. For a new plant, early and properly focused inspection information will be crucial in the future for determining whether degradation is occurring, and if so at what rate. This knowledge then can be used to develop cost effective and timely measures to mitigate and manage any degradation that does occur.

Systematic Assessments of System Maintenance and Monitoring facilitate program "optimization" by improving aging management effectiveness, possibly reducing maintenance costs, and can increase plant revenue through higher plant capacity factors, at a new plant as well as at older plants. During the PLiM Assessment Phase, approximately 40 critical systems are subjected to a systematic assessment of the effectiveness of the current programs for monitoring and maintenance. The assessment process is based upon reliability centered maintenance (RCM) analysis techniques and this process has been detailed elsewhere [3]. Briefly, the process includes an identification and categorization of each system function, system functional failure analysis, and a criticality analysis of components in the system. This criticality analysis segregates the system components into categories according to whether failure has a critical impact on the system function, and then determines a maintenance strategy specific to each component. Thus, a comprehensive maintenance strategy for all the system's components and the technical basis for that strategy are provided.

The RCM-based system assessments provide a mechanism to increase the effectiveness of maintenance activities in order to minimize the likelihood of system failure (as opposed to conventional practice of maintenance being solely or largely driven by original equipment manufacturer recommendations). For instance, RCM assessment work on the Main Feed Water System has identified and highlighted the criticality of the feed water check valves and the importance of maintenance and condition monitoring on these valves. This can lead to an avoidance of costs associated with unscheduled maintenance. For example, in this case, it can a) reduce discovery work in an outage and subsequent outage schedule impact, b) reduce risk over the life of the component by ensuring safety functionality and c) minimize the potential for loose parts that could cause damage to downstream components, such as SG tubes.

In addition, in the CANDU system maintenance optimization process, AECL has modified and enhanced the conventional maintenance optimization process. One example that facilitates implementation of the assessment outcomes at the plant is Task Packaging. The recommended maintenance tasks are grouped into packages based on suggested work group responsibility and frequencies. This facilitates plant work definition and program implementation.

There are other efforts currently underway to assist effective plant implementation of the aging management work. System Maintenance Assessments, Life Assessments, and any other related aging assessments, together result in a significant number of recommendations that must be dispositioned, with changes incorporated in the Plant O&M program. These reports contain significant amounts of information that provide the technical basis for maintenance, surveillance and inspection. The PLiM program at a new plant can take full advantage of this experience and technology, ensure the appropriate transfer of system related knowledge over time, and provide a means for controlled and continued optimization of the plant maintenance program. To do this, AECL has developed a System based Adaptive Maintenance Program (SAMP). This program is outlined in section 4 following, shown in simplified form in Figure 2 and detailed in [4].

3. PLiM FOR "OPTIMIZED" SYSTEM HEALTH MONTORING

At many CANDU plants, system engineers are required to develop, implement, and maintain system health monitoring plans (SHMPs). These SHMPs document system function and key components that are required to perform in each system function. These documents also define the monitoring and trending needed to determine system health. The system engineers use the SHMPs to assess system performance against specific goals and to report regularly on system health to station management.

As mentioned previously, AECL has modified and enhanced the System Maintenance Optimization process to facilitate implementation of the assessment outcomes at the plant (e.g. Task Packaging). Another enhancement is the generation of a System Surveillance Matrix (SSM), as a by-product of the maintenance optimization assessments.

The SSM details the monitoring and trending tasks to be performed by the system engineer. The system data required to support the monitoring program is determined from what is needed, not what is available. The data should include direct measurements (e.g. physical parameters, such as flow rate, temperature), and indirect measures (such as number of deficiency reports in a given period). The System Surveillance Matrix is developed by identifying degradation indicators for each system functional failure mode, as identified in the assessment process (technical basis). System parameters, which can be trended to give an indication of the degradation, are identified. Parameter acceptance bands with appropriate actions are given.

Hence, PLiM System Maintenance Optimization is an important element to develop and implement effective programs for system monitoring. Knowledge from these assessments at older plants can be used to both develop and improve effective System Health Monitoring plans at newer plants. Also the outputs of the System Maintenance Optimization assessments, including the System Surveillance Matrix, are important inputs to plant system engineers for their system health monitoring plans.

4. SYSTEM BASED ADAPTIVE MAINTENANCE PROGRAM

A newer CANDU plant can take full advantage of both their own system operational experience and that of other plants with similar systems, by ensuring that their plant programs are "optimized" and regularly updated as they proceed through plant life. However, to take full advantage of this experience requires ensuring system-based health decisions are systematically translated into changes to the component-based maintenance and inspection programs, in an auditable and well-understood process. To accomplish this, AECL has developed a process called SAMP (System based Adaptive Maintenance Program). In this context, "maintenance" involves much more than just the corrective or upgrading activities normally associated with traditional use of the word. It is used here in a broad sense, to cover all maintenance, monitoring and surveillance, inspection and operational (e. g. chemistry) programs that provide information important to aging of the system and its components. Hence, SAMP recognizes that a truly "optimized" maintenance strategy of a system uses the significant inter-relationships that exist between the inspection, surveillance, and traditional maintenance activities, and hence, SAMP facilitates an integrated set of plant activities that together result in a system based maintenance program optimized for the important functions of the system.

SAMP recognizes that there are two primary work cycles used at the plant, both of which influence the overall maintenance program and are used to review experience and update programs. The first is the ***Maintenance Cycle***, which includes all routine tasks that are carried out through a time-based program or on a demand basis. This cycle is primarily focused on components, although some system based monitoring is also included, such as chemistry monitoring. The maintenance cycle includes tasks in support of the condition based surveillance requirements, time based maintenance and inspection activities, corrective maintenance activities, and the relevant reporting activities.

The second cycle is the ***System Health Monitoring Cycle***. This typically involves the "System Health Monitoring Plan" (SHMP), which covers the collection and trending of all surveillance parameters and the generation of the resulting actions. An important activity is the gathering of all relevant system based performance data used for review of the maintenance program.

As these work cycles are executed in parallel at the plant over time, a large volume of data is gathered by the various groups involved in these cycles. To ensure maximum benefit to the overall program, it is important that these cycles be effectively linked, and that relevant data that could influence maintenance and monitoring actions, is shared, trended and brought to the attention of decision makers.

While details of the SAMP process are beyond the scope of this paper (and have been presented elsewhere [4]), it is important to note that SAMP leads to effective flow of information between the various groups who share responsibility for the programs that provide "Aging Management". Hence, SAMP facilitates the interaction of these different groups within the plant structure. Not only do maintenance, inspection, operations and chemistry staff all provide valuable information related to system and component health, but there are also relationships between system specialists, component program specialists, the reliability group, and the groups that provide support to these areas.

To "optimize" the use of plant information, SAMP facilitates the effective sharing of the information, particularly that related to aging. The SAMP process uses a ***System Maintenance Database*** (SMD), as the key vehicle to relate the various programs into the overall system maintenance strategy. The SMD is structured to contain the "current" detailed maintenance strategy (such as those generated from Maintenance Optimization studies), which is then maintained and updated on a periodic basis. For the periodic review, component performance in the system during the past period, and recent experience in execution of the surveillance and periodic maintenance activities, are considered. Additional information may also be added, such as:

- New safety or regulatory requirements or concerns
- New or improved methods or strategies based upon field experience
- External information regarding system condition such as Life Assessments
- New concerns arising from change in system operational constraints
- Identification of new maintenance or predictive maintenance technology
- Identification of new approaches identified from external sources

This data is subject to a documented review, approval and disposition process, involving key groups. The SMD is updated and changes to the related plant programs are then triggered. It should be noted that the SMD is the hub for PLiM Implementation at the plant, as the age-related inputs to the various plant programs all reside in the SMD. As maintenance moves towards condition based activities, the success of this type of “optimized” program relies upon effective flow of the right information to the system or component engineer who takes action from the condition information. SAMP facilitates this process within the plant organization and uses computerized information management to enable effective management of all the predictive, preventive or corrective maintenance programs. Together these optimized programs then become the aging management program.

If the maintenance program is adapted through the SAMP process, plant owners have assurance that the equipment will be effectively managed, and the maintenance/monitoring program will be continually and effectively optimized to provide for maximum availability and reliability, while optimizing O&M costs. Supported by corresponding inspection and test programs, this process will serve as a reliable predictor of future plant performance, provided the recommended aging management practices are followed.

5. PliM IMPLEMENTATION, PLANT STAFF INVOLVEMENT AND KEY SUCCESS MEASURES

In performing the PLiM Assessment work and in PLiM Implementation, effective interaction with utility staff has been important. An in-depth understanding of the plant operational history and the current plant programs related to aging are both key inputs to the PLiM assessment process. Also, involvement of utility staff in the process provides better understanding of the assessment outcomes, which in turn assists in the implementation of assessment recommendations. Developing an efficient and effective interface with key utility staff (such as aging management experts, system engineers, component engineers, reliability engineers and maintenance personnel) has contributed to the success of the PLiM assessment program. Some of the key interfaces between the various groups in the PLiM Assessment Phase are shown in Figure 3. Other groups are involved in the assessment work, such as equipment suppliers and inspection/maintenance service providers.

System maintenance optimization assessments, in particular, involve transfer of the results, and the supporting technology, to plant staff who then utilize it and keep the work current during future plant operational life. While plant staffs, particularly system engineers, are involved in the initial assessments, the transfer of the technology for the on going updating does involve some new skills. Training packages have been developed to facilitate effective transfer. While some other PLiM assessment activities such as Life Assessments can be viewed as having a one-time impact, an enhanced value to the plant is obtained when these studies are also considered for improvements as further plant-specific experience is obtained. Again, training of plant staff in aging degradation mechanisms and assessment techniques is important in adapting the program.

Implementing a PLiM program will result in demonstrated and measurable benefits. A set of Key Success Factors can be defined for use by plant management to measure PLiM performance. While there is not one "single" set of PLiM success measures, a set of generic performance indicators has been developed and assessed for potential use, based on the PLiM program experience. For example, an effective PLiM program will ultimately improve capacity factors by reducing the number of unplanned shutdowns. PLiM can also be a major force in optimizing and even reducing plant OM&A costs when applied properly and early in the life of a plant. This involves methods of prioritizing recommended actions prior to implementation.

6. SUPPORTING ADVANCED PLANT MAINTENNCE INFORMATION MONITORING AND CONTROL

Effective use of age-related information at the plant and the timely flow of this information to key decision makers becomes a greater challenge to manage as maintenance strategies move to more condition-based decision making. To facilitate this, AECL is developing the ***Maintenance Information, Monitoring, and Control (MIMC)*** system. The MIMC tool provides an interface for users to access various health monitor information and also acts as an electronic portal to the maintenance review and updating as well as to the work management system.

The requirement for ensuring that maintenance information is sent to the appropriate personnel can also be handled via MIMC. For instance, MIMC has a means build into it to assign information received from the Work Management System on a completed work order, to the personnel that need to see the information. For instance, for Condition Based Maintenance to be successful, the information must be received by the right person and then acted upon. If the action is to be initiated by specific personnel, the tool needs to monitor and act if that person does not acknowledge the receipt of the information. It must also ensure that the information is not orphaned e.g. no one reviews it or takes action, if needed. MIMC addresses these capabilities.

Another information flow related function is providing feedback. MIMC can be augmented to ensure that certain types of information, such as suggested changes to a maintenance task, are acknowledged and indicate the follow up action taken. In this case of the suggested change, feedback would indicate that the suggestion was forwarded for consideration in updating the SMD.

AECL is also developing a number of advanced system health monitoring tools to aid the effective aging management of CANDU plants. One of these, called ***ChemAND***, is an on-line chemistry monitoring and diagnosis system. An application of ChemAND is to aid in the achievement of maximum steam generator life, where this tool is used to identify the effects of impurities in the secondary side water on local steam generator crevice chemistry and fouling. The ChemAND system has undergone extensive field trials at Hydro Quebec's Gentilly 2 nuclear power plant and is available for other plant applications. On-line access by the operators to current including these crevice regions of the SG, enables appropriate responses while on-line and also facilitates planning of shutdown maintenance actions (such as cleaning specific areas). This is a key to successful management of health and long life on this critical plant component.

7. CONCLUSION

In summary, a comprehensive and integrated Plant Life Management (PLiM) program has been developed and is being implemented at several CANDU plants at mid-life. The program is helping CANDU reactor owners achieve goals for safe, economic and reliable operation for design life and beyond. The PLiM Assessments have provided utilities with effective aging management actions that are being used to refine current programs. When implemented they provide optimized, cost effective and well-documented programs for system monitoring, maintenance and inspection for life attainment and plant life extension. The focus of the PLiM program is currently on the effective implementation of the analysis results.

For newer CANDU plants, the PLiM experience at the older CANDU plants can provide many benefits The knowledge of CANDU degradation mechanisms and “stressors” gained at the older plants can be used in designing inspection, monitoring and maintenance programs and strategies. They should thus be capable of preventing similar degradation mechanisms from occurring at the newer plants, if possible. Since not all aging effects can be prevented, a further objective is to “optimize” plant programs to detect any significant degradation at the earliest possible time, should it occur, and thus to determine degradation rate.

In order to assure high capacity factors to the end of design life and beyond, there is also a major effort on system maintenance optimization integrated within the CANDU PLiM program. This system maintenance optimization program is now moving into the implementation phase. In order for the plant to take full advantage of these studies, ensure the appropriate transfer of system related knowledge over time, and to provide a means for controlled and continued optimization of the plant maintenance program, these studies need to be incorporated into a System based Adaptive Maintenance Program (SAMP).

SAMP provides a means to ensure system-based health decisions are systematically translated into effective changes to the component-based maintenance and inspection programs, in an auditable and well-understood process. The review and updating of the System Maintenance Database (SMD) is a key part of the SAMP process. If the maintenance program is adapted through the SAMP process, plant operators have assurance that the equipment will be effectively managed, and the maintenance/monitoring program will be continually and effectively optimized to provide for maximum availability and reliability, while optimizing O&M costs.

In addition, other PLiM implementation tools, procedures, performance measures and training are available for use in assisting a newer CANDU plant benefit fully from these programs at an early stage in the plant’s operating life.

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REFERENCES

[1] J. H. Nickerson et al, “An Overview of Plant Life Management and Life Extension Plan for CANDU 6”, paper presented at SMIRT 16 Conference, Washington, D.C., 2001 August.

[2] J. H. Nickerson at al, “CANDU™ Plant Life Management Assessments supporting Life Extension”, presented at the 2001 PLIM+PLEX Conference, London, England, 2001 November.

[3] R.A. Richard et al., “Streamlined Reliability Centered Maintenance (RCM) Application to CANDU 6 Stations”, presented at Fifth CNS International Conference on CANDU Maintenance, Nov. 19-21, 2000, Toronto, Canada.

[4] R. F. Dam et al, "The CANDU Experience with Reliability Centred Maintenance and Plant Life Management", paper presented at IAEA Specialist's Meeting on Optimization of NPP Maintenance Programs, Charlotte, NC, USA, June 2001.

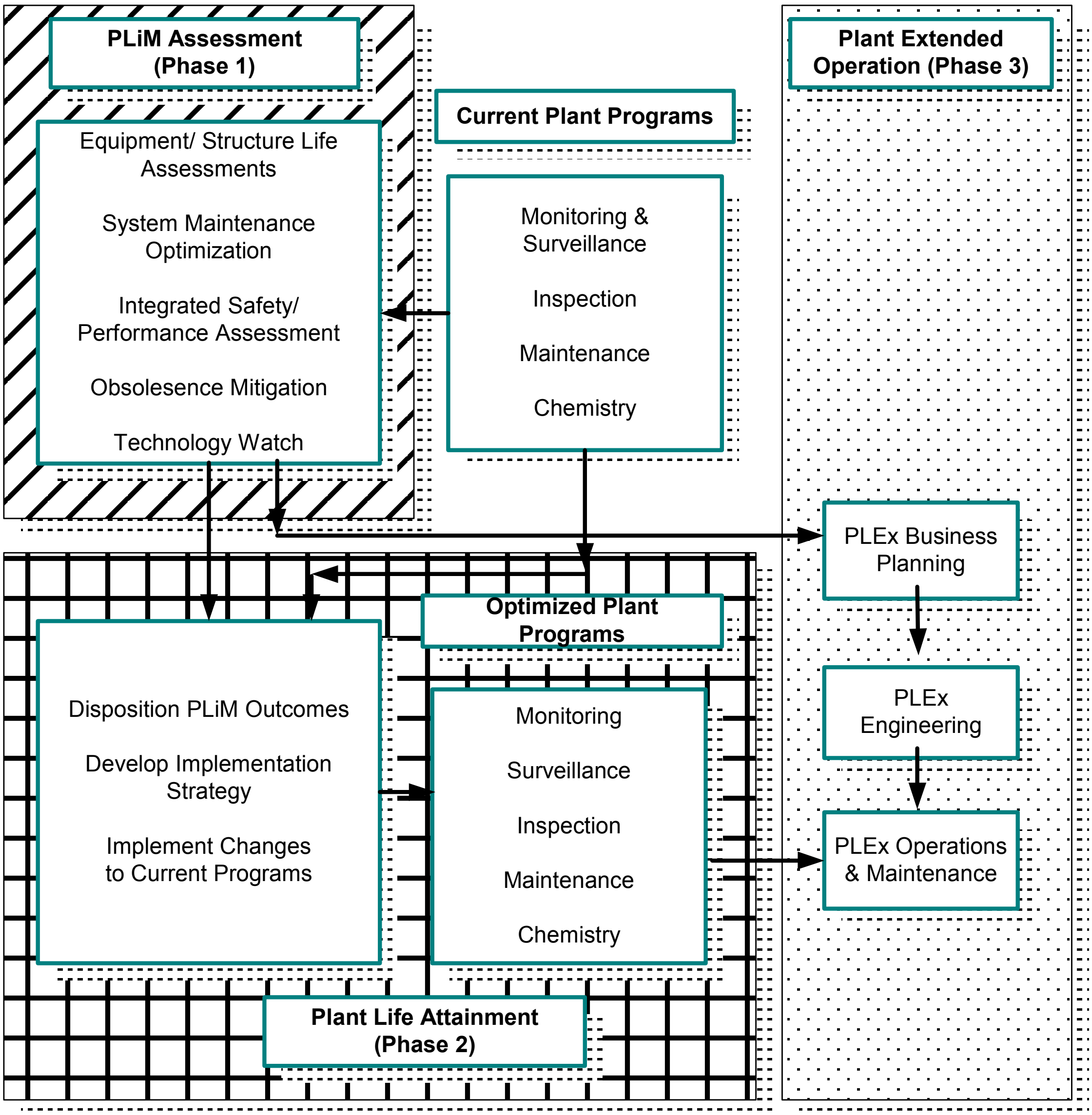

FIG. 1. Simplified CANDU PLiM Process and Phases.

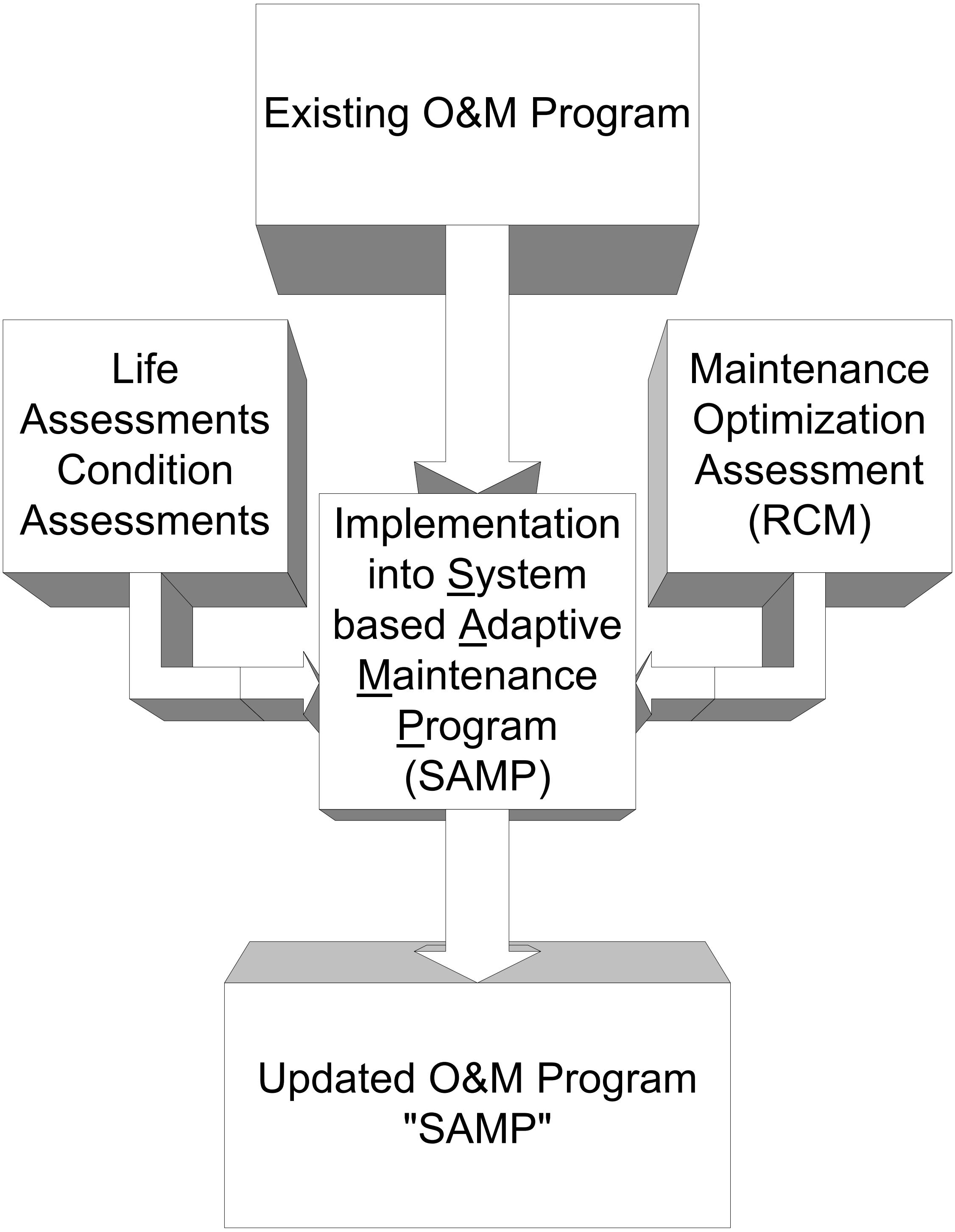

FIG. 2. Implementation of PLiM Assessments into Plant Programs using System based Adaptive Maintenance Program (SAMP).

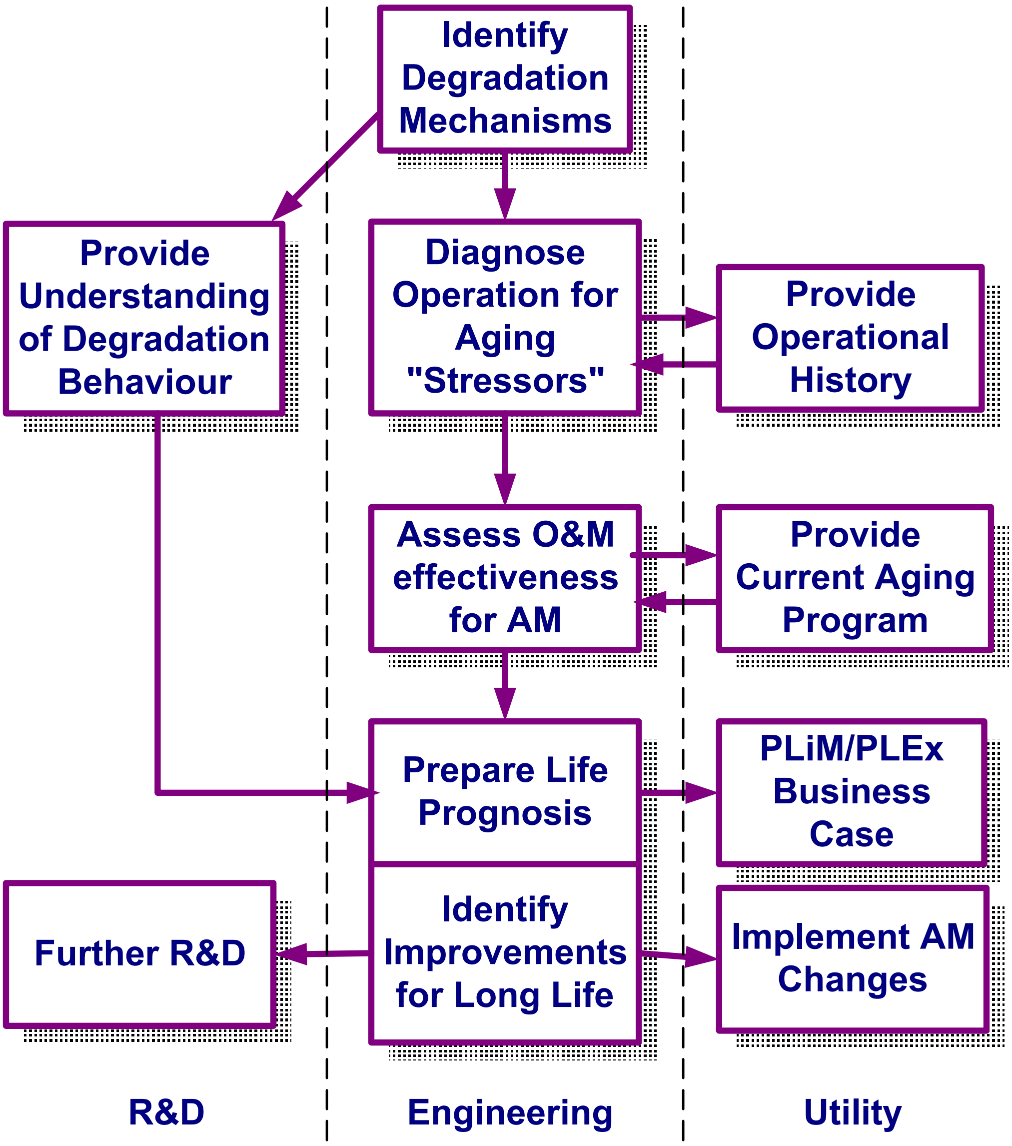

FIG. 3. PLiM Assessment Interfaces.

AGEING SIGNIFICANTLY AT TEMELIN STARTUP PROCESS

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Abstract

NPP Temelin is a unique case in Plant Life Management Programme application as it is now only in the period of start-up and thus all measures can be easily prepared and implemented in advance based on the experience of similar plants already in operation.

Detailed analyses of potential damaging mechanisms as well as of initial conditions and properties of main components have been already done. Regarding this analysis, several measures have been prepared and put into operation:

- List of important parameters to be measured, saved and periodically analysed have been prepared – some of these parameters are measured in the framework of standard I & C system; new sets of sensors were installed for some additional ones.
- Special software for on-line evaluation of stress/temperature conditions and subsequently also fatigue damage in some chosen parts (with flow stratification etc.) have been installed and during reactor start-up tests have been checked.
- Surveillance specimen programme for reactor pressure vessel material monitoring was modified for better and fuller characterisation of material changes during operation.
- Ex-vessel neutron flux measurement has been established as a part of Surveillance specimen programme.

The whole Plant Life Management Programme is governed by a special PLIM Group in the plant with necessary links and responsibilities and rules with other necessary plant departments.

1. INTRODUCTION

In NPP Temelín, Czech Republic, two units of WWER-1000 MW are built. These units are of PWR type (WWER = Water-Water Energetic Reactor) with 4 loop reactors designed in the former USSR (design 1982) by LOTEP Moscow.

Building of the NPP started in 1986 with a lot of modifications. Start-up of the first unit was in 2000. Now, the first unit is in test operation mode while the second one is in start-up programme on 30 % power level.

2. REQUIREMENTS ON AGEING PLNT MANAGEMENT IN TEMELIN.

According the Czech State Office of Nuclear Safety (SONS) decision dated to 1992 an APM on reactor pressure vessel as well as later a similar programme on cables relevant to nuclear safety was required. Some recommendation and world experience (WANO) was taken into account to form an APM program. No APM for other equipment is directly required by the SONS. Therefore the following activities have been realised:

- The NPP Temelin has prepared an APM programme for Reactor Pressure Vessel-RPV (1993-1997) and CEZ company programme for cables (1997- 2002), respectively

- additionally, the NPP prepared such programme also for Steam Generator (SG) and pressuriser

- additionally, it was also decided to prepare a complete programme for other equipments of primary and secondary circuit (such decision was involved in internal rules -25.05 Ageing Management). Together, 18 equipments/components are included into this programme. See attachment of this paper.

- SONS is preparing some Guides for APM of RPV, pipelines and other equipments.

- according to the world experience, the NPP tries to follow IAEA Guides as like 50-SG-08 or more detailed documents as TE 00C-932 “Pilot study the management of ageing of instrumentation and control cables”. Consultations and meetings on WANO level were also valuable.

3. MAIN ACITIVITIES IN THE NPP FOR AMP

- NPP Rules -25.05 “Ageing management” have been prepared and put into practice for NPP Temelin. This rules specify the basic activities, responsibilities, liability and information flow for providing such programme. The part of this document is also a list of equipments/components relevant to the APM programme.

- NPP has ordered an elaboration and also managed a detail programme for the most important components - RPV, SG, pressuriser, heat exchanger for reactor ECCS cooling, cables (see the attachment)

- NPP has been preparing the APM contracts for the rest of relevant equipment

4. THE CURRENT SITUATION

In the paper the unique detailed equipment programmes state is described. As an introduction, the common structures of each programme is described:

- assessment of present state (manufacturing, material, assembly, start-up programme, documentation, calculations etc.)
- analysis and calculations - input data according to the design
- analysis and assessment of real situation (experience, operation, measurement, etc.)
- selection of major degradation mechanisms /stressors and the methods of their evaluation.
- methods of ageing evaluation
- measures to mitigate negative effects of ageing in design, monitoring, start-up testing and maintenance

4.1. Reactor Pressure Vessel (RPV)

RPV is a product of Skoda-Nuclear Machinery, Plzen. This is a product according the Soviet licence. The ageing programme has been prepared by the Nuclear Research Institute Rez (UJV Rez).

The main conclusions of this programme are:

- The used current design and manufacturing documents are comparable with western standards (ASME, etc.)
- Effect of irradiation embrittlement on RPV materials is acceptable as well as an effect of irradiation embrittlement on in-core components.
- Surveillance specimen programme has been innovated (temperature and neutron flux measurement, location of specimen containers, reference materials, etc.)

Unirradiated surveillance specimens programme is now under testing and evaluation gives positive results for RPV integrity assessment and AMP realisation

4.2. Control Roads Actuators

These actuators will be replaced by more modern ones, thus this is not a topic nowadays. On-line Diagnostics system for CRDMs is installed to monitor them. *This system is able to monitor and count number of steps, number of lost steps, currents, speed of scram, etc. On the basis of these values it is possible to estimate the condition of these actuators.*

4.3.Main Coolant Pump (MCP)

These pumps were manufactured by soviet manufacturer M.B. Frunze *CKBN Petersburg*. Some stress analysis and fatigue calculation has been done. The APM programme elaboration has been running and some first conclusion is available. On-line Diagnostics systems MAFES (*temperature and pressure cycles)* and RECOP *(vibration)* are installed to monitor their behaviour.

The most danger operation mode has been estimated to be an assembly and disassembly of the main flange and the pressure tests. Packing surface of this flange is critical. The fatigue damage after start-up period reaches about 40% of the value for the whole lifetime.

4.4. Main Coolant Pipeline (MCPL)

This pipeline is a product of Modranska Potrubni, Praha, also according to the soviet licence. Some stress analysis and fatigue calculation has been done. On-line diagnostics systems MAFES and ACMS (*acoustic emission)* are installed on it. Some connected pipelines are analysed too.

Almost 26 regions have been selected to further analysis and investigation. They mostly contain nozzles of connected pipelines of different systems. The SG feed water nozzle and the part of surge line connection to MCPL were estimated as critical parts. This is in accordance with MAFES design assumption. MAFES is able to calculate the fatigue in 9 „sections“ including these two.

As the most danger operation modes has been estimated the feed water supply at low power of unit which causes an interrupted feeding, irregular pressuriser injection and stratification in surge line.

In spite of these phenomena, the fatigue rate (usage factor) is not excessively high and after start-up period is about 3% as a maximum.

4.5. Steam Generator and Pressuriser

Manufacturer of both equipments is Vitkovice, Ostrava that manufactured them on the base of soviet licence. Some design and material changes was involved in SG. The author of APM programme is a research institute UAM Brno. The main conclusions are:

- The used current design and manufacturing documents are comparable with western standards (ASME, etc.)
- The innovation of SG has been verified during the start-up measurement and APM programme.
- The diagnostics systems MAFES and ACMS were modified
- Additionally, some NDT tests should be performed

As a dominant stressor, corrosion is expected. A new R&D programme has been prepared by UAM Brno with the governmental support.

4.6. Pressuriser Release Valve

APM programme for this component will be cancelled. *This valve should be replaced in several years.*

4.7. Reactor Heat Exchanger of ECCS (SAOZ)

It is a product of KSB, Brno. An APM programme by TEDIS.KOR company has been provided and simultaneously monitoring of temperature cycles and strength will been finished in this year as a part of start-up testing.
The first conclusion is, that the heat exchanger operates in harmony with design assumptions.

4.8. Containment

This building system was built according to the soviet design and licence by Czech company Vodni stavby Bohemia. APM programme has been running.

4.9. Steam Pipeline

This equipment was manufacture by Modranska potrubni, Praha. Some calculations (LBB) have been performed for piping inside containment part. Some temperature and displacement measurement is installed on it as a part of secondary circuit diagnostics system. APM programme has been running.
This programme is involved in pipeline ageing programme.

4.10. Turbine Condenser

The heat exchanger brass tubes were replaced by titanium ones. In all secondary circuit, pH value has been increased up to 10.8 to 11.2. Therefore APM programme has been cancelled.

4.11. Turbine

The turbine is a prototype product of Skoda Plzen/Turbine Division. There is some co-operation on turbine enhancement on CEZ company level. The NPP has an intention to elaborate a base APM programme and after result of this *to decide about a final APM.*

4.12.Main Coolant Water Supply Pipeline

There are two points of view on this problem. The older one has been based on a suspicion of erosion/corrosion existence. Therefore, an additional inspection programme containing wall thickness measurement on some spots was involved. Some critical part of this approach has been based on zinc's layer. In spite of this the second opinion, the erosion/corrosion analysis (CHECKWORD programme) in this component is not estimated as a critical one. Thus, evaluation of the real situation will be further continued.

4.13. Cables

The estimation of cables ageing is already assumed in cable's traces design for two main groups - I&C and higher voltage cables. An APM programme on CEZ company level has been evaluated and first analysis and conclusions are known. The document IAEA-TEDOC-932, March 1997" Pilot study the management of ageing of instrumentation and control cables" is also taken into account On these bases, some cable family representatives for surveillance specimen programme for accelerated and artificial ageing have been chosen. The laboratory artificial ageing tests are performed in UJV Rez (using Co^{60} radiation field and operating temperature). Deposits/holders for cable surveillance specimens were built inside the containment.
First results from environment condition measurement are optimistic, because of the temperature, which is not higher, than 50°C even in the vicinity of MCPL.

4.14. Diesel Generator

Programme for this equipment was additionally involved into the APM. There is a support diagnostics measurement installed on DG, but there is not yet a good correlation with the anticipated ageing problems. A base APM is running.

5. CONCCLUSIONS

Ageing Plant Management Programme for NPP Temelin has been prepared in advance of its start-up. Information from manufacturers together with recommendations from IAEA Guides and the experience and knowledge from technical support organisations has led to a choice of critical equipments from point of view of potential ageing mechanism effects on equipment life and operability. For these equipments, individual APM Programmes have been prepared. The programmes realisation is in many cases supported by special diagnostics measurements, and, of possible, also by surveillance specimen programmes and by additional NDE testing as a part of in-service inspection programmes.

It is expected, that on the basis of evaluation of measurements realised during start-up programmes of both units, the APM programmes will be précised and/or modified, if necessary.

Attachment: list of equipment/ components related to APM programme on NPP Temelin

Equipment Name	Design Assignment	Manufacturer	Life Expectancy (years)	Status of APM programme preparation	Prepared by
Reactor Pressure Vessel	YC00B01	ŠKODA	40	contract is finished - existing	ÚJV
In-core Reactor Parts	-	ŠKODA	30	contract is finished - existing	ÚJV
Control Rods Actuators	-	ŠKODA	20 10 actuator	not applied	-
MCPL	YA11,21,31,41Z01	Potrubí Modøany	30	contract started in 12/99	UAM
Pressuriser	YP10B01	Vítkovice	30	contract is finished - existing	ÚAM
Surge Line	YP10Z01	Potrubí Modrany	30	contract started in 12/99	UAM
Pressuriser Release Valve	YP22,23S01	SEMPELL	30	not applied	-
MCP	YD10,20,30,40D01	USSR	30	contract started in 12/99	Dystiff
MCP - Rotating Part	-	USSR	30	contract started in 12/99	Dystiff
SG	YB10,20,30,40B01	Vítkovice	30	contract is finished - existing	ÚAM
Heat Exchanger of ECCS (SAOZ´)	TQ10,20,30W01	KSB	30	contract is running	TEDIS.KOR
Containment	-	VST	30	contract has been started in 12/99	CVUT
Steam Pipeline	TX50,60,70,80Z01	Potrubí Modrany	30	contract started in 12/01	UAM
Turbine Condenser	SD01,02,03W001	SES Tlmace	30	not applied	-
Turbine	SA00D001, SA01,02,03D001	Škoda Plzen	30	base programme will be applied	Škoda
Main Coolant Water Pipeline	VF10,20,30	KSB	30	programme is running	UAM
Cables	-	Kablo, and diff.	30	contract on CEZ level is running	UJV
Diesel Generator	PS 0.44, 1.44, 2.44	CKD		base programme is running	CKD

COMPONENTS SELECTION FOR AGEING MANAGEMENT

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Abstract

The paper presents a synthesis of methods and activities performed for the selection of critical components to assure plant safety and availability (as electricity supplier). There are presented main criteria for selection and screening process. For the resulted categories of components (critical components, non critical for safety, run to failure), shall be applied different category of maintenance (condition oriented, time oriented or corrective), function of the importance and financial effort necessary to fulfill the task.

1. GENERAL ASPECTS

In the general program of Ageing Management (Fig. 1), an important step is Critical Components Selection.

During operation these critical components shall be monitored, their degradation trends evaluated and remain service life established. Based on it, shall be established the requirements for maintenance.

For the rest of the components could be applied maintenance based on the manufacturing requirements (time oriented) or corrective maintenance (when component fails).

2. EFFECTIVE COMPONENTS SELECTION

Normally, critical components selections take in to account two NPP main important aspects: assurance of Safety and of Availability. But there are also other aspects, which could be considered, such as: NPP life assurance, decreased number of man-rem for maintenance, reduced operating costs. Here are considered only safety and reliability aspects.

First step in the selection begins with the considerations of main systems necessary to fulfill the above tasks, based on system functions in the plant. This is illustrated in systems Design Manuals, Operating Manuals, Accident Analysis, Reliability Analysis, Probabilistic Risk Assessment, Basic Nuclear Safety functions to be assured for the plant.

Following step is to establish support systems necessary to assure proper function of the main systems, till primary systems (as: Danube / infinite heat sink, National Electrical Grid, fuel supply).

This assembly of systems represents Plant Critical Systems.

Finally, components selection is taking place from the systems selected in previous steps. For every system there are considered following elements:

- The effect of components failure on the system task considered
- Is component degradation a potential cause of component failure?
- How adequate are normal operating procedures (reflected in Plant Surveillance Program / fig. 2) and maintenance procedure to timely detecting and stopping of component degradation?

In general, to avoid too much extension of the analysis for critical components selection, where excluded from the beginning: small lines (e. g. under 50 mm, due to their local effects in case of rupture), drainage and vent valves, manual valves (important valves are with actuators).

2.1. Selection from the Safety point of view

For the systems selection, from the Safety point of view (Fig. 3), was necessary, first, to define systems which are dangerous in case of failure (mainly by rupture and release of radioactivity) and the safety systems, which have to mitigate the effects. This is done based on accident analysis (from Safety Report).

Also there were taken in to account the assurance of the basic Safety Principles:

-Reactor shut down;
-Residual heat removal;
-Radioactivity products confinement;
-NPP status monitoring (in normal and accident conditions).

Following step is to establish safety support systems, which have to assure main safety systems operation. This could be done based on engineering judgement, or on PSA Level I analysis.

Finally there shall be completed chains of the support systems, which have to work, till primary systems (e.g. Fig. 4).

For the critical components selection, a Failure Mode and Effect Analysis (FMEA) was performed, considering the effects of components failures on system safety function (Table 2). The result of this analyse is in Table 3.

One important thing is that safety systems critical components are normal in stand-by condition.

Of course components selections could be prepared based on PSA analysis and ranking of them function of importance. It was not used extensive for Cernavoda, due to high cost of iterative calculations of the plant risk (also the Cernavoda reliability Data Base is at the beginning).

2.2. Selection from Availability point of view

The basic condition, for Cernavoda NPP (from availability point of view) is, 100% Power.

One important aspect is that systems and components important for plant availability are (with some exception/ as Fueling Machine) in continuous operation.

Based on this condition, the basic chain of the main systems (Fig. 5) was established.

The second step was to determine the support systems, which assure main systems operation. This was realized using a system evaluation file (see Table 1). There were considered components repair possibilities, and access conditions (temperatures, radiation fields, space available) in normal operation. Also there were considered possibilities of system isolation, during reactor operation.

Final step is components selection based on a Failure Mode and Effect Analysis, using a special analyze file (see table 2). Table 3 presents some critical structures and components resulted from analysis.

This table emphasize that there are many components important for the plant availability, which are also critical from safety point of view.

2.3. Representative components

A complete list of selected critical components is very large. Monitoring and evaluation of this amount of components is expensive. In the same time there are similar critical components, which work, in the same conditions.

For this reason, from the similar components shall be selected one or two, considered representative and conclusions, taken on them, extended to others (e.g. pumps, valves, lines, I & C components). For lines only the main lines and the most exposed part of them shall be considered (e. g. to: erosion-corrosion, water hammer).

Could be also a lot of I & C components, for which it is possible to have, from the manufacturer, the ageing rates (function of temperature, radiation level and time). Based on them, the respective components ageing should be evaluated using environmental conditions recorded at the place of installation

For the I & C components, at which there are not available ageing rates (e.g. old one or obsolete), but which are in large quantity in the plant (as spare parts), one possibility is to put sample of them in harsh conditions (2 –3 times stronger than in normal operation) and extend conclusion to other similar.

Another way to increase knowledge about ageing and in the same time to reduce the number of monitored components in the plant is to use:

- The results of the Environmental Qualification program.
- The results of analysis on the components removed from the plant to confirm that selection and evaluations where properly performed. This should compensate, in part, the expensive pilot studies for that type of components.

3. CORRELATION WITH MAINTENANCE CATEGORIES

A general view of the Maintenance Categories and correlation with different categories of components is presented in Fig. 6 and 7.

The critical components, from plant Safety and Reliability point of view (which results from the selection process), have to be monitored carefully, such as to determine their degradation trends and remain service life. The remain service life indicate WHEN the component have to be repaired and WHAT part of it. In this way, for them, shall be applied a maintenance based on knowledge of real components status (Condition Oriented Preventive Maintenance/ COPM). Fig. 8 present an example of the maintenance moments establishment, in this way, for the critical components of a system.

Components which are not critical from plant Safety and Availability could be considered as Run to Failure Components and applied for them the Corrective Maintenance.

However, for the components which are critical from Safety and Reliability, but for which is enough to apply manufacturer maintenance requirements, shall be realized Time Oriented Preventive Maintenance (TOPM), or replacements (monitoring and analyze is too expensive and without proven results, comparative with the cost of TOPM or replacement)

For components which are relative important only from plant availability, but which could be repaired relative easily during plant operation, or is economically more efficient to be repaired with the reactor shut- down, shall be applied Corrective Maintenance. Figure 9 present an example of ranking, by priority, of Time Oriented Preventive Maintenance and Corrective Maintenance activities for the different systems components.

General components selection for the application of different kind of maintenance shall be revised periodically and optimized based on the results in operation (as part of the Reliability Centered Maintenance program).

REFERENCES

[1] CANDU Safety Principles and Design Criteria (AECL-TDS)
[2] Cernavoda U1 N.P.P. Final Safety Report (2001)
[3] IAEA TECDOC 540 “Safety aspects of NPP Ageing “ (IAEA 1990)
[4] IAEA Workshop on Ageing Management for Cernavoda N. P. P. (November 2001)

Table I. System evaluation file for screening from availability point of view

1. **System**
2. **Functions (to assure plant operation, at nominal power).**
3. **Operating conditions: continuous, periodically (number of hours), when is necessary….**
4. **System functional failures (on different functions):**
5. **System interfaces:**
   - **5.1 With the systems on which system analyzed has influence: (immediate effects; effects after a period of time/ which period)**
   - **5.2 With the systems which influence system analyzed activity: (immediate effects; effects after a period of time/ which period)**
   - **5.3 With other process systems: (system/ reasons)**
6. **Separation between system and the interfaces: System/ valves/ position in normal plant operation/ layout/ access possibilities (in normal plant operation/ environmental conditions, including radiation fields)**
7. **System main components layout (room, environmental conditions, including radiation fields)**

Table II. Structure of the file for the analyze of the components from a system

X System functional failure

XX Component involved

XX1 Component Task
XX2 Main failure mode of the component
XX2 Effects:
XX21 Local
XX22 At the system level
XX23 At the plant level
XX3 Failure rate
XX4 Demand (hours per year)
XX5 Failure probability
XX6 Conclusion (specify if component is selected as critical or not)

XX

X

Table III. Example of critical structures and components

Critical structures and components	**Important for safety**	**Important for the plant availability**
Reactor Building (R/B)		
Containment	**X**	
Airlocks	**X**	
Reactor building internal structures	**X**	**X**
Service Building structure	**X**	**X**
Secondary Control Area	**X**	
Spent fuel bay	**X**	**X**
Turbine Building structure	**X**	**X**
Turbine Generator structure	**X**	**X**
Main cooling water pump-house	**X**	**X**
Emergency Water Supply structures	**X**	
High Pressure Emergency Core Cooling structure	**X**	
Reactor, PHT and Auxiliary systems		
Calandria	**X**	**X**
Pressure tubes	**X**	**X**
Feeders	**X**	**X**
PHT lines	**X**	**X**
PHT circulating pumps	**X**	**X**
Steam Generators	**X**	**X**
Pressurizer	**X**	**X**
D_2O feed pumps	**X**	**X**
Fueling machine head	**X**	**X**
Moderator system		
Main lines	**X**	**X**
Heat exchangers	**X**	**X**
Pumps	**X**	**X**
Reactivity control mechanism	**X**	**X**
Safety systems and support systems		
Dousing system valves	**X**	
ECCS lines,	**X**	
ECCS pumps	**X**	
ECCS valves	**X**	
ECCS heat exchangers	**X**	
R/B Local Air Coolers	**X**	**X**
R/B Ventilation fans	**X**	**X**
R/B Ventilation filters	**X**	**X**
Emergency water supply pumps	**X**	
Emergency water supply lines	**X**	
D_2O vapory recovery dryers	**X**	**X**
Containment isolation valves	**X**	
Main Control Room air conditioning fans	**X**	**X**
Spent fuel bay coolers	**X**	**X**
Cl. III Diesel Generators	**X**	
Emergency Power Supply Generators	**X**	
Recirculating water lines	**X**	**X**
Recirculating water pumps	**X**	**X**
Recirculating water heat exchangers	**X**	**X**

Demineralized water pumps		**X**
Chillers	**X**	**X**
Secondary circuit		
Main steam lines	**X**	**X**
Main feed water lines	**X**	**X**
Main Steam Safety Valves	**X**	
Maim feed water pumps	**X**	**X**
Auxiliary feed water pump	**X**	
Condenser steam discharge valves	**X**	
Degasser tank	**X**	**X**
Reserve feed water tank	**X**	**X**
Moisture separators and reheaters		**X**
Turbine Generator	**X**	**X**
Condenser	**X**	**X**
Main condensate pumps		**X**
Reheaters		**X**
Feed water regulating valves	**X**	**X**
Auxiliary Systems		
Service water screen	**X**	**X**
Service water pumps	**X**	**X**
Service water lines	**X**	**X**
Fire protection pumps	**X**	
Fire protection main lines	**X**	
Main output transformers		**X**
Plant service transformers	**X**	**X**
Representative cables	**X**	**X**
Representative instrumentation loops	**X**	**X**
Batteries	**X**	
Inverters	**X**	**X**
Rectifiers	**X**	

FIG. 1 GENERAL PROGRAMME FOR COMPONENTS AGEING MANAGEMENT

(CITON proposal for Cernavoda U1)

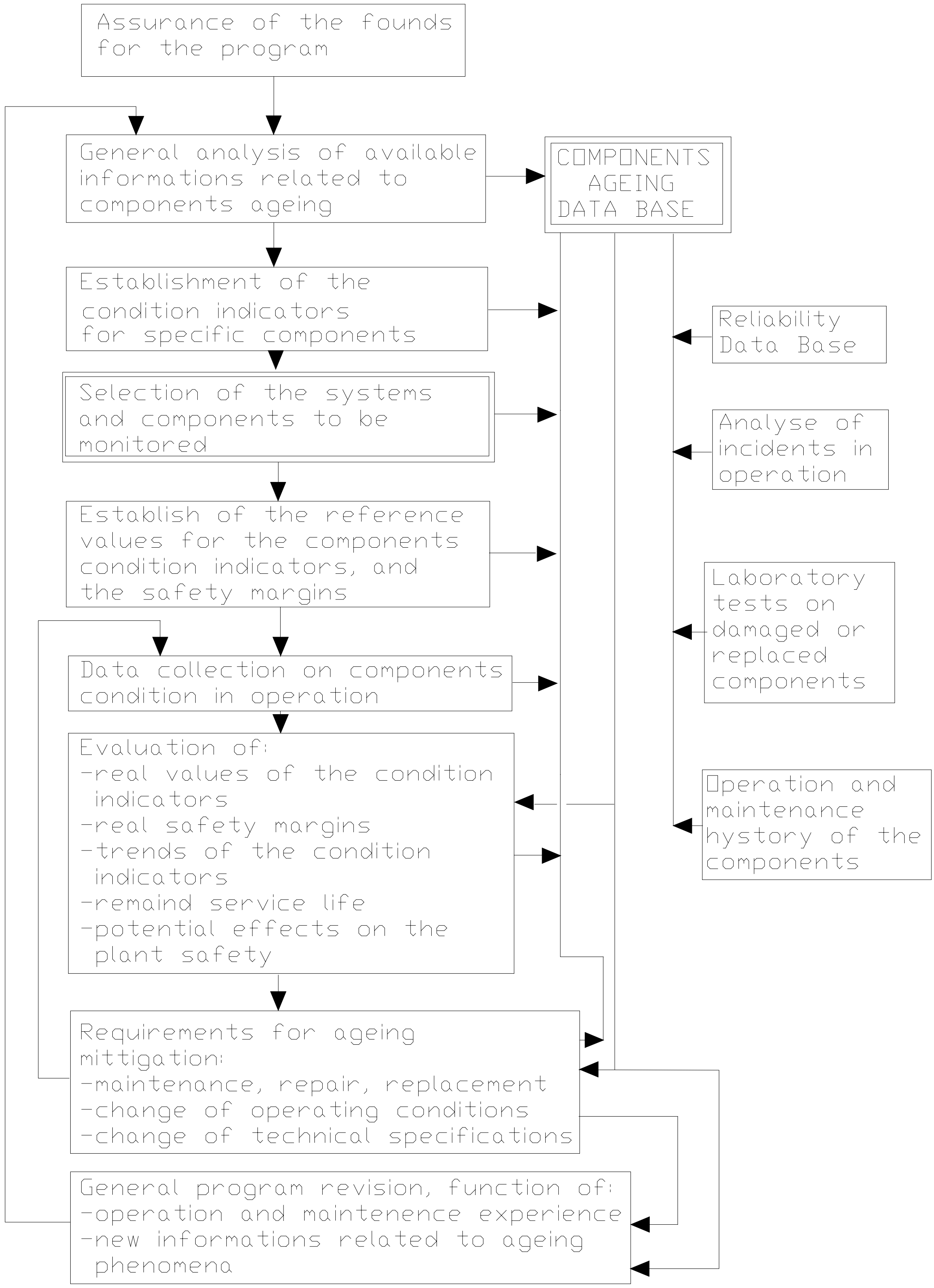


Fig. 2 GENERAL PLANT SURVEILLANCE PROGRAM

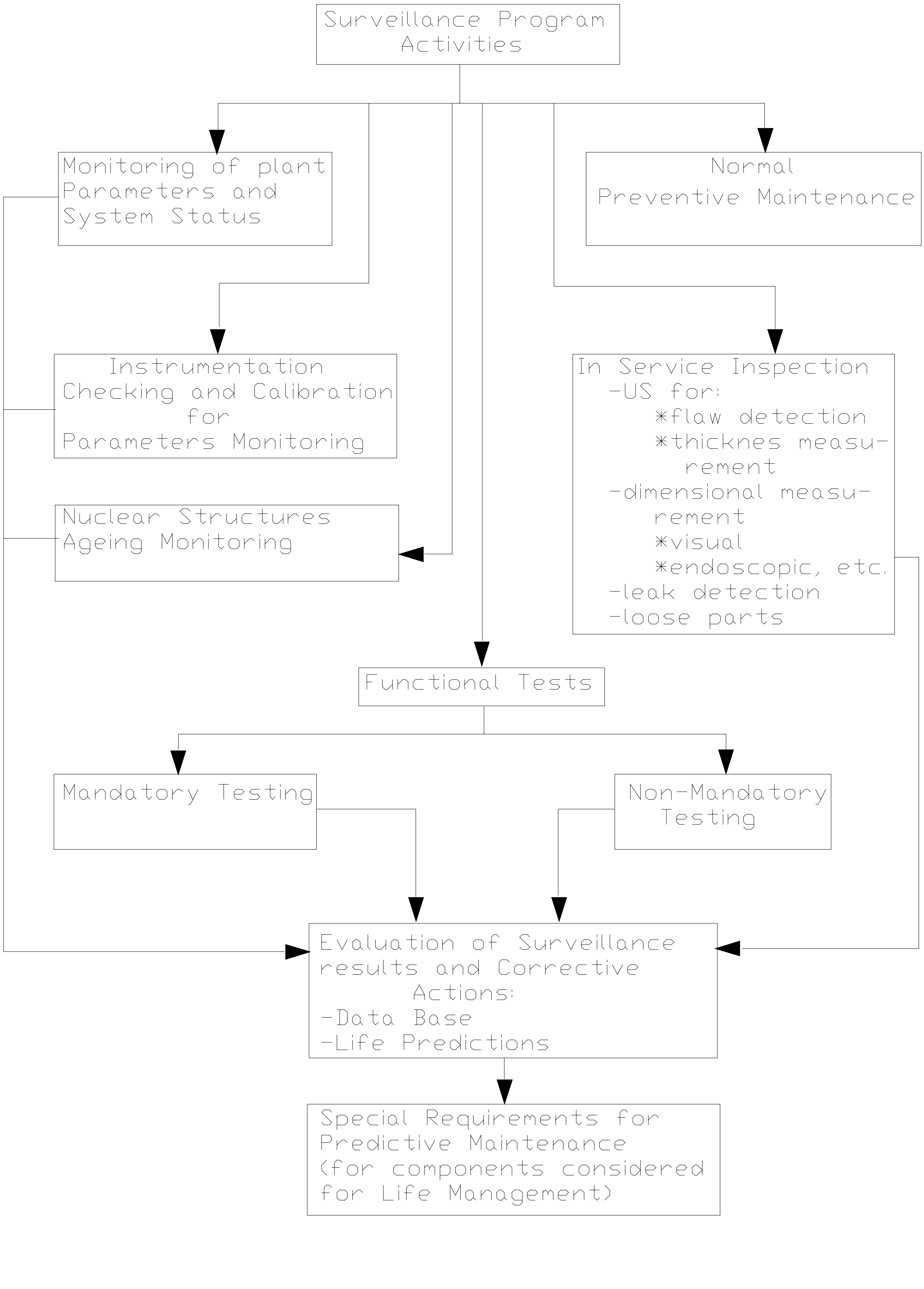
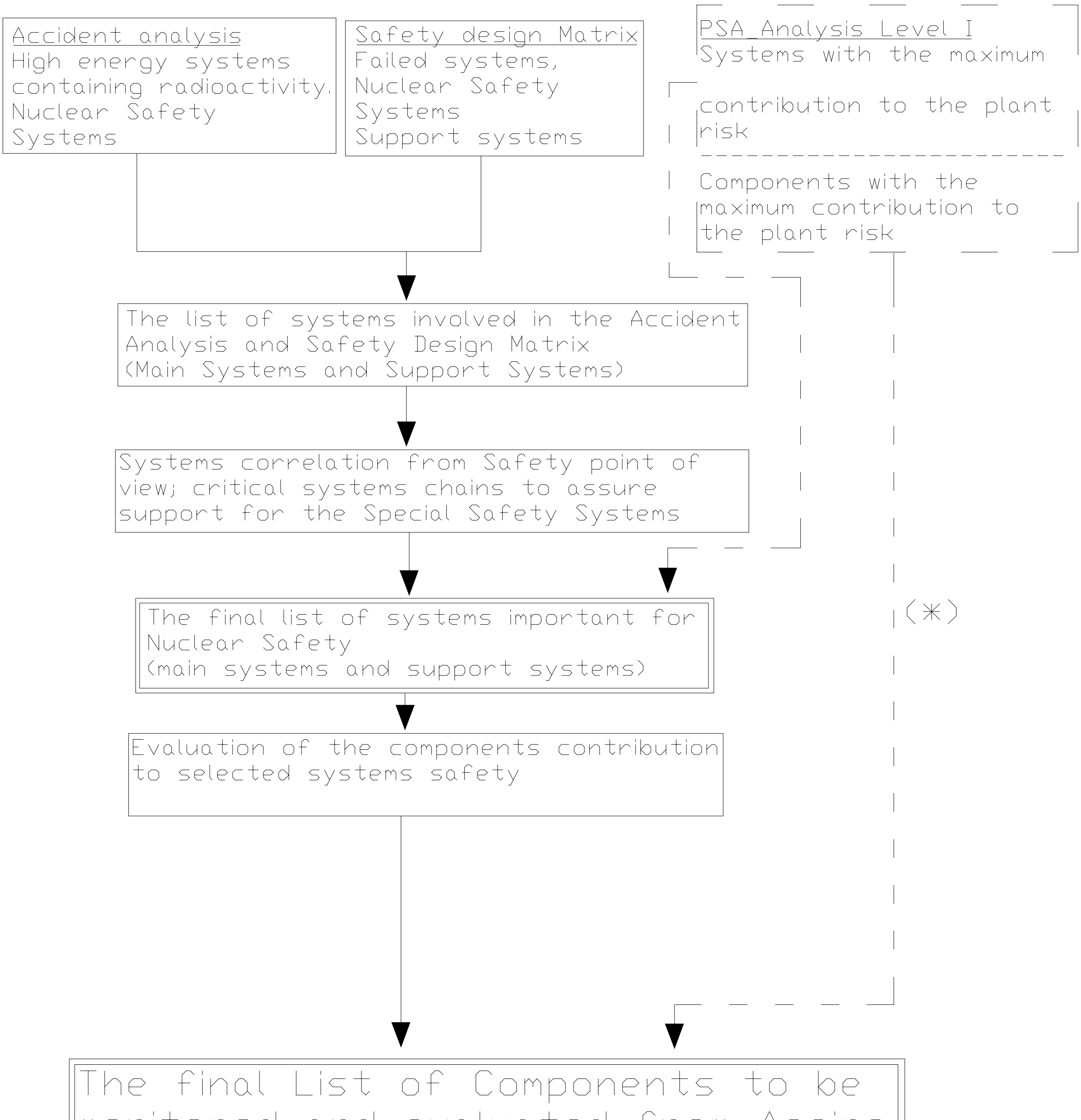


Fig. 3 Critical systems and components selection from Safety point of view.



(*) Could be a direct way, but need to be completed with engineering judgement, based on operating experience

Fig. 4 Critical Systems Correlation for Safety Assurance.
(part 1)

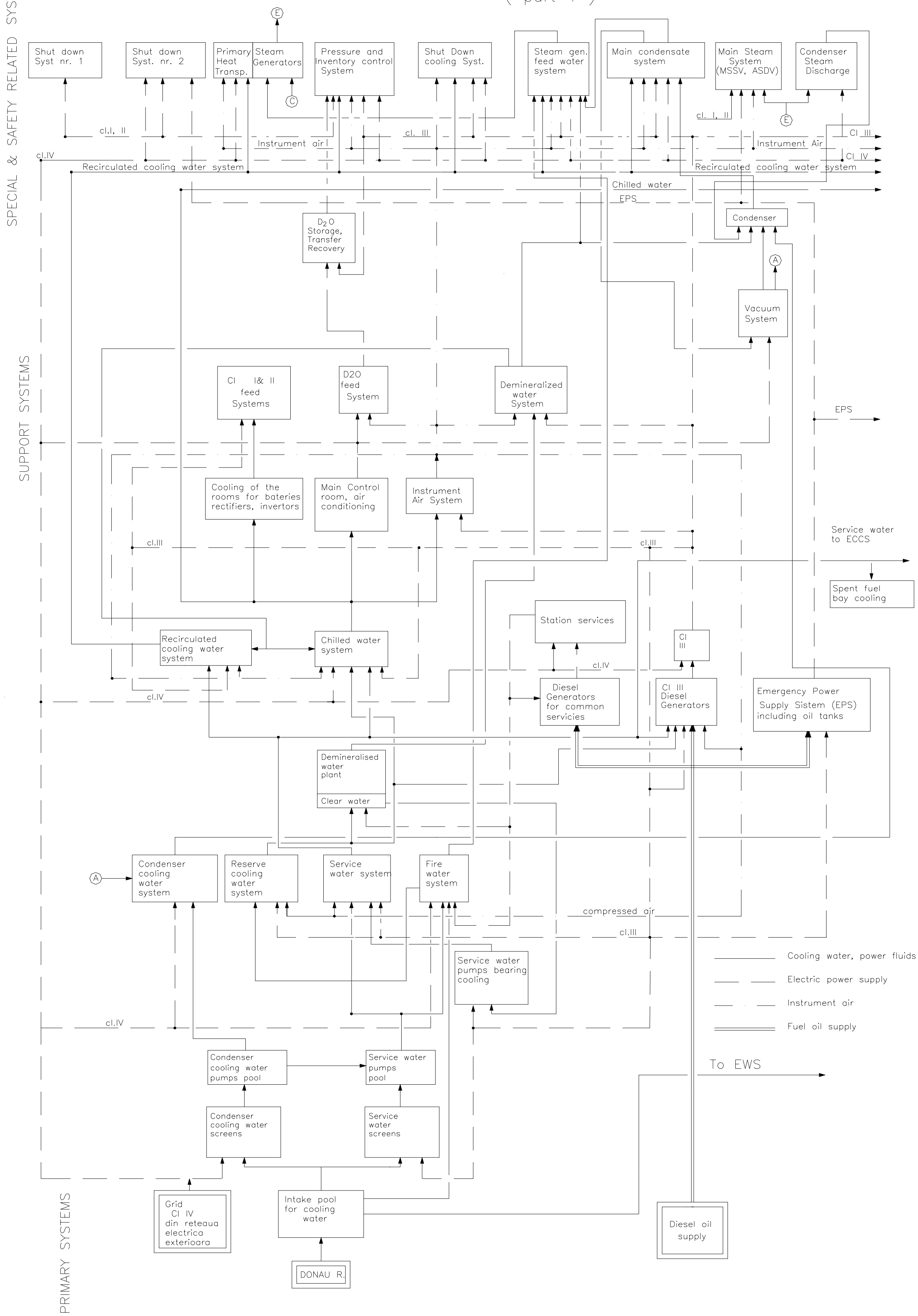


Fig. 4 Critical Systems Correlation for Safety Assurance.

(part 2)

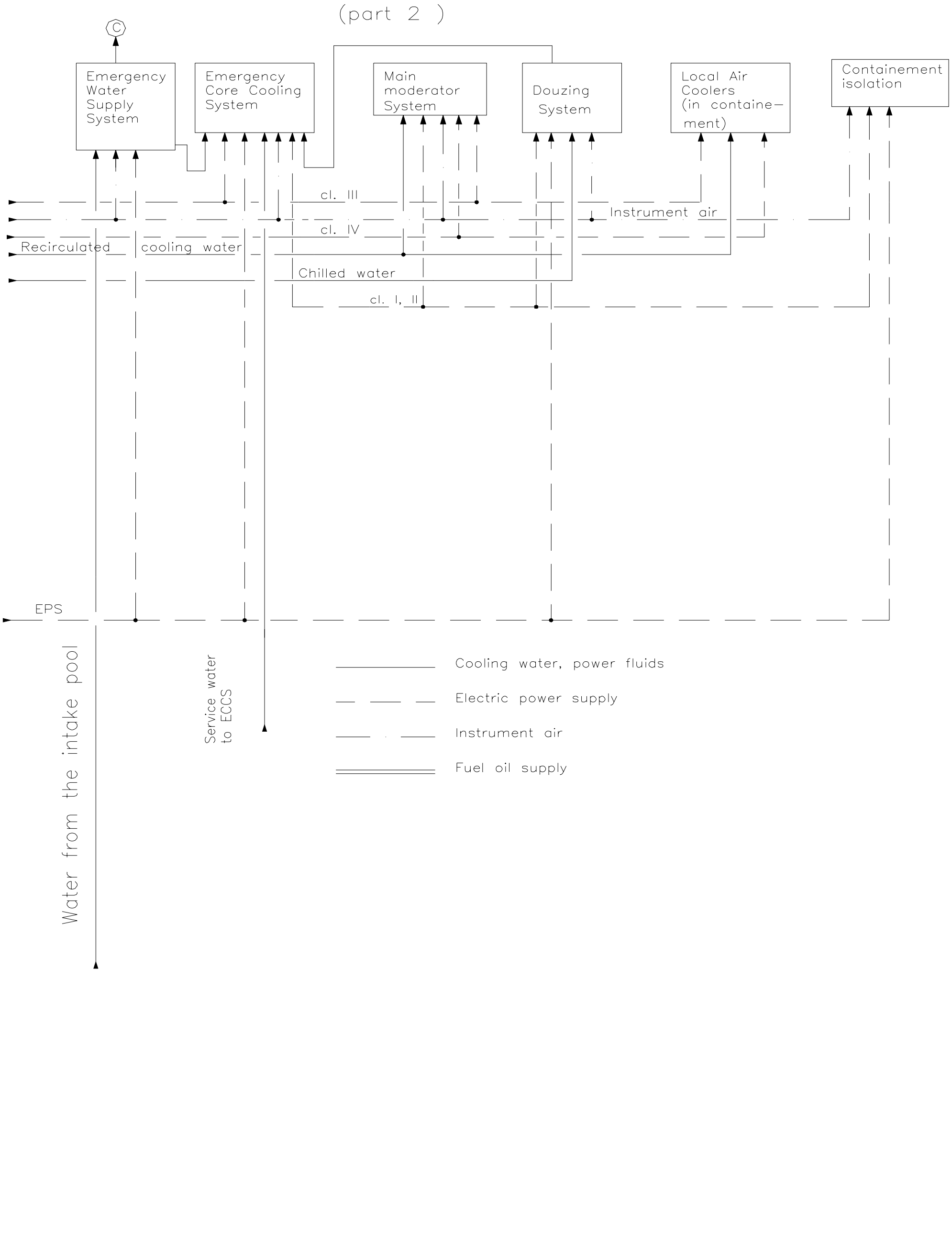
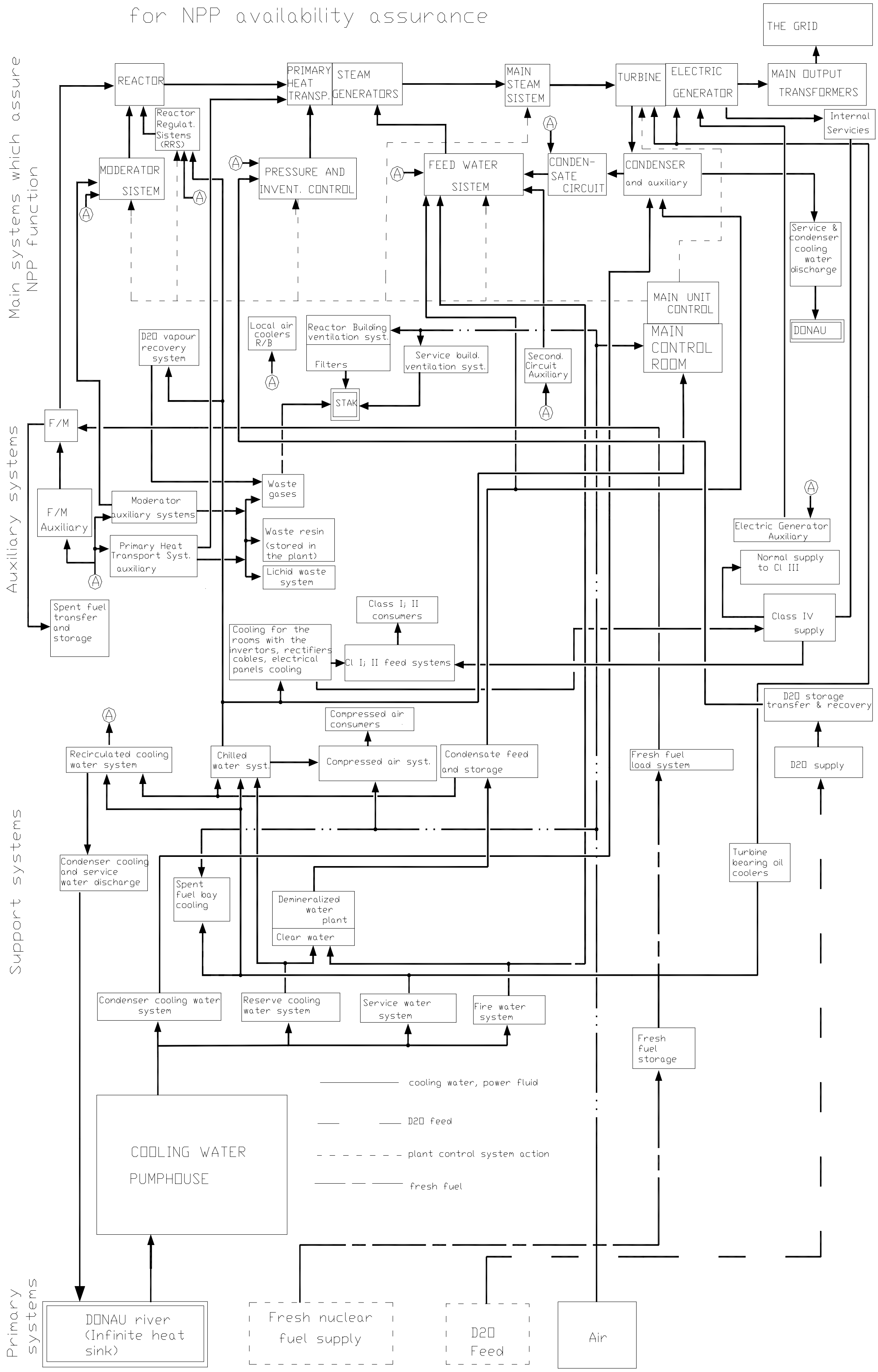


Fig. 5 Critical systems correlation for NPP availability assurance



PLANT LIFE MANAGEMENT AGEING IN NPP DUKOVANY AND SOME EXAMPLES

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Abstract

The ageing of lifetime management at Nuclear Power Plant (NPP) NPP Dukovany is oriented towards an effective usage of facilities while keeping their safety parameters, mainly concerning nuclear safety.

The goal is to achieve maximum usage of the design lifetime under the condition of fulfilling abovementioned requirements. Evaluation procedures have been prepared for selected components. These are subsequently discussed and approved by the State Office of Nuclear Safety (SONS). The databases of original data are built up regarding the evaluated components. The input data are completed as well according the approved procedures. Generally valid approach based on the evaluation of all possible degradation mechanisms is modified regarding the original status of particular component. The procedures are oriented towards a quantification of residual lifetime. They employ knowledge of initiation and kinetics of degradation mechanisms and subsequently also define parameters, which should be quantified.

For irreplaceable components or components for which replacement is difficult or economically inefficient, the programme is focused to the control of operating conditions and inspections of facilities. However, they are again based on the knowledge of degradation kinetics. If at least some parts of the facility could be replaced, better material selection regarding the operating conditions is an option.

Systems included into the ageing management programme will be specified in the paper. Also, the measures will be identified which serve as a check of implementation of all possible stressors and degradation mechanisms into the first screening.

The examples of lifetime evaluation of some systems and components will be given in the paper: reactor pressure vessel, reactor internals and steam generator. Specific approaches will be shown for different original status of components and different operating conditions.

1. INTRODUCTION

NPP Dukovany is situated in the Czech Republic and contains four standardized units with WWER-440 MW type reactors of V-213 version (Water-Water-Energetic-Reactor = PWR type). First of the unit was put into operation in 1985 and the last one in 1988, thus they are in operation between 14 and 17 years. During this operation, several mitigation measures have been applied as well as upgrading of reactor operation procedures. Plant Life Management Programme has been established as an important part of reactor control and operation as the utility expressed its plan to extend original design lifetime of the plant (30 years) and of the reactor pressure vessels (40 years) to approximately total 60 years, if possible.

2. PROCEDURES FOR LIFETIME EVALUATION

Periodical lifetime evaluation (LE) of NPP main components is required by the Czech Atomic Law No. 18/1997, as well as by a SONS Decree No. 214/1997. General requirements for lifetime evaluation of nuclear pressure vessels and internals are given in the State Office for Nuclear Safety (SONS) "Procedures and requirements of lifetime evaluation of WWER reactor pressure vessels and their internals during NPP operation" (1998).

Lifetime evaluation is based on potential damaging mechanisms for individual components as well as on their mode of possible failure – e.g. brittle/non-ductile failure for thick ferritic materials, ductile failure for austenitic materials, leakage or full failure for tubing and piping etc. Thus, individual procedures must be prepared for individual components – recommended procedures and necessary requirements as well as material design properties etc. are given in the ASI (Czech Association of Mechanical Engineers) CODE, Section IV "Lifetime determination of components and piping in WWER type NPPs".

Plant Life Management Programme (PLIM) or Ageing Management Programme (AMP) should be prepared in full accordance with the chosen procedure of lifetime evaluation. This is the only way to ensure necessary effectiveness of activities in mitigating material ageing to assure required components lifetime, which is the main aim of the whole process.

3. PRINCIPLES OF THE LIFETIME EVALUATION PROGRAMME AT DUKONANY NPP

The basic concept of the LE programme at Dukovany NPP (EDU) has been based on the assumptions:

- only operating conditions could be changed in order to control lifetime of the component or system which is irreplaceable or for which replacement is technically or financially difficult
- for replaceable facilities, the ageing can be suppressed by replacements of components of such facility, or even by the replacement of all facility. Moreover, a new component or facility may possess better technical or material design

To assure an effective usage of the property of CEZ (Czech Power Company) - EDU NPP with minimum costs and minimum increases of input costs is the very basic principle of the plant lifetime management programme at EDU. Moreover, it should support also an effort for competitiveness of the company. Together with the LE programme, the Programme of qualification of safety related facilities is run.

Such a principle will be fulfilled if the particular goals are met:

- utilization, operation, diagnostics and maintenance of the facility from the perspective of full usage of the design lifetime, together with the evaluation and keeping qualification of safety related facilities to reach a goal of safe and economical operation until 2025.

- to optimise usage of irreplaceable facilities or facilities with limited replaceability until the planned lifetime while fulfilling all safety and reliability criteria should be kept
- re-qualification of facilities in the safety approved cases, especially while fulfilling SONS requirements in the field of equipment qualification, in accordance with decree No. 214/97.

In 1999, EDU issued the Rules for the lifetime management of tangible assets under rule of CEZ-EDU. The position of Lifetime Programme Manager has been created. His principle duty is defined as activities that lead to a restriction of facility degradation. It includes also a co-ordination of persons responsible for the particular facility.

The most focused systems are as follows:

- Reactor pressure vessel and reactor internals
- Steam generator including heat exchange tubes
- Main circulation piping
- Main circulation pump
- Main gate valve
- Piping of safety class 1
- Pressurizer
- Feed water and steam piping
- Cables

In this paper, LMP of three of these components will be shown and discussed.

Example 1 of Lifetime Management Programme: Reactor Pressure Vessel

LMP of any component cannot be realized without a proper knowledge about component integrity and residual lifetime. Such evaluation should have to be performed periodically with an interval that is short enough for any necessary changes/mitigation in reactor operation if damage trends would be found to fast and residual lifetime prediction shorter than expected. Usually, five years period is sufficient.

Flow chart of reactor pressure vessel (RPV) lifetime evaluation is shown in Fig. 1. This chart is based on the procedures given in the ASI Code for Reactor Components, Section IV on Reactor Components Lifetime Evaluation during Reactor Operation. Actual initial material properties as well as results from stress analysis and non-destructive pre-operational examination summarized in databases are a necessary input for any evaluation. Then, real operational regimes and parameter transients together with changes in material properties (radiation damage), values of neutron field in reactor vessel wall (obtained either by calculations and/or by measurements) together with results of non-destructive in-service inspections are the next input for such periodical lifetime assessment.

Integrity/lifetime assessment consists from four parts – evaluation of potential non-ductile failure, fatigue damage, defects allowance and corrosion-mechanical damage. If all these evaluations results in required (design and/or extended) residual lifetime, then operation can be maintained with existing parameters (Y). In the other case (N), necessary mitigation procedures must be applied and residual lifetime evaluation must be repeated. This cycle should be repeated/corrected till required residual lifetime can be ensured. In such cases, desired residual lifetime is either assured, or only residual lifetime is determined.

Key elements of a WWER reactor pressure vessel ageing management programme are shown in Fig. 2.
In general, five steps of the programme are usually defined as based on the following scheme: TO UNDERSTAND – TO PLAN – TO DO – TO CHECK – TO ACT with a continuous feedback and mutual corrections.

1. *Understanding RPV ageing*
   This step is fully understood on the basis of many analyses of operating conditions and materials behaviour. Leading ageing mechanism is radiation embrittlement for the beltline region while fatigue damage can be important for parts with high stress concentrations like bolting joints, nozzles, etc.

2. *Co-ordination of RPV ageing management programme*
   This programme has been prepared on the basis of Step 1 as well as on regulatory requirements and potential procedures/activities to affect components integrity.

3. *RPV operation/use*
   This step is very important as it is fully connected with RPV operation – operation regimes and their conditions have been modified in spite of the basis of known damaging mechanisms and potential mitigation methods – low leakage core has been introduced practically from the beginning of reactor operation to decrease radiation embrittlement, and some regimes has been forbidden (e.g. connection of the cold loop to the reactor) to decrease fatigue usage factor.

4. *RPV inspection, monitoring and assessments*
   This step represents a continuous activity, which serves as a feedback for the component lifetime evaluation and assurance. Continuous monitoring of operating conditions (pressure, temperature of the coolant and of the vessel, flow) is supplemented by periodic in-service inspections – non-destructive (mostly ultrasonic, eddy current and optical/TV) inspections in four/eight year period and destructive testing of standard and supplementary surveillance programme specimens including ex-vessel fluence measurements.

5. *RPV maintenance*
   This step represents mostly potential mitigation activities that are connected with a local damage (flanges, threads, etc.) or the substantial part (radiation damage in beltline region = use of low-leakage core). Such mitigation can extend components lifetime by a substantial damage.

All these steps are connected together by their feedback properties and functions. For reactor pressure vessels in their young life, the most important part can be seen in a proper inspection and monitoring, and finally in periodic lifetime assessment (based on qualified damaging trends construction) to be able to catch in time any anomalous behaviour of materials. On such basis, a plan of necessary modifications of operating conditions in the case of small imperfections, or a plan for some major mitigation activities (vessel annealing) could be prepared in time to be cost and functionally effective.

Example 2 of Lifetime Evaluation Programme: Reactor Internals

The main objective of the Programme is coming from the decision of State Office of Nuclear Safety No. 197/95. It has asked utilities to include into the reactor internals strength calculations also evaluation of both strength and lifetime from the viewpoint of influence of irradiation on the changes of material properties. That means also to determine principal damage mechanisms and to prove an ability to control them for all time of planned operation.

Most of reactor internals (core barrel, etc.) are manufactured from a 08Kh18N10T type titanium-stabilised austenitic stainless steel similar to AISI 321. Thus, main difference in comparison with steels used in PWR type reactor internals is in steel titanium stabilization. Calculated neutron fluence for end of life (40 calendar years) is about 25 dpa. Use of low-leakage core which was applied in NPP Dukovany, can decrease neutron fluence approximately 1.4 times in its maximum.

The principle tasks have been identified:

- To secure an ageing management of materials and structures of RPV internals influenced by radiation embrittlement and irradiation assisted stress corrosion cracking (IASCC). To define the limit values of critical parameters suitable to be monitored on the four year period basis.
- To determine sensitivity to radiation embrittlement and IASCC degradations for 40 years of planned operation. To determine current status of mechanical and corrosion mechanical properties of reactor internals material.
- To propose measures to monitor changes of material properties of reactor internals. To define procedure how to build these results into the current models of damage kinetics.
- To calculate stress and deformation status of structures, to compare them with the material properties after 10 - 40 years of operation.
- To determine threshold level of stress and fluence and to estimate kinetics of cracks growth. Based on these values, prepare a model of degradation and subsequent calculation of residual lifetime.

As a result of the work, a prediction of degradation of reactor internals will be prepared together with the methodology of in situ measurements and stress and deformation calculations.

The principle structure of the Programme is shown in Figure 3 where main tasks are present.

As no fully applicable model exists for lifetime evaluation of WWER reactor internals, all efforts must be directed to gain necessary experimental and calculation data to be able to create such a model and also to validate it.

Thus, this programme consists from four parts, in principle. First of them are inputs from calculations – Detailed neutron flux calculation was planned in order to determine real fluences during the planned period of operation. The next input is stress and deformation status of the structure during normal operation as well as during transients including potential earthquake cases. Then, further main task contains material irradiation and testing in order to prepare a quantified model of materials degradation. Determination of changes in material properties have been realised in an experimental irradiation programme – material specimens were irradiated to several fluences up to the design end-of-life fluence to able to evaluate fluence dependence of these changes. Material selection for the tests comes from the detailed material technology knowledge. Material testing consists of quasi-static tensile tests to determine tensile properties – yield strength (YS), ultimate tensile strength (TS), total elongation (TE) and reduction of area (RA). Fracture feature/type has been also determined. Next type of testing was concentrated on testing sensitivity to stress corrosion cracking when slow-strain rate testing (SSRT) under operating conditions (temperature, pressure, primary coolant) was applied to irradiated specimens. Microstructural investigations have been also performed on highly irradiated specimens to evaluate nature of radiation damage. Based on all these results, a phenomenological model of radiation damage has been proposed – it is suitable to predict status of reactor core internal materials in future of reactor operation. Then, based on this model, allowable deformations that still prevent the failure can be assessed and together with results from stress analysis, a status of the component/residual lifetime can be evaluated. In situ measurements of material properties changes will serve for a periodical checking of the proposed model – instrumented hardness measurement of the component will be applied which allows to evaluated tensile properties of tested materials. Eventually, a procedure for lifetime determination will be proposed.

Example 3 of Lifetime Evaluation Programme: Steam Generator Tubes

Within more than twenty years of operation of the WWER 440 steam generators, three most frequent types of damage have occurred: stress corrosion cracking of collectors, stress corrosion cracking of heat exchange tubes, erosion-corrosion damage of the feeding pipes. Whereas the latter problem, in principle, has been removed, remedial measures have been introduced to reduce the primary collector damage. The trend of damage in some of the steam generators can play a significant role in ensuring their planned life even damage rate is rather low.

WWER 440 steam generators are horizontal bodies with two cylindrical collectors for primary water inlet and outlet. The heat exchange tubing, tube support plates and primary collectors are made of a 08Kh18N10T type titanium-stabilised stainless steel similar to AISI 321. The most significant difference lies in the geometry, given by the vertical design of PWR steam generators in comparison with the horizontal arrangement of WWER steam generators. They lead to differences in the amount of solid particles deposited from the water volume of a steam generator, absolute volumes of water, proportions of the water and steam volumes, extent of the water free surface, differences in formation of locally concentrated regions in the water phase, possibly also processes of deposition of compounds in crevices and others.

Another difference is the surface temperature of heat exchange tubes, whose level in PWR steam generators is higher by up to 40 K. While the PWR steam generator primary side suffers from stress corrosion cracking, on the WWER steam generator primary side such damage has not been observed. On the WWER 440 steam generator secondary side the damage manifests itself primarily by stress corrosion cracking resulting in tube plugging. In different nuclear power plants the extent of steam generator tube plugging varies, nevertheless in Czech and Slovak plants it attains values of the order of only one tenth of percent of the total number of tubes.

Up to now any simple relationship between operating modes and the defect initiation and propagation has not been established. Stress corrosion cracking results in formation of non-through as well as through-wall cracks oriented primarily in the axial direction, exceptionally also in circumferential direction. The defects are predominantly intergranular. The cracks are initiated either on smooth surface or at the bottom of pits.

At present, tube damage is detected primarily by the eddy current method that, however, is burdened with a considerable measuring error. The resulting large scatter can lead to two sorts of misleading: a steam generator tube, which is not significantly damaged, may be plugged or a seriously damaged tube can be omitted. The risk of the latter mistake is reduced by conservatism of limits and by application of results of studies concerned with assessment of critical stresses in a tube with a crack and with measurement of leaks through a damaged tube. It is evident that a considerable over-estimation of damage may occur with unnecessary tube plugging.

In order to evaluate the trend of damage, it is necessary to know the kinetics of the defect initiation and growth. On the basis of this knowledge it is then possible to evaluate the relative increment of damage, as indicated by measurement, especially if larger data files are available, enabling the use of statistical methods. By correlation of measured and predicted values it is possible to assess the extent of conservatism in determination of tubes for plugging, and, if necessary, also to check the efficiency of remedial measures, including changes in water chemistry.

Deviations of the feedwater chemistry from the long-term standard levels bring about changes in the chemistry of crevice environment and in conditions of crack initiation and growth, presumably also in exhausting the steam generator service life. The changes may apply to more sophisticated water chemistry used to slow down the ageing of components, as well as to unforeseen deterioration of water chemistry, e.g. during the operation of a condensate polishing plant. Knowledge of these parameters enables to predict the extent of service life exhaustion within a time period, e.g. a campaign, providing a model of damage is available.

For assessment of the steam generator ageing an evaluation system has been prepared, enabling to perform the above-mentioned analyses for real conditions of WWER steam generators.

Principal features of the system are:

1. Relations for critical crack size and permissible and measurable leak through the defect.
2. Introduction of a hideout return evaluation procedure. At NPP Dukovany analyses have been carried out to identify chemical conditions in crevices.
3. Knowledge of the local crevice environment, based on a HOR analysis by the MULTEQ code, a temperature analysis and a model of dynamics of concentration. In case of occurrence of short-term deviations in water chemistry, these changes can be considered.
4. A model of damage initiation and kinetics of SCC crack growth. It will be experimentally determined on a structurally identical material under conditions simulating the local environment including concentration factor and mechanical stressing of tubes.
5. Time to defect initiation is to be measured by the method of constant loading of C-rings.
6. By slow loading of CT-specimens (RDT test) a threshold value of the stress intensity factor for crack growth should be determined, together with the dependence of the growth rate on the J-integral.

The evaluation procedure is shown schematically in Fig. 4. On-line and off-line gathering of data on the feedwater and blowdown chemistry, together with the HOR evaluation, should enable to model the local crevice environment. Based on the developed damage model, evaluations should be carried out after each service cycle and they are planned also in case occurrence of over-limit values in the water chemistry. From the NDE measurements and the predictions the extent of lifetime exhaustion in the given operating cycle is determined.

4. CONCLUSIONS

The paper describes principal activities in the field of Plant Life Management of main components in NPP Dukovany that should ensure design plant lifetime and further lifetime extension with minimum economic expenses with required level of reactor safety as required by state regulatory bodies – SONS.

Description of main principles for LMP is given as well as requirements that must be fulfilled. Explanation of the LMP is given in two examples of the most important components – reactor pressure vessel and reactor internals. In both these LMP, calculation, monitoring as well as material specimen testing and in-service inspection is required to be able to periodically evaluate components residual lifetime and, if necessary, to proposed possible mitigation procedures and assess their potential effects.

FIG. 1: Flow chart of reactor pressure vessels lifetime evaluation

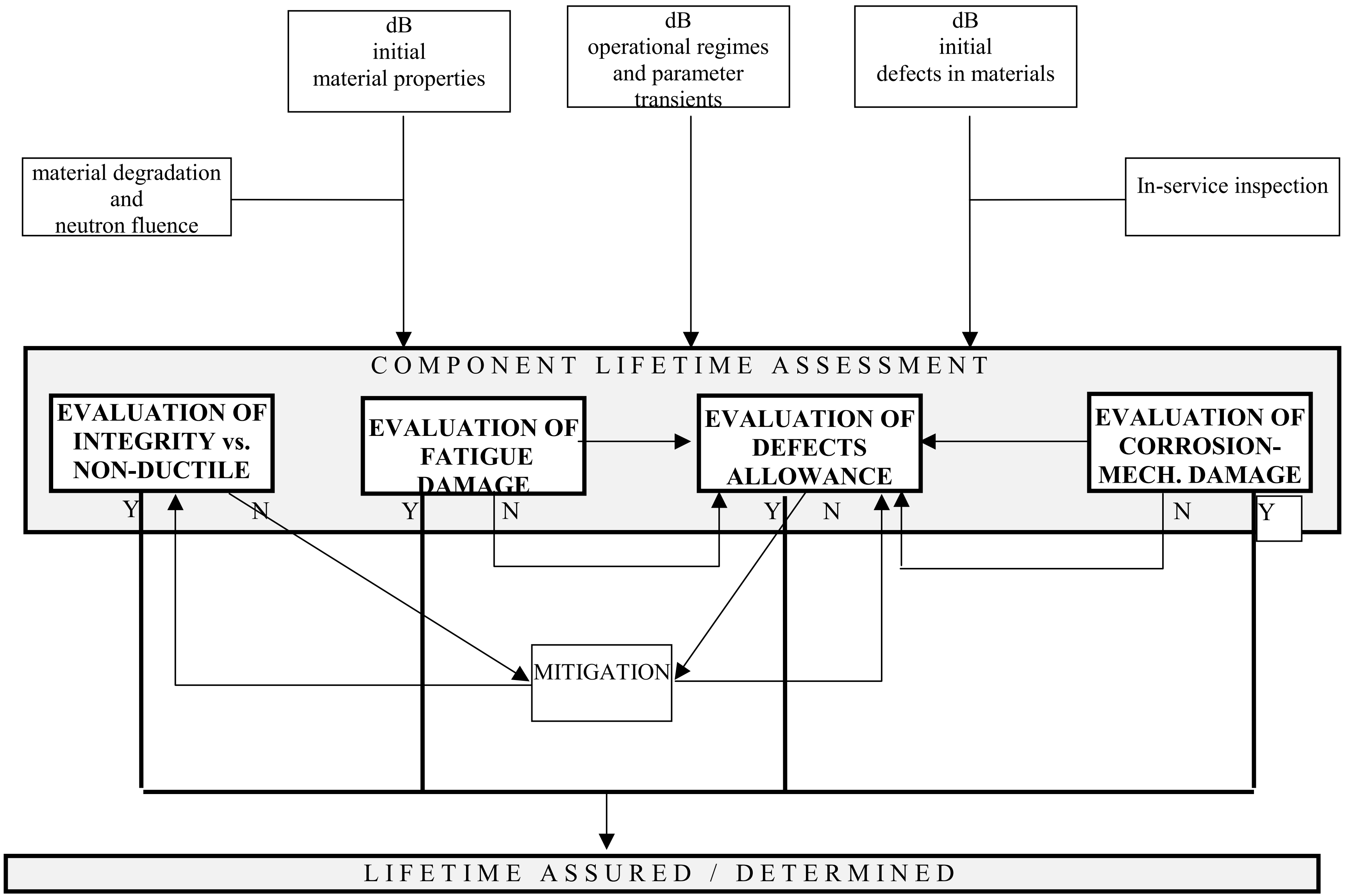


Fig. 2: Scheme relationsships and principle programmes of NPP related to Lifetime management Programme

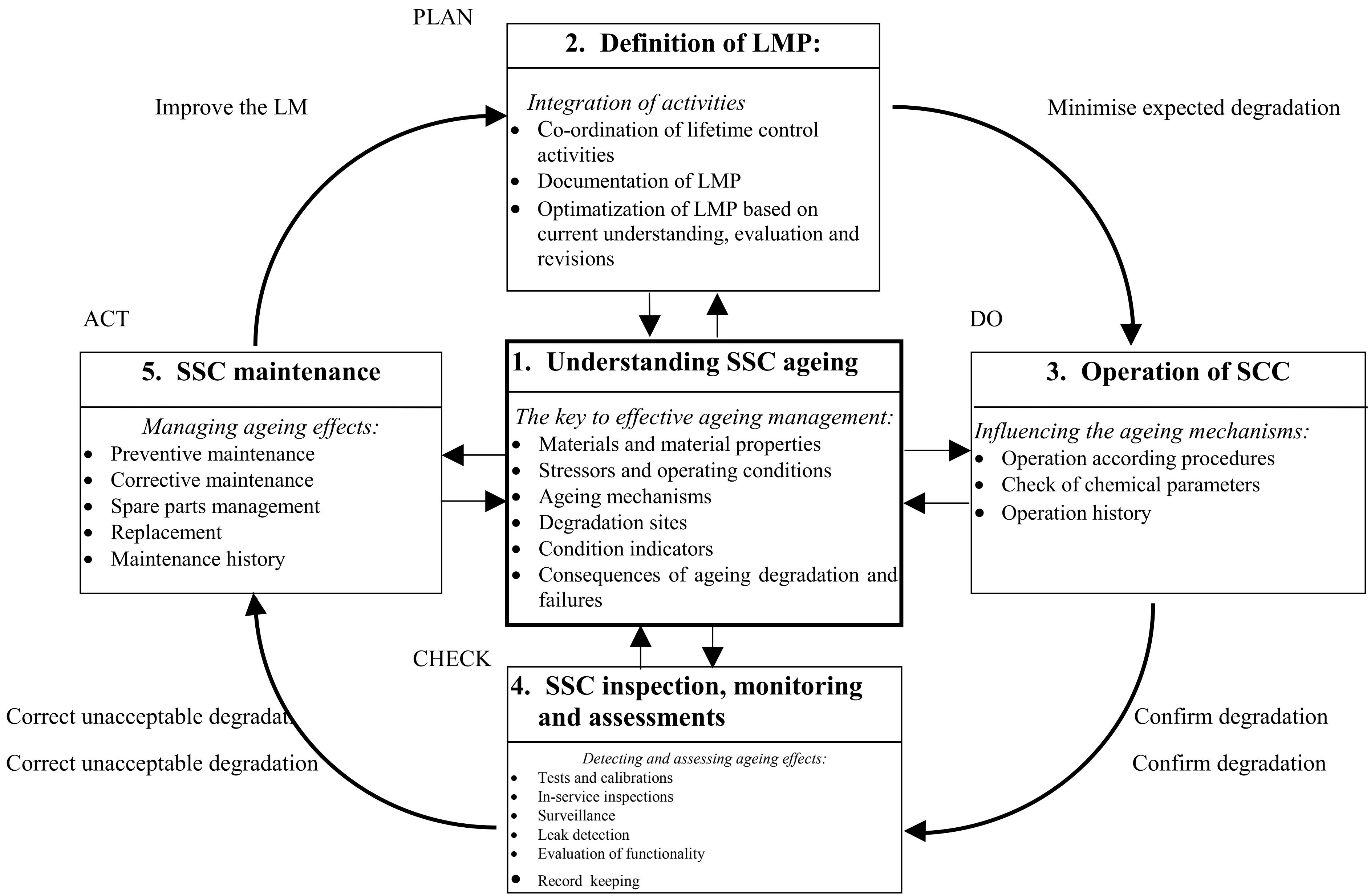


FIG 3: Scheme of WWER reactor internals lifetime evaluation programme

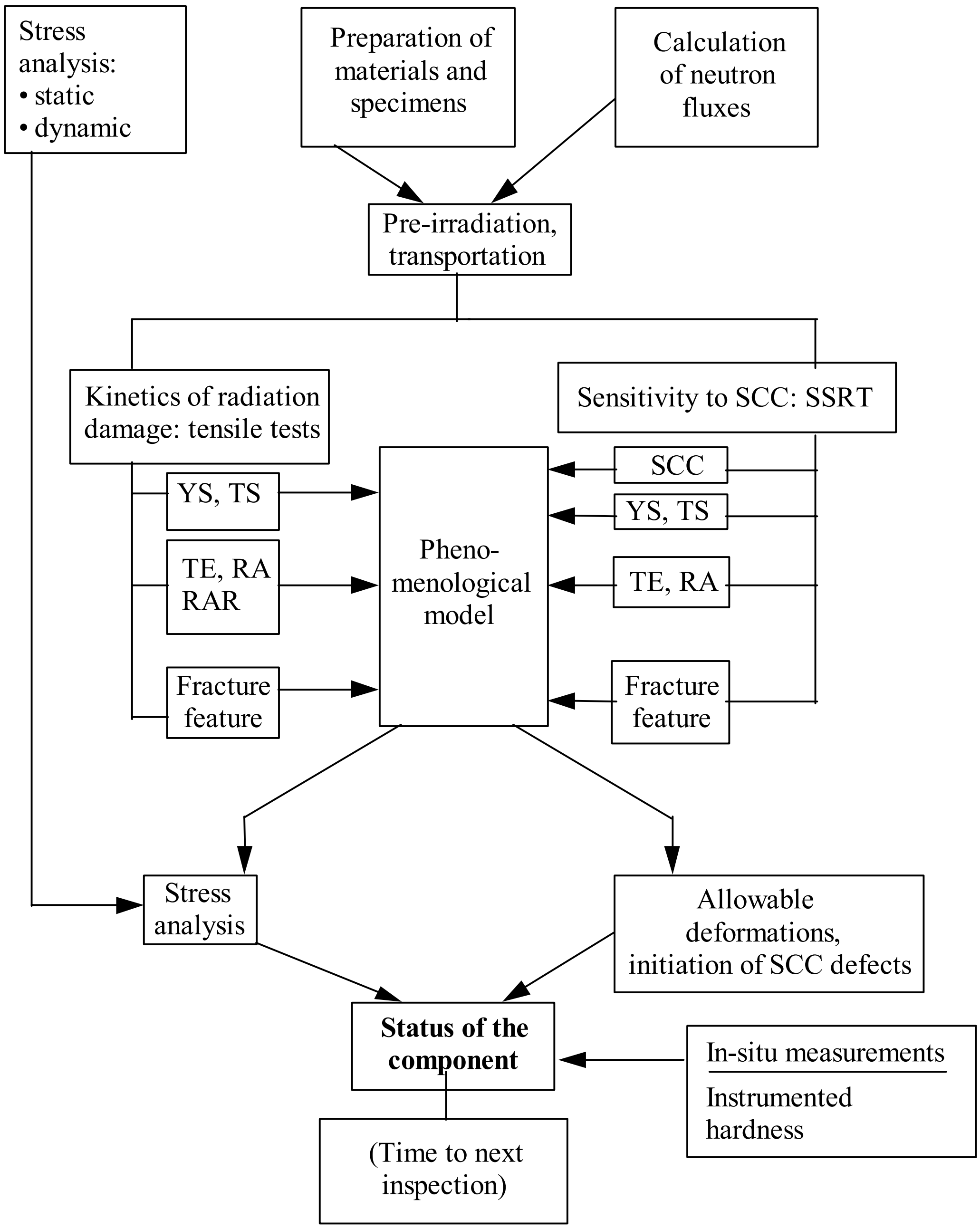
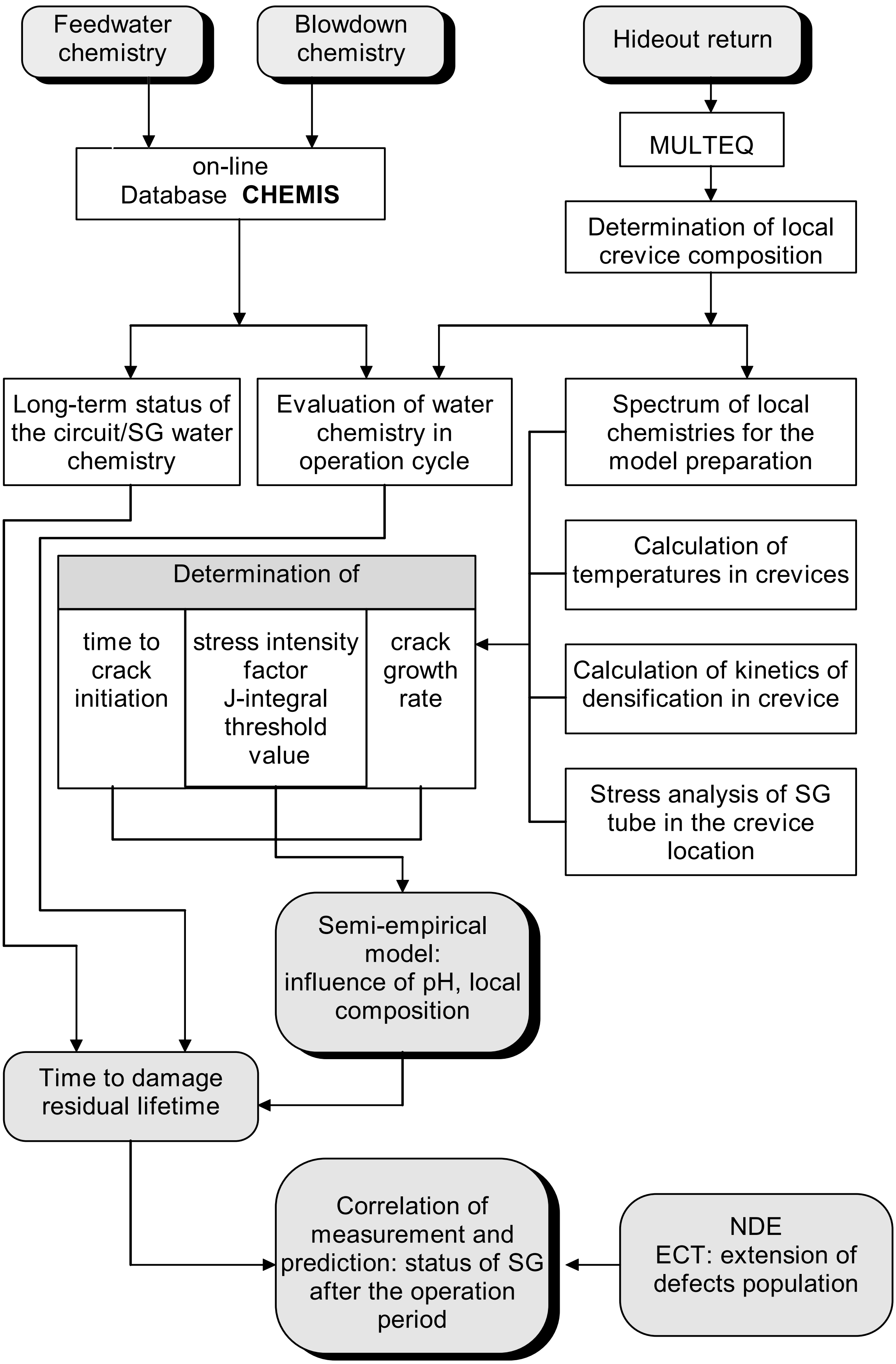


FIG. 4: Scheme of the lifetime evaluation programme for the steam generator tubes



Session 2

PLANT LIFE MANAGEMENT (PLIM)

Sub-Session 2.4

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REQUIREMENTS AND TRAINING OF MANPOWER FOR CO-GENERATION NUCLEAR POWER PLANT

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Abstract:

The plans of Egypt to populate the arid zones, implies the use of co-generation nuclear power plants. A nuclear power plant of medium size may be constructed on the northwest coast near Alexandria. In this work, the requirements of qualified personnel for the construction, operation and maintenance are suggested depending on the gained experience of advanced countries. The German experience in the field is taken as a reference. Also, a training program is suggested to qualify the required personnel during the time of construction. This training includes both nuclear and desalination technologies and extends for about 5 years.

1. INTRODUCTION

In Egypt where the desert lies beside the sea in most cases, the co-generation power plants seem to be the solution of the population problems in these arid zones. Through the production of water and electricity, new towns may be constructed depending on agricultural, industrial and tourism activities [1]

The implementation of a medium size nuclear power plant in the northwest coast of Egypt near Alexandria is considered nowadays. The manpower quality is an important factor to achieve the project. The features of such plant can be seen in many studies [2]

2. MAN POWER REQUIREMENTS

The implementation of a nuclear power program in a developing country is a complicated dynamic social, economic and technical process. It starts with the planning stage where all available energy resources and load forecasts are analyzed to determine the feasibility, size and schedule of the program. Methods of implementing nuclear power programs in a developing country may range widely between two extremes; one would be just importing turnkey nuclear power stations and operating them to satisfy the electricity demand. The other extreme is to have comprehensive nuclear power and fuel cycle programs to achieve self-sufficiency in nuclear technology, such as the line followed by India and Brazil. The manpower requirements will also range widely between these two extremes. The need for trained manpower also depends on the national contribution made by a country to the construction of nuclear power plants, and in the management of the fuel cycle as well as of the share the national industry is able to contribute in such programs. Qualified personnel are needed specifically for the following main areas associated with the implementation of nuclear power plants:

(a) Pre-project activities, and program planning.
(b) Nuclear regulations, licensing and safeguards.

(c) Nuclear Power project management.
(d) Plant construction and commissioning.
(e) Operation and maintenance.

At least in the construction of the first nuclear facility in a country, prospective personnel are not likely to acquire practical experience in the country itself. If, in addition, a national nuclear industry and fuel cycle industry is to be set up, the personnel requirements increase by a corresponding margin together with the need for additional qualifications. The level of growth cannot be stated precisely because it is a function of the conditions prevailing in the individual countries, such as:

- Scope of the nuclear program.
- The country's own personnel reserves.
- The status of industrial development.
- The educational structure.

Various estimates of manpower requirement for different scopes of nuclear power programs have been carried by several studies. In developing countries it is expected that numbers of those engineers and technicians needed for the operation stage will be much higher than the lAEA guidelines for these reasons:

(a) Lack of industrial base and consequently lack of semi-trained personnel in related subjects, such as power industry and petrochemical industry.
(b) Need for extra numbers of maintenance engineers and technicians with special skills in nuclear systems who can be called upon from local manufacturers of nuclear power systems,
(c) Need for extra numbers of trained personnel for the further expansion and development of the nuclear power program

As the training level of domestic personnel improves, the contribution from abroad can be reduced correspondingly. It should be taken into consideration, however, the need for a scale up of the manpower requirements in those countries where the personnel efficiency is somewhat lower due to insufficient practical training. The increase in the personnel requirement with rising number of nuclear power plants in a country is much more difficult to assess. It can be assumed that the requirement for operating staff will rise approximately proportional to the number of stations, whereas less increase is sufficient for the areas of planning, consulting, licensing, commissioning and maintenance due to the fact that work in these areas could be done in series with growing number of stations.

For a dual-purpose plant, the estimated number of manpower may be the same of a single purpose plant. The difference is mainly will be in the specializations of personnel.

The suggested method of desalination is the Multi- Stage flash evaporation (MSF) [2].For the pre-project activities, about 26 well-qualified professionals of different scientific areas have to provide the bases of the project.

The project management is the most important organization unit during the implementation of the power plant up to 100 persons among them 80 professional, 15 technicians and 5 accountants are needed. Their Task covers all the stages of the project.

The manufacturing of equipment and components spreads over a period of 7 years. The plant construction takes about 6 years in parallel to the manufacturing. The desalination plant construction takes place with the nuclear steam supply system (NSSS). In this stage, 2500 persons are needed composed of 300 professional engineers, 300 technicians and 1900 craftsman. In this stage, a considerable part of work may be done in the supplier country [3]. The manpower loading of this stage is shown on Fig.1

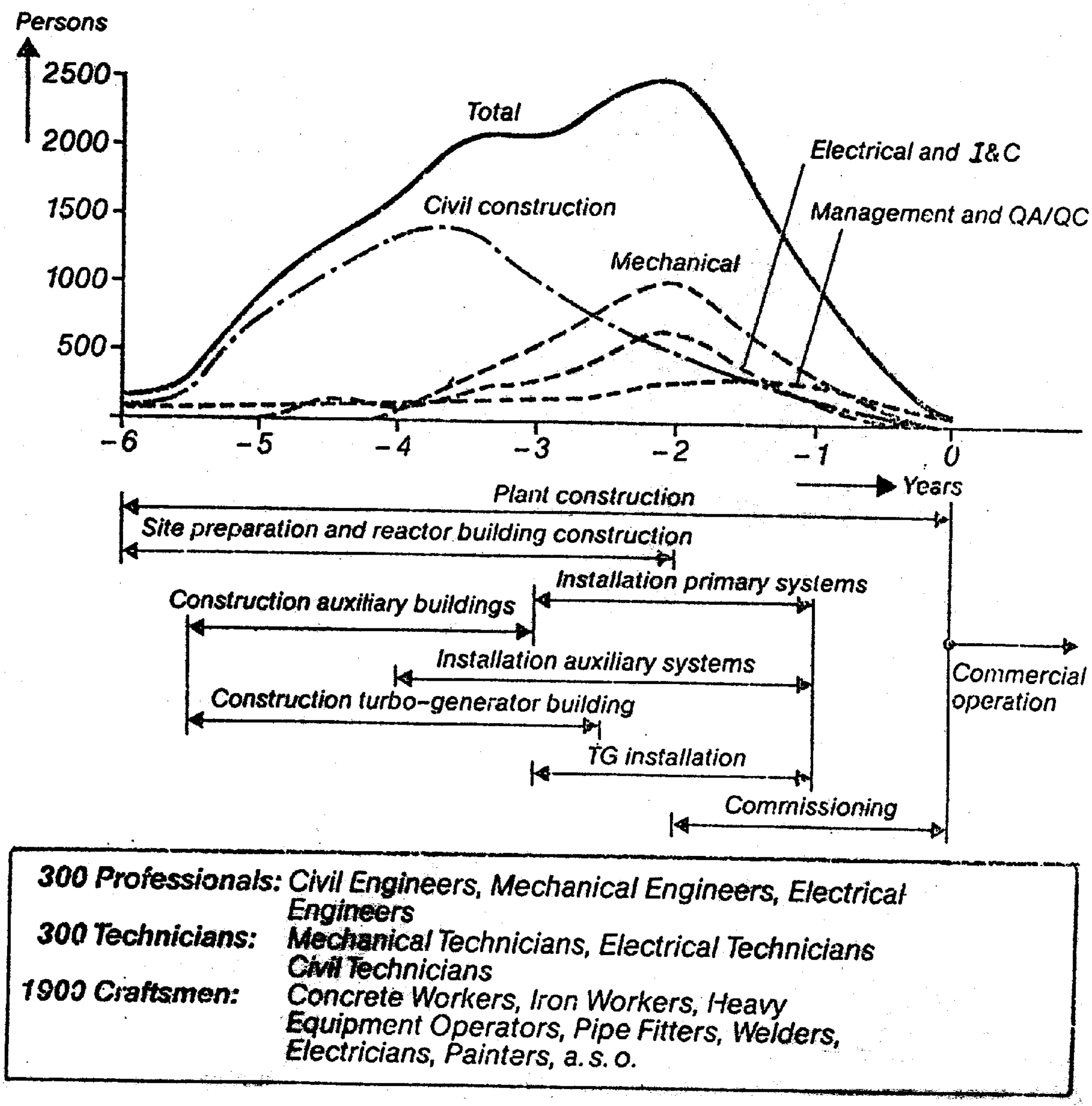
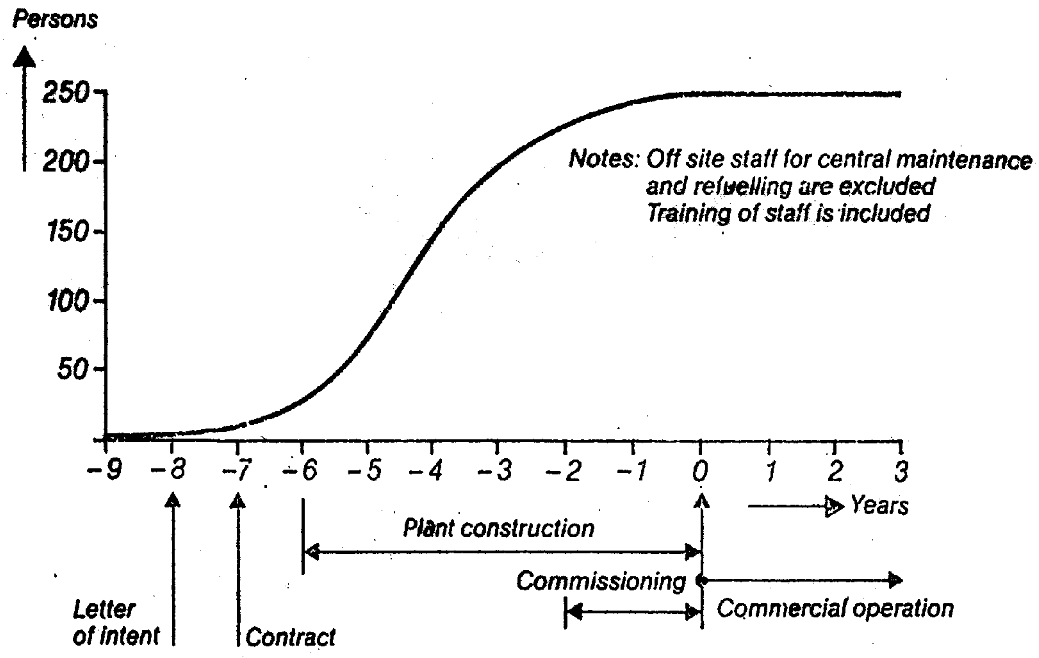


FIG.1. Manpower Loading for Plant Construction.

The operation and maintenance staff is trained in parallel to the manufacturing, erection and commissioning of the plant. The team of operation consists of 50 professionals, 100 technicians and 100 craftsmen. The manpower loading of operation and maintenance is shown on Fig.2.



50 Professionals: Mechanical Engineers, Electrical Engineers, Physicists, Chemists, Chemical Engineers
100 Technicians: Mechanical Technicians, Electrical Technicians, Instrumentation and Control, Technicians
100 Craftsmen: Mechanical Crafts, Electrical Crafts, Electronical Crafts. Civil Crafts

FIG.2. Manpower Loading for Operation and Maintenance.

3. MAN POWER TRAINING

The introduction of new technologies in general, and nuclear power technology in particular, necessitates adequate efforts for the preparation of manpower requirements. Nuclear energy can be introduced and implemented in a meaningful way and without undue risk to the public only if a sufficient technical and legislative infrastructure and technological base exists in the country. Unlike other technologies the time lag between the decision to adopt a nuclear power program and the criticality of the first power reactor is quite long. This offers the possibility to start education and training in advance and to correlate this to the national economic planning in the best possible way. However, the uncertainty of the nuclear plans in many countries demands very close coordination between the training program of a country and its general development strategy. Frequent system analysis and studies should keep the gap between supply and demand of manpower as small as possible.

The economic and technical consequences of either shortfall or overproduction of well-trained personnel may be severe and result in hampering the success of the whole program. Despite the difficulty of a reliable forecast, it is possible to establish some general guidelines for the training of necessary manpower if the boundary conditions are known. These boundary conditions depend largely on the general policies of the implementation of the nuclear power program, the capacity of domestic training institutions and the support of bilateral on multilateral cooperation programs.

From available experience in both developed and industrially advanced countries, it was found that the numbers and categories of manpower requirements depend on the exiting circumstances of the individual country such as:

(a) Available experience in the country in other related industries such us power industry, steel industry, etc.
(b) Level of industrial development in the country.
(c) Growth of the unclear power program, which could allow some specialists to work successively in more than one plant.
(d) Possibility to contract for certain jobs with special characteristics.
(e) Bilateral and multilateral cooperative agreements with other countries

These factors determine the extent and scope of training programs required for the preparation of manpower requirements. To take advantage of the experience of other countries and to adopt some of their programs it is important to compare the education and technical training systems. The following criteria could be taken as basis for the caparison:

i. Number of years necessary to reach certain level of education.
ii. Subjects and syllabus required reaching certain education grade.
iii. Age at the start and end of each education stage.

It is now clear that training is one of the very important and essential factors needed for the development of suitable manpower required for a successful nuclear power program. For developing countries it is important to establish the education systems necessary to raise the level of personnel to the level employed by an advanced country for each individual job. This dictates some steps to be taken. These are:

(a) Division of technical personnel to different levels and categories according to basic qualifications.
(b) Preparation of training programs suitable for each category according to future duties.
(c) Selection of technical subjects for the training programs.
(d) Determination of the time schedule for the training of each category

Figure (3) shows the time schedule needed for the training of each category and the essential training required.

In order to have a successful training program local participation should play important role. The following factors determine the extent of local participation in the training program:

I. The local effort could be introduced such as technician training centers vocational schools and practical training in thermal power plants and industry.
II. Role of local universities and technical schools.
III. Availability of qualified training staff.

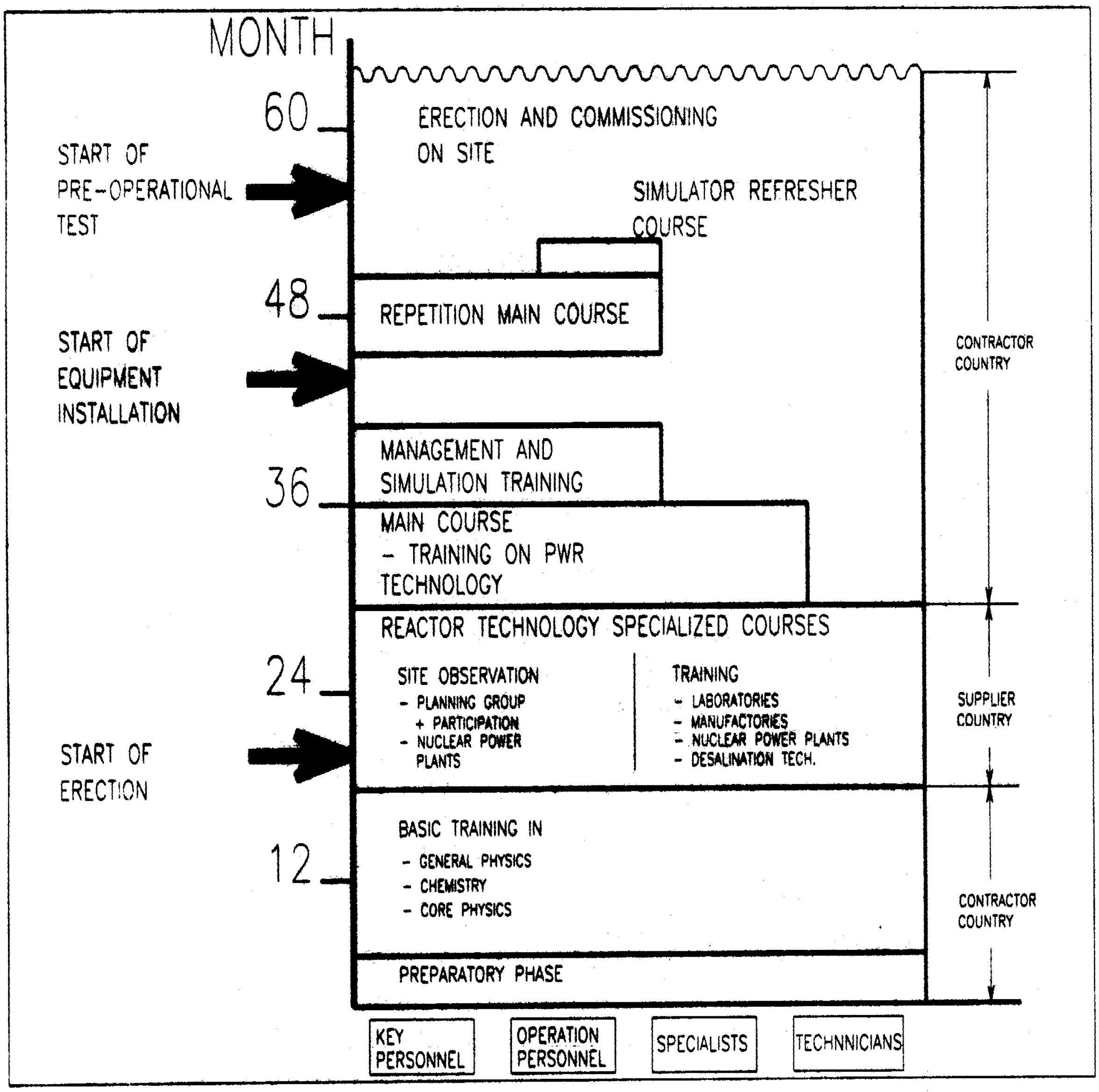


FIG.3. Training Program for Plant Operation Personnel.

4. CONCLUSIONS

4.1- The co-generation nuclear power plants are the best solution for the problem of population of arid zones in Egypt.

4.2- The preparation of manpower in Egypt is necessary to begin a nuclear program. The numbers and qualifications of the staff required to construct and operate the first co-generation plant are calculated.

4.3- The staff-training program in the fields of nuclear energy and water desalination is suggested. The cooperation with more advanced countries in this training is recommended.

REFEREENCES

[1] DARWISH, M. A., " Co-generation power-Desalination Plants", Desalination, vol.69, pp. 27-46(1988)

[2] FOUAD, H.Y., " A Preliminary Study of Dual Purpose Nuclear Plant Features for Egypt", Nucl. Eng. And Design, vol.23, pp. 402-409 (1975)

[3] HERZOG, G. "Nuclear Power Training Course" lectures by staff of KWU Company (FRG) held in Cairo (1984)

REVIEW ON AGEING MANGEMENT OF NPPs - EXPRERIENCE FEEDBACK FROM RESEARCH REACTORS

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Abstract:

Extensive ageing studies were carried out on the 40 MWt research reactor Cirus after 30 years of service. The reactor operated very efficiently till early nineties after which the ageing degradation of Systems, Structures & Components (SSCs) started affecting the reactor operation. Detailed ageing studies were taken up and the SSCs were categorized into in-core components, important safety systems, important out of core components/systems and important structures & auxiliary system. Based on ageing studies and in-service inspections carried out, a detailed refurbishment programme was drawn up and a long outage of Cirus was taken for refurbishing the reactor. Ageing management programme is presently underway at the Tarapur Atomic Power Station (TAPS). These 2 X 160 MWe Boiling Water Reactors at TAPS were commissioned in 1969 and have been in service for over three decades with a plant capacity factor consistently remaining around 80 %. Based on the ageing studies Ageing Management Programme (AMP) for the TAPS has been formulated in line with IAEA Technical Series report no. 338, IAEA TECDOC-540 and other relevant documents. SSCs that are important to safety have been identified with a view to detect their degradation due to ageing and to take advance action through extensive maintenance for their repair or replacement. For each such component the mode of degradation has been identified, ageing assessment has been done and corrective action plan has been defined. Information exchange and experience feedback on various operation, maintenance, inspection and ageing management activities between Cirus and TAPS have helped in formulating and carrying out the ageing management programme for both the reactors.

Ageing of Systems, Structures and Components (SSCs) is a natural process and sets in along with the construction and commissioning of plants in spite of best design provisions and maintenance practices. Plant operators & maintainers need to plan and take measures against ageing degradation of SSCs to maintain the high standards of safety. As safety is a continuously evolving phenomenon, incorporating safety upgrades from time to time and carrying out ageing management towards improved safety for research and power reactors is very important. Cirus research reactor, which was commissioned in 1960, and Tarapur Atomic power station, which was commissioned in 1969, are two such examples of older generation nuclear plants in India which are presently undergoing extensive refurbishment towards implementation of ageing management programme.

The 40 MWt Cirus Research Reactor located at the Bhabha Atomic Research Centre, Mumbai, is a vertical, closed tank type reactor with natural uranium as fuel, demineralised light water as primary coolant, heavy water as moderator and graphite as reflector. [Sketch-I] The reactor vessel is made of aluminium and has 199 lattice tubes rolled into top & bottom tube sheets. It has an expansion joint between the top tube sheet and the shell to allow for thermal expansion. The reactor vessel is surrounded by two annular rings of graphite reflector, two rings of cast iron thermal shields and a ring of barytes concrete biological shield. Water-cooled aluminium and steel thermal shields are provided on top and bottom of the reactor vessel. Concrete biological shields are also provided on top of the reactor. The uranium fuel rods are cooled by demineralized light water circulating in a closed loop. Shut down core cooling is provided by one pass gravity assisted flow of water from a storage tank located at higher elevation. The heat generated in primary coolant, moderator and thermal shields is transferred in heat exchangers to sea water which is the ultimate heat sink. The graphite reflector, the cast iron thermal shields and the inner surface of radial concrete shield are cooled by reactor building ventilation system.

The reactor operated very efficiently with an availability factor of around 70 % till early nineties. After this ageing degradation of certain SSCs started affecting the reactor operation. Plant availability factor showed a declining trend due to frequent breakdown of equipments. Detailed performance review was therefore carried out on various equipments and a list of equipments that needed replacement was prepared. Equipments, for which availability of spares was becoming difficult due to obsolescence, were also included in this list. Detailed ageing studies were then taken up on various SSCs[1]. The SSCs were categorized into in-core components, important safety systems, important out of core components/systems and important structures & auxiliary system. Based on ageing studies and results of in-service inspections carried out, a detailed refurbishment programme was drawn up and a long outage of the reactor was taken for refurbishing the reactor.

The Reactor Vessel (RV) is one of the critical in-core components. Strict chemistry control of heavy water and cover gas helium in the RV is maintained within the specified limits to mitigate corrosion. The inventory of heavy water and helium systems is closely monitored. The ventilation air, which circulates in RV region, is also sampled routinely for tritium content to detect any heavy water leak. Out of 199 RV lattice tubes, two tubes which had earlier shown moderator leakage were plugged. Investigations revealed that the tubes did not have any generic defect. Tritium and helium gas sniffing carried out subsequently indicated that there was no leak in the RV & lattice tubes. Following ageing studies were carried out on the RV:

1. Eddy current testing of all the RV lattice tubes was carried out using an in house developed differential coil bobin type probe. The probe was made collapsible, as the diameter of the hole in the thermal shield just above the RV tube is lesser than the RV lattice tube diameter. About 10 % of the R.V tubes showed wall thinning of more than 20%, only a few of them showing thinning of more than 40 %. Detailed inspection of these tubes has been taken up using bore-scope also and surveillance checks on these tubes have been increased. Visual inspection of five selected tubes was also taken up using a micro video camera and no significant pitting damage on tube wall was noticed.

2. Inspection of outer surface of RV shell was carried out using the micro video camera and its condition was found satisfactory. At one location, where a hairline crack was noticed on the RV shell, plasticine impression was taken on modelling clay and the depth of the crack was analyzed to be insignificant.

3. As sample coupons of the RV material were not irradiated in core, one flow tube of an isotope irradiation assembly which was made of same material as RV and had seen a total neutron fluence comparable to RV shell was used for testing. Samples made from the irradiated flow tube were examined for embrittlement caused due to fast and thermal neutrons. Uniform elongation reduced to 3.7% but a local ductility of 42% reduction in area is retained. It was noted that the tensile properties showed a tendency of saturation

thereby indicating that the RV shell material will have sufficient ductility for continued safe operation.

4. RV expansion joint, which is exposed to neutron irradiation and is subjected to cyclic thermal stresses was also assessed for its fatigue life.[2] A finite element analysis using NISA computer code was performed and fluctuating stresses were checked to be well below the endurance limit of the irradiated expansion joint material.

Detailed studies were carried out for estimating stored Wigner energy in graphite reflector. Transient temperatures in the reflector were measured after reactor trip from 4 MW and 10 MW with reduced cooling air flows.[3] These data were utilized for developing a computational model for predicting steady state and transient temperatures of graphite following a reactor trip from 30 MW and 40 MW power levels with normal as well as reduced cooling air flows. Experiments were then carried out at these power levels and the predictions were found to be in excellent agreement with the experimental observations.

Measurement of the stored Wigner energy in graphite reflector was carried out. Two graphite blocks from the reflector were taken out during refurbishing outage and stored Wigner energy level was measured using differential scanning calorimetry. These results indicated that there was no need to carry out in-situ annealing of the graphite reflector.

Water cooled aluminium & steel thermal shields provided on top and bottom of the RV were hydro tested to check their integrity. The weld joint between one of the two coolant inlet pipes and top plate of upper aluminium shield showed minor leakage. The leak was located by inserting an inflatable seal assembly inside the pipe, inflating it at different locations for sealing the pipe and then monitoring the leak. A special hollow plug having sealing rings of "C'-shaped cross section at the ends and a straight pipe in the middle was designed, developed and installed to isolate the leaky portion of the pipe. As the plug had to be installed remotely inside the 1 1/4" NB (32 mm ID) coolant inlet pipe and direct access to the pipe was not possible due to site constraints, a flexible shaft assembly was used for installing and expanding the sealing rings in-situ to isolate the leaky portion.

Helium loss from the cover gas system had gradually increased and the leak was identified to be from the non-isolable zone of the reactor. Detailed helium and tritium sniffing were carried in the reactor pile region and the leaks were located from helium piping flange joints located between the upper steel thermal shield and the biological thermal shields. These flange joints having rubber gaskets are located in a 200 mm inaccessible gap about 4 M below top of pile. Replacement of these gaskets required removal of long test loop assembly from the reactor, various piping and a number of concrete biological shields having over 300 pile holes and weighing about 15 tonnes each. Realignment of these holes was expected to pose problems due to 40 year old dowels and dowel holes. Remote repair to leak was therefore considered and a split sealing clamp (SSC) having tapered inner faces was developed. On tightening the clamp on a mock up set up the tapered faces compressed the flanges closer together to rectify the leak even with old and broken gaskets. A full scale mock up station was set up for development of the SSC, various tools for site measurements and tightening the SSC and for trials on remote repair operation. The SSCs were lowered using the central 150 mm dia pile hole, shifted sideways by about 1.5 M, manoeuvred around the flange joints and fixed on the flange joints. The SSCs were lowered and manoeuvred using a number of nylon strings like a marionette. The remote viewing was achieved by inserting two video cameras through pile holes to the flange location and connected to video monitors located on top of pile. Detailed piping analysis using computer code Caesar - II was carried out to finalize the extent and the sequence of tightening. SSCs were fixed on all the eight flange joints and tightened using specially developed tightening tool. After completing tightening of all the flange joints the helium leakage has come down to less than 20 %. It is expected that the leaks will come down further during reactor operation.

Detailed inspection and ageing checks for various components of reactor regulating and protection systems, shut down core cooling system, electrical power supply system and containment system were carried out and refurbishing requirements were accordingly planned and implemented.

Ageing assessment & inspection of important out-of-core components of primary coolant system and moderator and cover gas systems was carried out in detail. During inspection it was found that the cork pad foundations of the primary coolant recirculation pumps had sunk due to ageing. Consequently the load of the pump-gear-motor set was getting transferred to system piping resulting in frequent misalignment of pump-gear-motor sets and failures of gear sets. Accordingly, replacement of the cork pads for all pump-gear-motor sets was taken up. In addition, as the normal maintenance frequency on these pump-gear-motor sets had increased due to ageing, it was decided to replace the old pump-motor sets with new sets.

Tube bundles of primary coolant heat exchangers were replaced with new ones as frequency of tube leaks had increased. Towards assessing the condition of inside surface of the primary coolant loop piping, a one meter long piece of pipe from the hot leg of coolant piping was cut and subjected to metallurgical examination. The corrosion rate which had resulted in reduction in wall thickness was calculated and checked to be well below the design allowance. This was attributed to maintaining of excellent water chemistry for the primary cooling system. Metallic pipe expansion joints of primary coolant system piping were checked visually and by liquid penetrant test and found healthy. All the elastomer sealing rings of the pipe couplings were replaced as they were in service for a very long time. All the subsoil pipelines were exposed by excavation and the condition of external protective coating as well as the pipe was assessed for any external damage due to coastal environmental conditions. Based on the inspection, the external coating of bituminous material was replaced with cold applied protective coating tapes for ease of installation. Portions of pipes with pitting damage were suitably repaired or replaced. Pipe coupons identical to actual pipes were buried subsoil for future inspection and testing. Wave-guides were also provided at suitable intervals on the subsoil pipes for acoustic emission testing to locate any leak in the future.

During hydro-test of core coolant outlet header made of SS-347, one of the cross headers showed a 270 deg. circumferential crack. In situ metallography in the cracked portion revealed sensitized microstructure and evidence of inter granular stress corrosion cracking due to presence of a weld joint in the pipe and its heat affected zone (HAZ). The cross header was cut removed from site and inspected in detail using Ultrasonic Testing method and in situ metallography. The pipe was seen to have cracked at the weld HAZ between the SS 347 pipe and a small spool piece made of SS 304. The original weld had only partial penetration. The SS 304 portion was cut removed and replaced. As the cross headers have a number of branching for connecting them to fuel channels, their geometrical locations, orientations and dimensional tolerances were strictly maintained. Suitable fixtures were developed and a mock up test was carried out to determine weld shrinkage and distortion. The cross header was then rehabilitated.

A portion of SS-304 piping for heavy water and helium system showed signs of stress corrosion cracking probably due to occasional ingress of trichloro-ethylene, which is used in refrigeration circuit of freezer driers in helium system for recovery of heavy water. Detailed pressure testing of the piping was carried out and the damaged pipe segments were replaced.

Ageing studies were carried out on important civil structures and auxiliary systems and their refurbishing requirements were finalized. Major civil structures like the 400 feet high exhaust air concrete stack and the 8,50,000 gallon capacity emergency water storage concrete tank (which stores demineralised water for shut down core cooling) were inspected. Core samples were taken from the concrete structures to assess the compressive strength, carbonation, chloride and sulphate depths of concrete. Ultrasonic pulse velocity and rebound hammer tests were conducted to check the homogeneity and integrity of concrete. Results of these tests indicated that all the important civil structures are in a healthy condition. Seismic evaluation of important civil structures was carried out to check that their design meets the current safety standards. During seismic evaluation it was found that the junction of the central inspection shaft and the cupola slab in the emergency water storage tank needed strengthening. Based on this finding, seismic strengthening of this civil structure was carried out by jacketing the central shaft and cupola slab joint with steel plates and epoxy grouting.

Many safety upgrades were undertaken during the refurbishing outage. A new fire detection and protection system was installed. This included a fire alarm system, elaborate fire hydrant piping system, fire retardant coating on cables and installation of fire barriers between safety related equipments. Physical separation and providing electrical power supply from separate power buses were carried out for emergency cooling water storage tank make up pumps. Automatic ground fault location system was incorporated for quick location of ground fault. Major modifications were also carried out in failed fuel detection system and radioiodine removal system. Control room instrumentation has been replaced with microprocessor based systems.

At Tarapur Atomic Power Station (TAPS), two nos., each of 160 MWe capacity boiling water reactors have been in service since 1969 and have been operating at high capacity factor. The plant uses direct cycle steam generating system in which steam produced in the reactor vessel is directly fed to the turbine generator for electric power generation. Extensive Ageing assessment studies are planned in line with Atomic Energy regulatory Board recommendations towards renewal of authorization for continued operation of the station. Accordingly, Plant management with the help of various specialists is presently carrying out ageing studies to check for any degradation in SSCs, specially those which can not be replaced or can be replaced with lot of difficulty only after extensive development work. Based on the studies, Ageing Management Programme (AMP) for the TAPS has been formulated in line with IAEA Technical Series report no. 338, IAEA TECDOC-540 and other relevant documents. SSCs that are important to safety have been identified with a view to detect degradation due to ageing and to take advance action through extensive maintenance for repair or replacement. These SSCs have been classified as Major Critical Components, Important systems and other critical components and further classified as not replaceable, replaceable with re-engineering and replaceable on routine basis. For each such component, the mode of degradation has been identified, ageing assessment has been done and corrective action plan has been defined.

Elaborate In Service Inspection (ISI) programme exists and was repeatedly carried out for the Reactor Pressure Vessel (RPV) and other in-core components.[4][Sketch-2] RPV material condition has been found satisfactory on the basis of tests on RPV surveillance specimen and the results indicate that it has adequate fracture toughness to assure the safety of the Pressure Vessel till the end of service life of 40 full power years. Fatigue analysis for the reactor pressure vessel has also been taken up considering its ageing due to thermal and pressure cycles. The condition of reactor internals and core shroud is satisfactory. The excellent coolant water chemistry has ensured negligible corrosion of materials of construction. The primary coolant system pressure boundary is inspected as per ISI programme and its condition is satisfactory. In order to improve safety with respect to inter-granular stress corrosion cracking of SS 304 material, a large portion of SS 304 piping has been replaced with improved quality material SS 304 LN. Reactor Containment is periodically leak tested and the leak rate has been well within the specified limit. However since integrated containment leakage rate testing is done at a pressure lower than the design pressure, measurement of wall thickness of dry well and assessment of various containment penetrations has been taken up. Also, simultaneous outage of both the units of TAPS has been planned to carry out inspection of water submerged components of suppression pool such as vent headers, down comers, pool liner and supports etc. to assess the ageing related degradation and take necessary steps to repair/ replace such components.

Important systems (engineered safeguard system and support systems) are being routinely monitored and are in good condition. The piping, valves, pumps and instruments are regularly inspected, their condition monitoring is done and repair/ replacement actions are taken as & when required.

Other critical components such as safety valves are inspected periodically as per inspection programme and are maintained in good condition. Reactor feed pumps, main exhaust fans, emergency ventilation fans etc. are inspected under performance monitoring programme. Balance life of cables connected to safety related systems has been assessed and replacement schedule has been prepared. Plant up-gradation/ modifications have been carried out from time to time on the basis of operating experience at TAPS and experience feedback from other NPPs.

Information exchange and experience feedback on various operation, maintenance, inspection and ageing management activities between Cirus and TAPS have helped in formulating and carrying out the ageing management programme for both the reactors. Experience gained in assessment of concrete structures, expansion bellows, foundations of structures, piping etc. has been mutually shared Experienced technical manpower with first hand knowledge of similar activities already carried out at Cirus participated in finalising the scope of work that needed to be done at TAPS within the stipulated time. Based on the review a list of SSCs that will need further studies for the completion of the activities and their acceptance criteria have been finalised. Experience gained in implementing Ageing Management Programmes at Cirus and TAPS would be very useful in updating and implementing such programmes for other ageing nuclear installations.

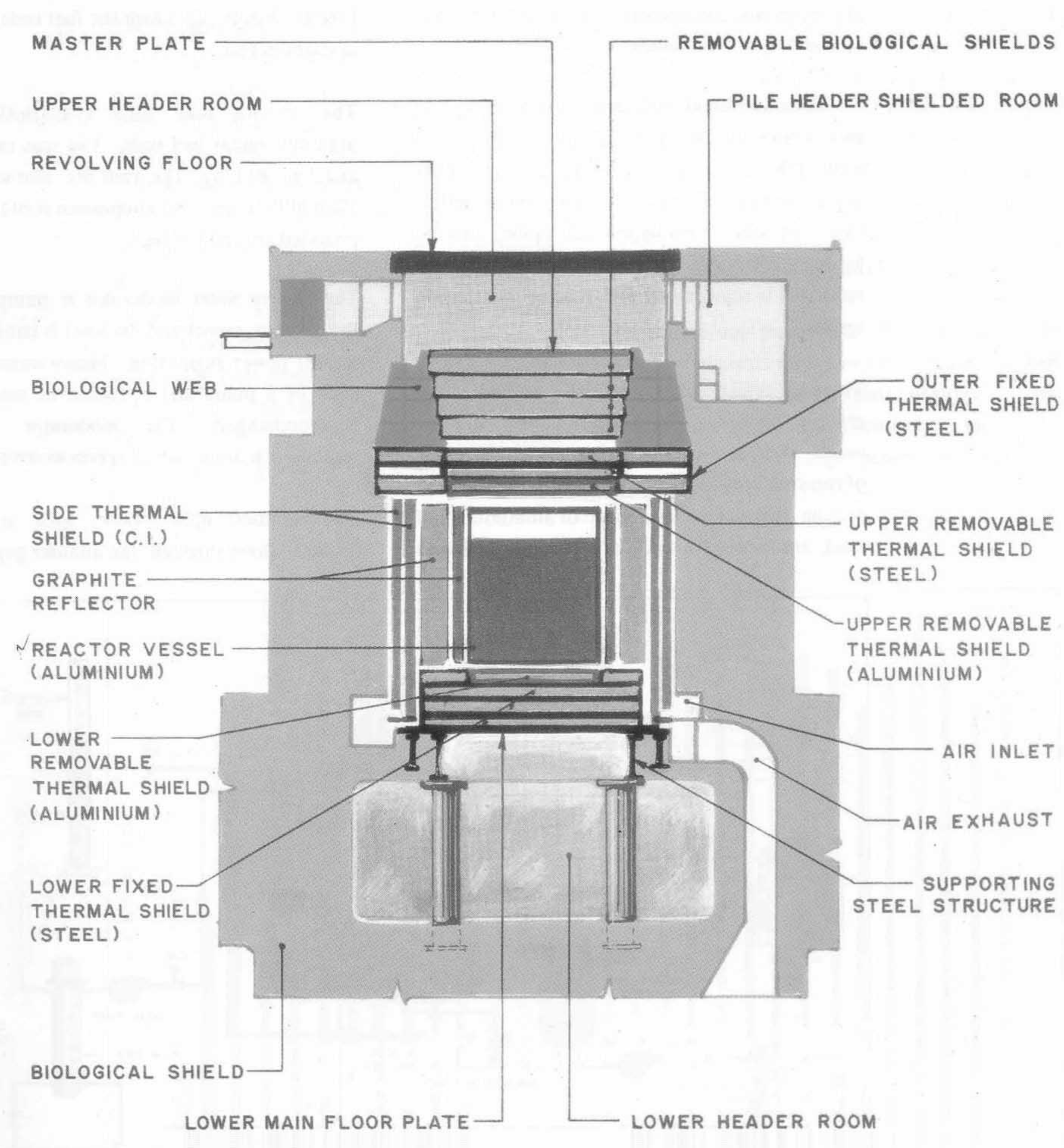
REFERENCES

[1] A. Bhatnagar, R. C. Pant, S. Chatterjee & J. N. Kayal, "Assessment of in-core components of Cirus after 30 years of service" Paper presented at The First national symposium on Ageing management of nuclear facilities, Jan. 13-15, 1994 (AMNF-94).

[2] W. V. Ramana, B. K. Dutta, H. S. Kushwaha, A. K. Sahu, A. Bhatnagar & R. C. Pant, "Ageing study of Cirus reactor vessel expansion bellow", Paper presented at AMNF-94.

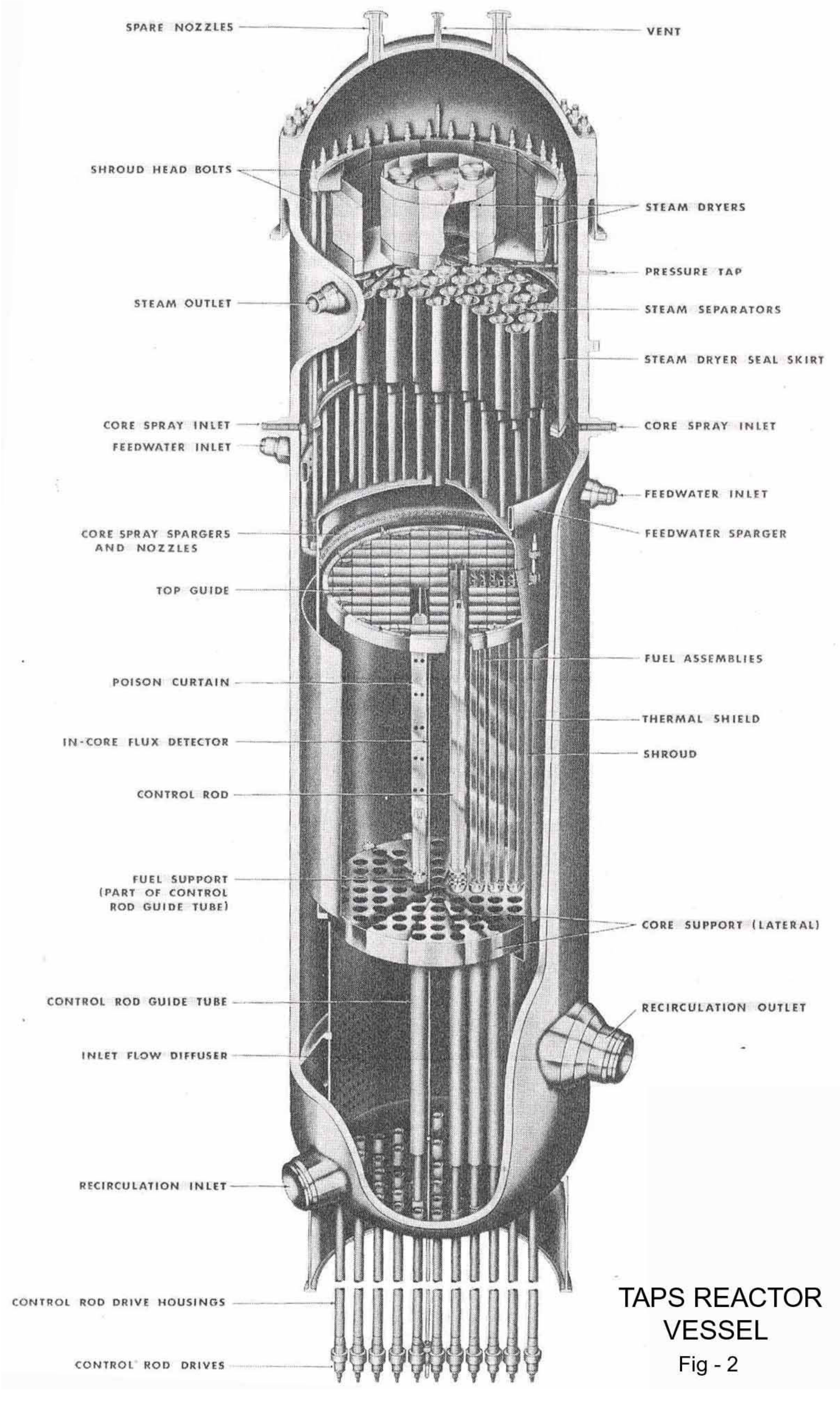
[3] S. K. Sharma, "Ageing assessment and refurbishment of the 40 MWt research reactor Cirus- An overview" IAEA/RCA Regional Seminar on Ageing Management (RAS/4/019)

[4] V. S. Srinivasan, Y. S. R. Prasad, " Ageing management of BWRs" Paper presented at AMNF-94.



CIRUS REACTOR STRUCTURE

Fig - 1



TAPS REACTOR VESSEL

Fig - 2

SPECIFIC ASPECTS FOR CERNAVODA – UNIT 1 NPP LIFE ASSURANCE

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Abstract:

The main scope of a Plant Life Management Program is to operate the NPP in a safe manner and at a competitive cost during the reactor life. To achieve this goal, it is important to continuously evaluate the degradation of its main structures and components. For Cernavoda NPP, a specific aspect is the long delay between the fabrication of the main components and the start-up. In evaluating the degradation history of these components, the conservation period must be observed.

BACKGROUND

Cernavoda NPP is a five units CANDU 600 PHWR, using heavy water and natural uranium. It is located in Constantza region, near the Danube - Black Sea navigable channel, at about 2 km South-East of Cernavoda town.

At present, unit 1 is in commercial operation (since 1997), unit 2 is in construction (with start-up target date in 2005) and units 3, 4 and 5 are in preservation (due to interruption of work).
Cernavoda NPP design life is 30 years. Compared with this target, the operation history is not long. It is still important to begin a plant life management program early to identify the critical components and structures, to establish the data needed for their monitoring.

SPECIFIC ASPECTS

Unit 1 construction began in 1979, with start-up target date in 1985. The next four units were planned to follow after short periods of time. This ambitious program was combined with the intention to fabricate as much as possible components in Romania (for units 3, 4 and 5 it was planned an assimilation factor of almost 100 %). The situation became critical after 1982, when the external credits for Romania from west were blocked. But the Romanian industry could not produce the equipments in the desired rhythm, so the start-up was delayed.

After 1989, the decision was taken to continue the construction of the Cernavoda NPP and in 1991 a new contract with AECL – Ansaldo consortium was signed. Before resuming the work, an evaluation of the state of the plant was performed and the components of the plant were tested for acceptance. These tests demonstrated that the components were good “as they are”, not good “as new”.

All these “historical” facts resulted in a long delay between the fabrication of the main components and the start-up. Most components were procured 10 – 15 years before start-up. First criticality was achieved in 1996, but the containment perimeter wall sliding was complete in 1983, the Calandria vessel was installed in 1985, the Steam Generators were in position in 1987, the fuel channels were installed in 1989. In evaluating the history of these components for a Plant Life Assurance Program, the conservation period must be observed.

PREVIOUS ACTIVITIES

In 1994, CITON began a program of ageing studies for the components of Cernavoda NPP – Unit 1. These studies referred to the improvement of knowledge about ageing mechanisms, the selection of critical components, data collection and data processing.

Also a special program for Environmental Qualification up dating of the safety related components has been developed by CITON and the owner of the plant. This included the review of the Environmental Qualified (EQ) component list, assessment reports for the EQ components, and establishment of programs for maintenance, inspection and monitoring of these components. Some specific EQ procedures have already been developed.

The installation of a multi-channel Data Acquisition System at Cernavoda NPP - Unit 1 is the scope of an ongoing IAEA Program (ROM 04/027).

PLANT LIFE MANAGEMENT PROGRAM

The first steps of a Plant Life Management Program for Cernavoda NPP – Unit 1 started in CITON this year.

The main proposed goals of the PLIM program are:

- to operate the plant in a safe manner all along its whole life;
- to maintain high capacity factors so to provide energy at a competitive cost;
- to preserve the option to extend the life of the plant.

The strategy that we adopted to achieve these goals is:

1. Identify the critical components and structures, important for the plant life management. Evaluation must refer to safety, reliability, and economics;

2. Define the safety margins inside which the components work satisfactory from the point of view of the regulatory requirements;

3. Evaluate the actual state of the critical structures and components, by reviewing the existing records regarding their history (design, fabrication, installation, preservation, operation conditions, maintenance) in order to establish a reference database;

4. Identify the degradation mechanisms, based on the previous ageing studies and find the methods to mitigate the degradation of the long-life passive components and establish a Condition Oriented Maintenance Program (instead of the Time Oriented Maintenance, which is now used) for the short life active components; and

5. Develop a specific monitoring and maintenance program.

IDENTIFICATION OF CRITICAL COMPONENTS AND STRUCTURES

The first step is to identify the components and structures that are important for the plant life management. Evaluation must refer to safety, reliability, and economics. For these components and structures, the potential degradation mechanisms must be identified.

For the selection of the critical components and structures we used the following criteria:

- safety criteria – components and structures whose failure can cause a release of radioactivity or which have to mitigate the release of radioactivity in case of a failure.
- availability criteria - components and structures whose failure can cause a shut down or a reduction of power of the plant.
- replacement criteria – components and structures that cannot be replaced or the replacement is difficult and costly.

In a first stage, based also on the experience of other Candu NPPs and on referenced IAEA TECDOCs [1] to [4], we identified the components presented in Table 1. These components and structures will constitute the subject of the first phase of the PLIM studies.

New components will be added in a second phase, such as pressure vessels, heat exchangers, other pumps, actuated valves, air compressors, instrumentation and control systems, electrical systems etc.

Table 1 Critical components and structures

Long-life passive components	Calandria vessel
	Calandria internals
	Fuel channels
	Steam generators
	Feeder pipes
	Reactor cooling pipes
	Pressurizer
	Containment Building
	Concrete and steel structures
	Cables
Short life active components components	Reactivity control drive mechanisms
	Turbine
	Generator
	Diesel generator
	Heat transport pumps
	Local air coolers

A few comments on some selected components follow:

Calandria vessel

The Calandria vessel was originally installed through the "A" opening into the perimetral wall, and the Calandria vault was then raised around it. It's not feasible to replace the Calandria vessel, but, because of the low functional parameters and the design allowances, no problems are expected for the life period of the plant.

Calandria internals

For the Calandria internals (Calandria tubes, reactivity mechanisms thimbles) problems may be caused by contact between the Calandria tubes and the reactivity thimbles. This can result in Calandria tubes cracking. Affected internals can be replaced.

Fuel channels

Fuel channels major problem is the sag which can lead to a contact between the Calandria tube and the pressure tube. This contact will produce cracks in the pressure tubes. The problem can be avoided by the correct positioning of the garter springs and the control of the elongation of pressure tubes. For Cernavoda NPP, the new (compared to older Candu reactors that faced this problem – see Pickering A for exemple) design of the garter springs help preventing the contact between the Calandria tubes and the pressure tubes. Some Candu NPPs performed already the total or partial retubing. However, this operation is difficult and has high costs. For exemple, for Pickering A, the retubing was the most expensive upgrading operation - see reference [4] .

Steam generators

Steam generators have been introduced inside the containment through the "B" opening. They can be replaced by reopening the containment, which is a difficult and with high cost operation. Pluged tubes seem to be the major potential problem. Some internals on secondary side may also be affected and need to be monitorized. Chemistry control is the most important factor to avoid problems. Monitoring technics for the degradation of the internal components of the steam generators must be developed.

Feeder pipes

Feeder pipes can face an unacceptable wall thinning. due to flow accelerated corrosion. Periodic measurements of the wall thickness are necessary. The chemistry control, as for the steam generators, can mitigate the degradation process and extend their service life.

Containment Building

Containment Building is one of the few irreplaceable components of a CANDU NPP and it provides the final protection to the public against radioactive emissions. The containment is also expansive to repair. Both concrete and reinforcing and post-tensioning steel components can be affected, so the possible degradation processes must be defined.

Other concrete and steel structures

Eventual repairs needed by other concrete and steel structures may be expensive and difficult to do. That is way we have included them into the first selection. During the development of the program, the critical structures will be defined.

Active short life components

Active components are components with a shorter life than the plant and which can be repaired or replaced. Defining the ageing mechanisms of these components is important for establishing the condition oriented maintenance program.

FUCTURE ACTIVITIES

Once defined the critical components and structures that are included in the first phase of the PLIM program, a review of the design will be done to evaluate the design safety margins.

The next step will be the review of the existing records regarding the history of their fabrication, installation, preservation, operation conditions and maintenance in order to evaluate the actual safety margins and to obtain a reference database. From this database together with the results of the ageing program in defining the degradation mechanisms will result the necessary monitoring and maintenance actions.

REFERENCES

[1] IAEA TECDOC 1084/1999 - "Review of selected cost drivers for decisions on continued operation of older nuclear reactors: Safety upgrades, lifetime extension, decommissioning".

[2] IAEA TECDOC 1123/1999 - "Strategies for competitive nuclear power plants".

[3] IAEA TECDOC 1141/2000 - "Operational Safety Performance Indicators for Nuclear Power Plants".

[4] IAEA TECDOC 1309/2002 - "Cost Drivers for the Assessment of Nuclear Power Plant Life Extension".

Session 3
SPECIAL ASPECT OF PLIM:
ECONOMICS AND END OF LIFE CYCLE

Sub-Session 3.1

Chairpersons

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United States of America

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Russian Federation

EDF DECOMMISSIONING PROGRAMME - A GLOBAL COMMITMENT TO SAFETY, ENVIRONMENT AND COST EFFICIENCY OF NUCLEAR ENERGY

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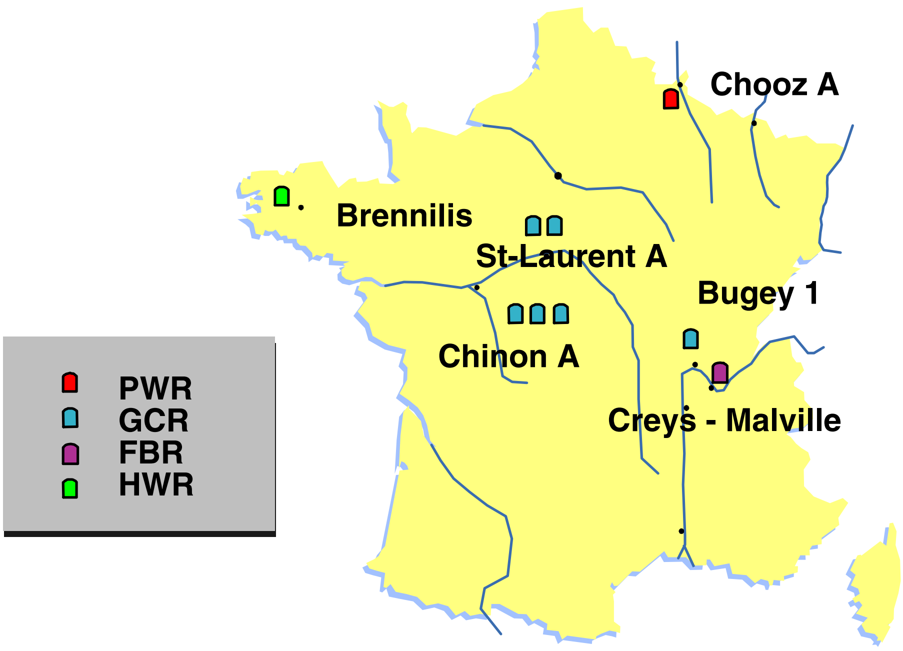
Abstract

Nowadays, decommissioning of nuclear power plants has become a key issue for nuclear industry in Europe. The phasing out of nuclear energy in Germany, Belgium and Sweden, as well as the early closure of nuclear units in applicant countries in the frame of EU enlargement, has largely contributed to consider decommissioning as the next challenge to face. The situation is slighty different in France: Nuclear energy is still considered as a safe, cost-effective and environment friendly energy source and EDF is still working on the development of a new generation of reactor to replace the existing one. Nevertheless, to achieve this objective, it will be necessary to get the support of political decision-makers and the acceptance of public opinion.

1. EDF DECOMMISSIONING STRTEGY

EDF has 9 of its nuclear power plants which have been definitively shutdown and which are currently under decommissioning.



Most of them are first generation units which started operating in the 60s and were definitively shutdown at the end of the 80s or at the beginning of the 90s mainly for economical reasons. They were not competitive against the new type of reactors under erection at this time (PWR 1300 MW and N4 series).

Unit	Reactor type	Capacity	Operation Life
Brennilis	HWR	70 MW	1967/1985
Chinon A1	GCR	70 MW	1963/1973
Chinon A2	GCR	200 MW	1965/1985
Chinon A3	GCR	480 MW	1966/1990
St Laurent A1	GCR	480 MW	1971/1992
St Laurent A2	GCR	515 MW	1972/1994
Bugey 1	GCR	540 MW	1971/1992
Chooz A	PWR	300 MW	1967/1991
Creys-Malville (Superphenix)	FBR	1240 MW	1986/1996

Until January 2001, EDF's policy regarding the dismantling of its decommissioned nuclear power plants was to reach "level 2" (release of non-nuclear facilities) about 10 years after final shutdown and to postpone final dismantling for another 30-40 years to take advantage of radioactive decay. This strategy was satisfying 3 categories of stakeholders :

- The owner, because expenses were deferred,
- The operator, because there is still some activity on site
- The regulatory body because decision about final storage solutions could be postponed

Only public opinion was suspicious about the real possibility to return to green fields in a reasonable timeframe.

Today, EDF considers that, if the nuclear option is to remain open, it is necessary to deal with increasing public opinion concerns for environmental and waste management issues. EDF and the nuclear industry have thus to demonstrate their ability to control the back end of nuclear power plants life cycle. Therefore, EDF decided one year ago **to achieve total dismantling of all nine already shutdown reactors in the next 25 years**. This new strategy will provide the tangible demonstration of the feasibility of dismantling, from the industrial, waste disposal and financial (adequate funding) points of view.

There are several benefits to this more aggressive strategy :

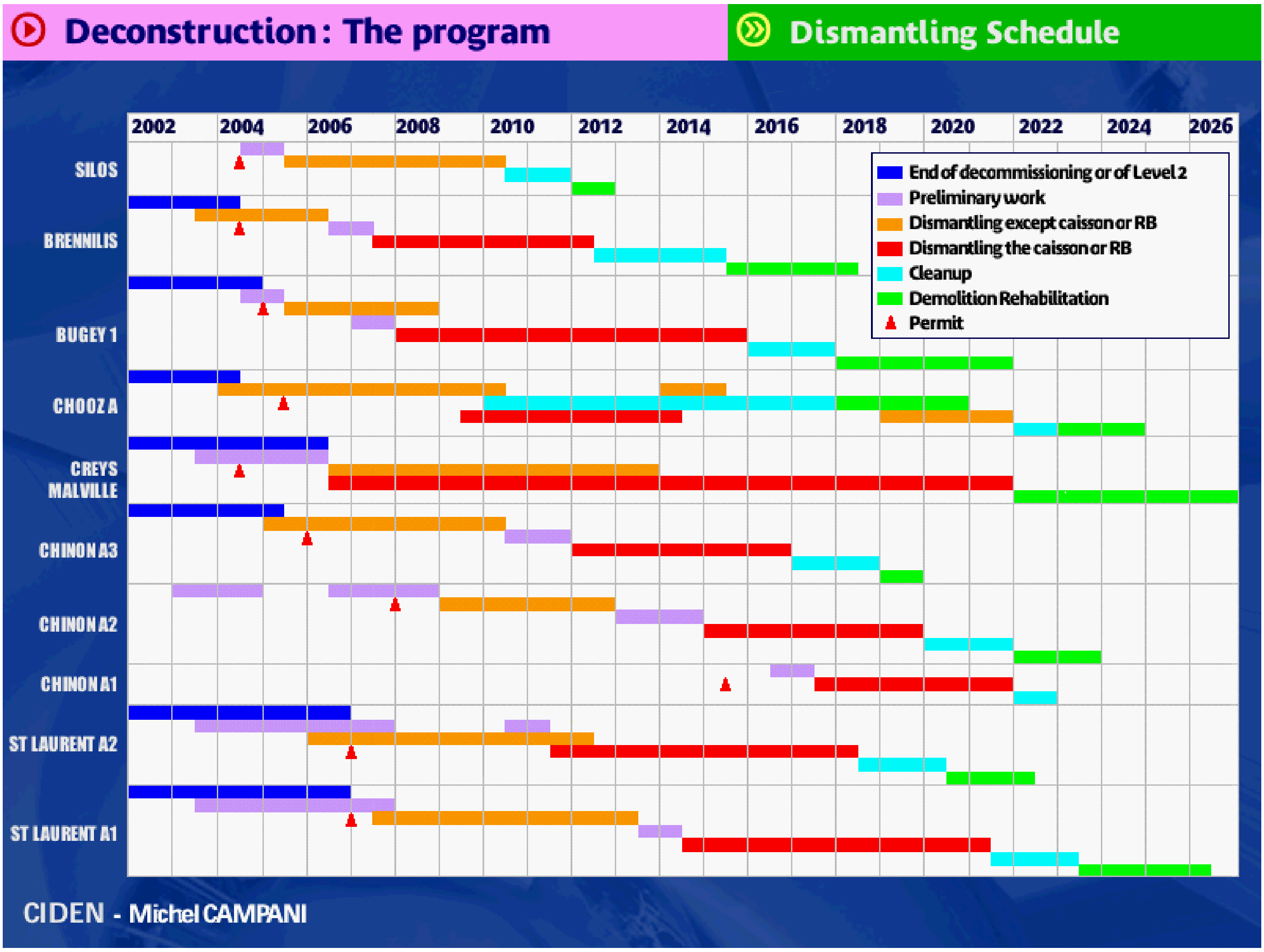
- It will allow addressing safety- and environment-related issues as yet unresolved.
- The cost of dismantling first generation units will already have been met when comes the time to invest in the renewal of the operating PWR park.
- Last, it will also provide the opportunity for structuring the industrial organization and preparedness (engineering and industrial) on which to rely for the final dismantling of the existing PWR park beyond 2020 (32 units).

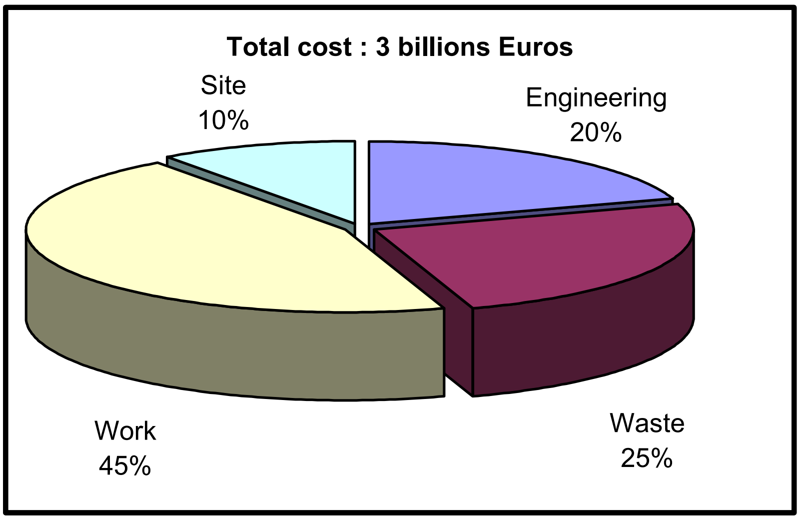
To implement this strategy, EDF decided in 2001 to set up an new Engineering Department, CIDEN (French acronym for Decommissioning and Environment Engineering Department), with 2/3 of the activity of its 400 employees dedicated to decommissioning.

2. EDF DECOMMISSIONING PROGRAMME

The decommissioning programme of the 9 EDF units already shutdown has to be completed in 2025. It will be organised in two stages :

1. The first stage includes :
   - Final dismantling of Brennilis (green fields) in 2015
   - A dismantling demonstration of a PWR reactor building (Chooz A) before starting replacing the population of PWRs currently in operation
   - Final dismantling of reactor containment of a GCR (Bugey 1) as a first of its kind
2. The second stage includes :
   - Dismantling of following 5 GCR units (Saint-Laurent A1&A2, Chinon A1,A2 & A3)
   - Final dismantling of Chooz A and Bugey 1 in 2025





Programme cost breakdown.

The successful implementation of this programme relies on :

- The simplification of regulatory processes and procedures (3 authorisations are currently needed to cover all the decommissioning process)
- The availability of treatment, conditioning and disposal facilities for specific categories of wastes (graphite, sodium, long-lived, …)
- An effective nuclear industry (contractors and suppliers) that will ensure the technical, cost and schedule aspects of this programme.

Availability on time of waste solutions is of the utmost importance. Among them, the main critical issues are :

- Opening of a Very Low Level Wastes disposal in 2003 (130 000 tons)
- Opening of a new disposal for graphite and radiferous wastes (17 000 tons) in 2010
- Opening in 2007-2008 of a centralised interim storage facilty for long-lived Medium Level Wastes (500 tons including filters, control rods, reactor internals for example)

In order to secure the execution of the decommissioning programme EDF is considering the possibility to erect “buffer” storage facilities on site to mitigate the impact of potential delay in the licensing and commissioning of the new facilities.

Regarding the closely concerned and related issue, namely disposal of high-level radioactive waste (HLW), the so-called « 1991 Bataille » Law defined three prospective investigations that were to be carried out before 2006. Waste transmutation (CEA mission with assistance from EDF), sub-surface storage and deep geological repository (under ANDRA responsibility).These possibilities are all open and under investigation and EDF intends to be active in all issues.

Because its responsibility as nuclear operator is at stakes, but also because in fine it will have to bear the cost of waste disposal, EDF is becoming more and more involved in these projects.

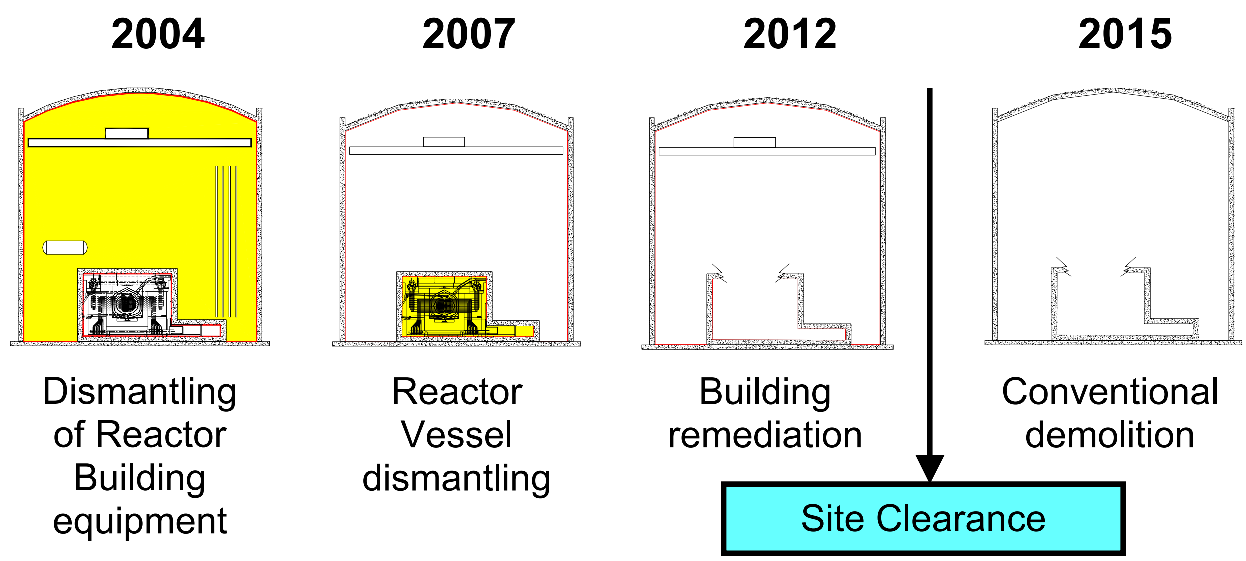
3. ACHIEVEMENTS AND CURRENT ACTIVITIES ON SITES – NEXT STEPS
BRENNILIS – HWR

BRENNILIS will be the first EDF NPP to be fully dismantled with a definite release of the site.

Dismantling activities started in 1997 with the dismantling of electromechanical equipment from auxiliary buildings and the conditioning of dismantling wastes. These wastes are now shipped to a final repository operated by ANDRA (Centre de Stockage de l'Aube) or to a melting and incineration facility (CENTRACO).

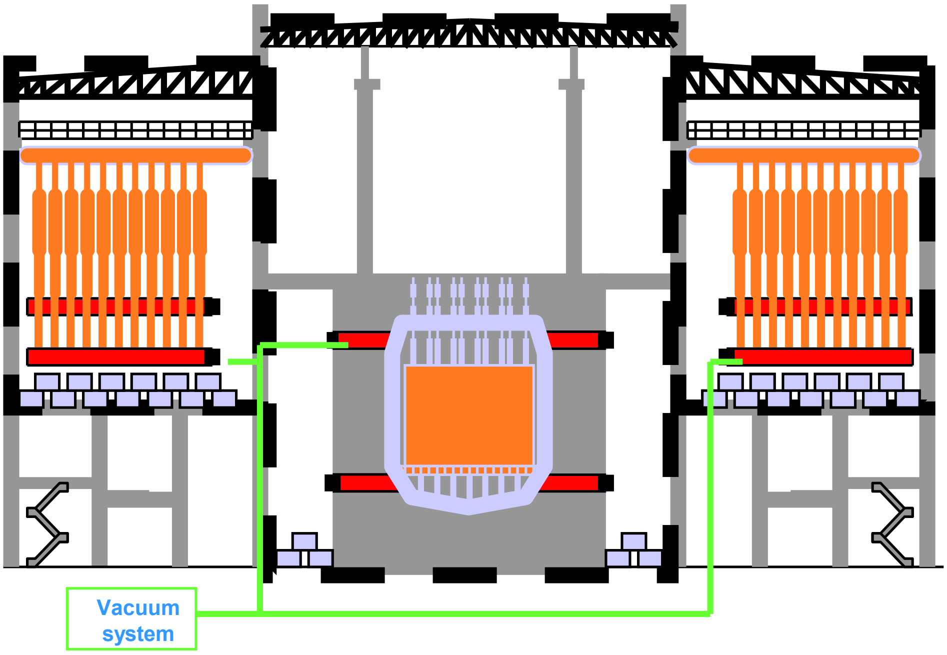
The remediation and demolition of auxiliary buildings started in 2000 and will be finished in 2004. In order to reduce the volume of radwastes to be treated, the contamination is removed from concrete walls by hands-on or remote techniques. Thus the remaining structures can be considered as conventional wastes. The buildings are then demolished with conventional techniques and the rubbles are unconditionally released. The French Safety Authority has approved this procedure and a first nuclear building has been demolished in April 2002.

The studies for the final dismantling of the reactor building have started this year and will be finished in 2004. The objective is to start the dismantling of the rector vessel in 2007 and to complete it in 2012. After remediation of the reactor building, its demolition will start in 2015. At this time the site will be cleared from any nuclear regulation and will be rehabilitated



CHINON A 2 – GCR

As a result of EDF previous decommissioning strategy, CHINON A 2 has reached the level 2 of the decommissioning process defined by IAEA. The conventional island has been demolished and most of the nuclear electromechanical equipment has been dismantled as to reduce the safe enclosure perimeter. Only the reactor vessel and the heat exchangers are remaining (see drawing below). They will be dismantled after the dismantling of the BUGEY 1 reactor containment which will consitute a first of its kind for EDF GCRs.



Current status : SAFE ENCLOSURE

BUGEY 1 - GCR

Last of the graphite-moderated reactors built by EDF, Bugey 1 was shutdown in 1994. Partial dismantling involved with decommissioning work is expected to be completed by the end of this year. Removal of graphite cladding packages started this year. The studies for the dismantling of the reactor building started this year. The works will start in 2008 to be completed in 2015.

CHOOZ A - PWR

The Chooz A NPP is located in the Ardennes region and was definitively shutdown in 1991. It is the first PWR unit involved in a deconstruction program. A unique feature of this site is that the reactor and its auxiliaries are installed in two rock caves excavated in a hill.

Demolition of conventional island has started this year and dismantling work in nuclear auxiliary buildings located on the top of the hill is under way. After remediation, the buildings will be considered as conventional buildings Studies for the dismantling of equipment located in the two rock caves will start in 2003. Dismantling works of the reactor vessel will start in 2006 to be completed in 2014.

4. CONCLUSION

The increasing mobilisation of EDF for the decommissioning of its already shutdown NPPs shows its willingness to demonstrate its capacity to control the nuclear life cycle from end to end. The successful implementation of its decommissioning programme will not mean the end of nuclear energy as an efficient way to generate electricity but it will constitute a pre-requisite for the erection of new nuclear power plants in France.

ELECTRIC UTILITY DEREGULATION - A NUCLEAR OPPORTUNITY

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Abstract

The implications of electric deregulation are and will continue to be pervasive and significant. Not only will the fundamental monopoly regulatory concepts of managing electric utilities change but deregulation will have a profound and dramatic impact on the way electric generating plants are managed and operated. In the past, under the various approaches to financial regulation, the economic benefits normally attributed to competition or that would have otherwise been derived from competitive or open market forces, were assumed to be embodied in and inherent to the various processes, methods and principles of financial oversight of utility companies by regional, state and municipal regulatory authorities. The result was often escalating electric prices and over supplies of electric capacity, by justifying unnecessarily high reserve margins based on long planning horizons (typically 20 years or greater) with extrapolated demand requirements that were generally in excess of what actually occurred over time. To the extent that a utility company justified to its regulators the need for additional capacity and the added expenses to operate it, the utility was usually reimbursed for its cost and profited accordingly. The consequence was that through regulation of electric rate design, the ultimate price electricity was determined by the aggregate of costs to produce it, independent of the forces of supply and demand. In the nuclear power industry, over time, this process, in conjunction with obvious requirements and imperatives for improvements in safety and reliability, not only had a profound impact on the costs to construct new nuclear generating facilities, but also on the cost to operate, maintain and decommission them. Over time, the cost to construct new nuclear power plants dramatically rose from $200 to $300 per KWe to in excess of $6000 per KWe in certain regions of the world.

INTRODUCTION

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In the past, under the various approaches to financial regulation, the economic benefits normally attributed to competition or that would have otherwise been derived from competitive or open market forces, were assumed to be embodied in and inherent to the various processes, methods and principles of financial oversight of utility companies by regional, state and municipal regulatory authorities.

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Although the regulatory process varied from country to country and region-to-region, the fundamental principles that influenced and ultimately determined the price or tariffs for electricity to consumers were generally the same. Utilities revenue requirements were founded upon complex cost of service formulas which emphasized and allowed the recovery of all "reasonable" costs including operating expenses, taxes, depreciation of investments and additionally assured a reasonable rate or return on all outstanding investments.

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THE REGULATORY PROCESS

Typically, under the various forms of regulated monopolies, a utility company, in exchange for an exclusive franchise to produce and sell electricity in a particular region, was obligated to provide an adequate supply to all consumers wanting it, at a price that was "just and reasonable". The determination of adequate supply and reasonable rates was a matter of interpretation by utility companies as well as their regulators. In essence, the ultimate economic benefits, normally attributed to price equilibrium, in balance with supply, demand and other market forces, were expected to be achieved and sustained through a complex, political process of financial regulatory oversight, in which utility companies were usually reimbursed for most if not all annual expenses or their "cost of service" and additionally allowed to earn a "reasonable" rate of return on plant investments. For example, regulated US electric utilities, based on the "Reasonable Revenue Requirements" and "Cost of Service" principles, are allowed by their utility commissions to earn a "Reasonable Return" on their investments. Although it is difficult to define what is reasonable compared to other expense components of utility cost, US regulatory commissions and the courts have attempted, over time, to establish a "Zone of Reasonableness" which is founded upon the standards of "Capital Attraction" and "Comparable Earnings". In general, the "Zone of Reasonableness" is such that the rate of return approved by a utility commission, should allow a utility to attract capital investments and also should be reasonable, compared to other non-regulated companies that have similar investments subject to similar risks.

SOME IMPLICATIONS OF DEREGULATION

With the transition to deregulation, the price electricity will ultimately be determined by economic factors within the market place. In the following paragraphs, a number of implications of operating nuclear plants in competitive electric markets are addressed and discussed.

Nuclear Asset Revaluation - Deregulation of electric utility monopolies in the US is becoming extensive, with many states having recently enacted deregulation laws. These laws typically require that electric utility monopolies "unbundle" their services and divest themselves of their generation assets through direct sale or competitive auction. The result has been that the asset value of these facilities is "revalued" by the competitive desire and forces of the marketplace to own and operate them in a truly competitive wholesale electric market. In certain cases, in exchange for their prior, exclusive franchise to generate and sell electricity in a specified region, the former owners are allowed to receive a reasonable return on those assets that become "Stranded" in the process of divestiture. Stranded being the difference between the net asset value of the power plant prior to divestiture and the price at which the facility is sold. Most nuclear units, recently sold in the US, have been sold at prices substantially less then their asset values prior to divestiture.

Shift of Investment Risk - Under deregulation, the risks, costs and rewards, associated with debt versus equity financing of nuclear plants will inevitably change. In order to attract sufficient capital at reasonable cost, the capital structure of an enterprise is intentionally chosen such that a portion of the total invested capital is financed through debt or the issuing of bonds and the remainder is financed through the issuing of additional common stock or equity. For the case of regulated nuclear utilities, the ratios of % debt to % equity, of total invested capital often exceed 50/50 or even 75/30. In this situation, with long term, assured returns, the risk associated with debt investment is relatively low and as such, so is the rate of return or interest rate. Typically in the US, the weighted average cost of capital for regulated utilities is in the order of 10 - 15 % including interest on long term debt in the order of 6-10 % (assured long term returns, low risk to investors, relatively low rate of return). Under deregulation, the financial risks associated with nuclear power plants operating in competitive markets are not clearly understood. It's likely that a greater portion of risk will be assumed by owners as opposed to debt holders and the capital structure, cost and rewards will change accordingly.

For regulated utilities, the annual depreciation expense (return of investment) is the net depreciation, on an annual basis, for the recovery of all invested capital over the life of the generating unit, frequently computed on a "straight line basis". Since, recovery of depreciation expense is generally assured by the regulator, the book life or the time period to recover the investment is usually the license life or operating life of a nuclear unit, which, in the US, is typically 40 years. Compared to an unregulated or competitive enterprise, 40 years of assured return of investment is a relatively long period of time to recover investments with little or no risk incurred. For generating companies operating in competitive markets, return of and return on investment are not as assured. Only to the extent that a nuclear plant can generate sufficient revenues, through the competitive sale of electricity and other generation products, at market prices, can there be sufficient income to cover all costs including depreciation and interest on debt, as well as sufficient earnings to provide a reasonable rate of return or dividends to the common (equity) shareholders. Also, the risks associated with competitively operating nuclear power plants in unregulated markets are not well understood. Consequently, the risks are relatively high as are the assumed rates of return. In recent economic evaluations for the operation of nuclear power plants, it is not uncommon to assume the weighted average cost of capital to be in excess of 15-20 % with typical book life to be in the order of 10 to 20 years.

Cost of Capital - The cost of capital and expected returns on investment may be significantly different for different companies and generating facilities that compete in the same market. In general, since the remaining life and the net asset value of any nuclear power plant may be different, the rates of return may very well be different, assuming all other costs and key variables were the same. In determining economic value, each situation must be carefully evaluated, based on its individual merits, taking into account assumptions for all significant economic and operational variables. These variables include: How much investment will be required to construct a new nuclear unit; What is the current asset value; What is the design Electric Rating (MWe); What levels of nuclear operating performance will be assumed (operating capacity factor, refueling outage duration, etc.); How will capital investment be financed (debt Vs equity); What are projections of annual Operating & Maintenance (O&M), Administrative & General (A&G) and Fuel Expense; What book life will be assumed for depreciation of nuclear assets; Will life extension be a consideration; What rates for cost escalation will be assumed; What levels of risk tolerance will be assumed for selling the output of the unit (high risk/reward – energy sold at spot market, low risk/reward – energy sold by power purchase agreements or bilateral contracts). Compared to the past, more sophisticated tools will be required to conduct effective analyses such as dynamic computer modeling of significant nuclear operational, financial and economic variables.

Consider a recent example in the US, New England marketplace. Over the past few years, deregulation laws in most New England states required that electric utilities divest their generation assets, through auction or negotiated sale. Approximately four years ago, Entergy, through negotiated the sale, purchased the Pilgrim Nuclear Power Station from the Boston Edison Company for a price of approximately $20/KWe. This price was significantly less than the original capital cost as well as the current asset value of Pilgrim station at the time of the sale. Since Entergy's price was clearly market driven, it's reasonable to assume that, in determining the value they were willing to pay, Entergy had to make certain assumptions about the future price of electricity, as well as all other significant financial and operational variables for operating Pilgrim Station. At the time of this sale, typical spot market projections for the wholesale price of energy and capacity, in New England, were in the order of $25/MWh and $15/KWe respectively. This equates to an "all in price" of approximately $31/MWh. Also, at the time of the sale, other large New England electric utilities, that had substantial equity positions and or operating experience in other nuclear generating facilities, showed little or no interest in purchasing Pilgrim, even at a bargain price of $20/KWe. Approximately two years later, in the same, but more mature wholesale electric market, Dominion Resources purchased, at auction, Millstone Station for approximately $550/KWe, the highest resale price paid for a US nuclear generating facility by that time. Apparently, Dominion's justification for this seemingly high capital investment was based on their energy price projections of approximately $40/MWh with expected returns in the 15% to 20 % range and nuclear performance at relatively high industry levels, typical of their past nuclear operating experience at Surry and North Anna. By US industry standards, both Entergy and Dominion (Virginia Power) are considered to be excellent nuclear operators. By 2001, the average wholesale price of energy, for the ISO New England electric market, was in the order of $65 to $70/MWh. All other financial and operational variables considered, if the current energy prices in New England are sustained, both Entergy and Dominion Resources stand to earn substantial profits on the purchase of these nuclear facilities.

Nuclear Plant Profitability - Traditionally, in typical regulated electric utilities, profit/loss or income statements as well as other financial measures, were prepared and evaluated at the "operating company" level or the holding company level and did not include detailed information at the nuclear plant level. With the advent of deregulation and competition, utilities and generating companies are now looking more carefully at the revenues and earnings of individual and combinations of nuclear generating units. Today, the practice of preparing detailed financial statements including income statements, at the generation unit level, is becoming more common.

Simplification of complex Regulatory Accounting - Under deregulation, Independent Power Producers and utility generating companies, which do not have utility monopoly status, would not be subject to the highly complex accounting requirements and methods that have evolved under utility regulation such as accelerated depreciation, tax normalization, and the methods required by utilities to account for investments, leases, expenses and ultimate consumption of nuclear fuel. Deregulation will simplify these processes.

Allocation of Corporate Expenses - The process of deregulation can have serious implications on traditional allocation methods previously used for allocating corporate administrative expenses (indirect O&M costs) to various operating functions including nuclear generation. The allocation of administrative costs to regulated and unregulated functions must be accomplished in a reasonable way to meet the "Reasonable" test of regulation as well as the "competitive" test of the marketplace. Theoretically, neither test will tolerate unreasonable and unnecessary costs. Hypothetically, consider a case where an electric utility is required, by deregulation law, to divest itself of a large nuclear asset, currently valued in billions of dollars. As a result of divestiture, it is not unreasonable to assume that the new asset value of that facility, based on market forces, may become substantially less (tens to hundreds of millions of dollars). Obviously, if the allocation of administrative expenses was originally based on asset value, either a new method must be devised or a substantial reduction in headquarters expenditures would have to occur, or both. In the US, in fact, deregulation and utility combinations and mergers, have resulted in extensive downward pressure on utility corporate headquarters administrative costs.

Advancement of Business Literacy – With the transition to competitive generation, nuclear plant management and analysts at all levels will need to embrace and implement financial and business concepts into the day to day operations of nuclear plants. A much greater sensitivity to and awareness of fundamental business and economic principles will be required. The development of sound business literacy will become an imperative for all nuclear plant decision makers.

MEASURES OF ECONOMIC VALUE AND ECONOMIC ASSUMPTIONS

With deregulation and competition, there will inevitably be a more serious and greater emphasis on the need for continuous process improvement and benchmarking activities within the nuclear generation industry. The metrics required to establish baseline process standards as well as those for continuing improvement will need to embrace, integrate and correlate key plant operational and safety as well as business and financial parameters. Just as the standardization and industry acceptance of safety and operational performance measures and the cooperative sharing of relevant performance best practices has, in the past, significantly contributed to industry advances in these areas, similarly, the standardization of economic measures, financial definitions and a keen sense of business literacy specifically applicable to the operation of nuclear power facilities will be required for the nuclear power industry to become and remain economically competitive.

Measuring economic performance for regulated electric utility companies has been a tradition at the corporate or utility company level. Financial performance beyond the straightforward measures of resource consumption or budget performance such as investment returns, revenues and earnings were also focused at the corporate level, often consolidating the individual performance contributions of electric generating facilities including nuclear power plants. Economic performance indicators specifically for individual nuclear generating plants were either not formally identified or generally not embraced and used by plant management and utility regulators. With the onset of competitive electric generation, the requirement for financial and economic measures of performance, specifically applied and focused at the generating plant level will be inevitable. With electric competition, individual nuclear plants will sell their output competing on electricity price, ultimately to ensure the safe, reliable and economic dispatch of their generation either into open spot markets or by competitive bidding for forward priced bilateral contracts. In either case, a sound and confident understanding of the operational and economic factors and key measures which gauge the competitiveness of an individual generating plant will be required. Economic and financial performance indicators will be needed for individual plants to measure, evaluate and continuously improve the operating and management processes needed to become and remain competitive in open electric markets.

Production Cost - In the US, Nuclear Production Cost is defined as Nuclear O&M plus Annual Fuel Expense divided by the Net Generation of the unit. Typically, the units for Production Cost are calculated in cents/KWh or Mills/KWh. Unfortunately, Production Cost not only excludes Indirect Costs (A&G), a significant portion of annual O&M (based on recent EUCG nuclear data, indirect cost is approximately 24%, on average, of total nuclear O&M), but additionally includes none of the annual expenses associated with the plant capital investment such as interest on debt and annual depreciation expense. Similar measures to Production Cost, which do include capital related costs, are “Busbar Cost” and “Going Forward Cost”. These measures do include capital carrying costs and, to a greater extent than Production Cost, represent the “total Cost” of producing electricity at a nuclear unit. Although frequently used when making economic comparisons and analyzing industry trends, Production Cost should not be used to represent the total cost of producing electricity on a cents/KWh basis. It does, however, provide an effective measure of the variable and controllable costs for the operation and maintenance of nuclear units (see Figure 1 Below). These costs are critical in evaluating and judging the past and future economic performance of nuclear power plants, among themselves, as well as against other generation options. Yet, as many electric utilities transition to deregulated electric markets, in which the returns on nuclear investments are no longer assured, Production Cost continues to be used as a key measure for judging the economic competitiveness of nuclear power facilities. In the US Nuclear Energy Policy Report, recently published in May of 2001, US nuclear Generation is presented to be “competitively priced” against Oil, Gas and Coal electric generation, based on Production Cost alone.

Although US nuclear plants' average Production Cost has generally improved over the past several years (in excess of 25% improvement, based on 1995-1999 EUCG Nuclear O&M data, see Figure 1 below), production cost alone should not be used to characterize the competitive benefits of nuclear generation, especially in competitive electric markets.

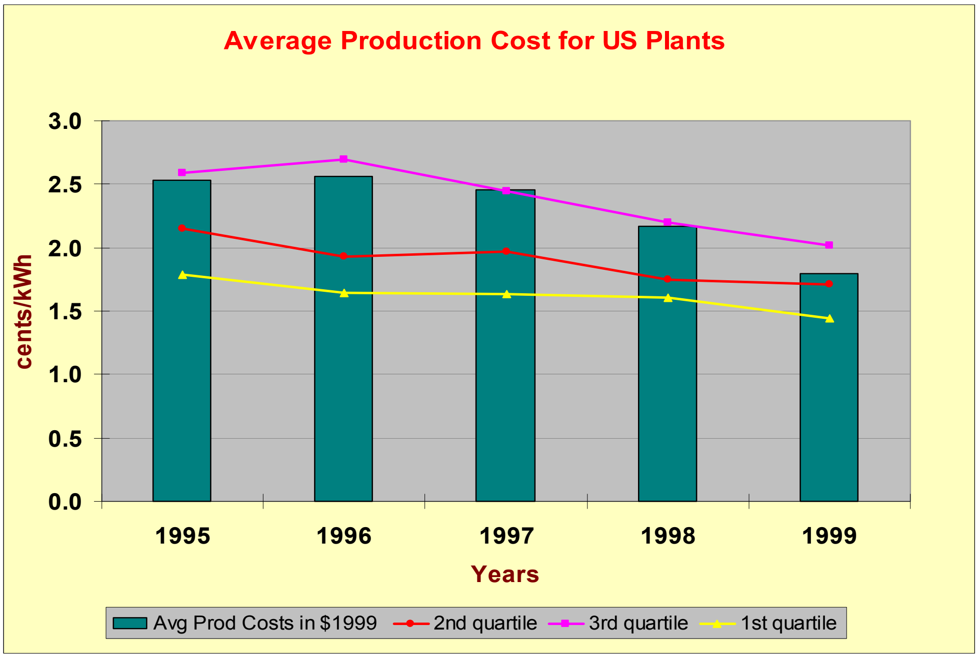

FIG.1. Average Production Cost for US Nuclear Power Plants.

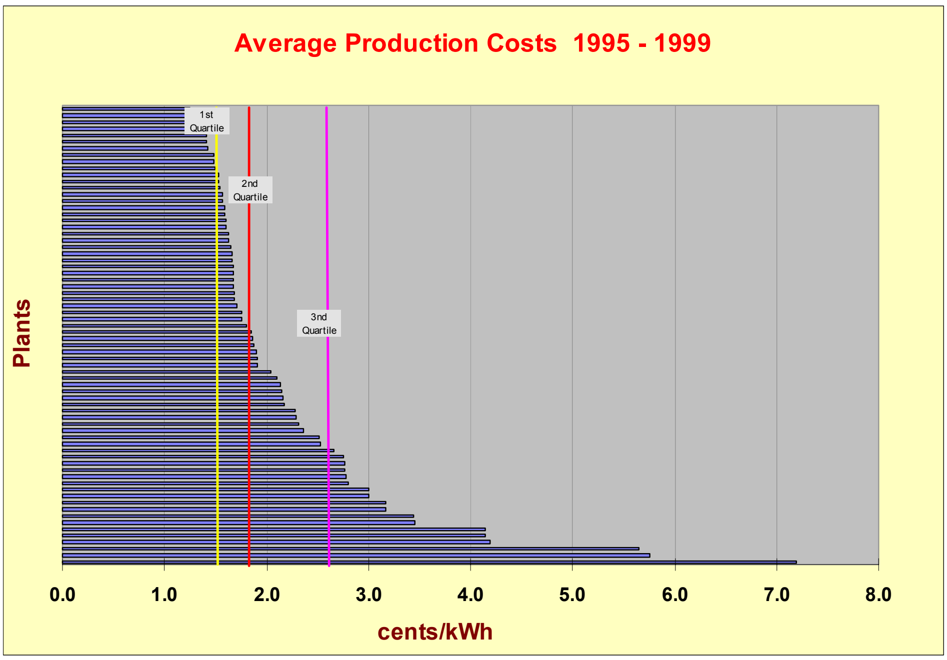

FIG. 2.US Nuclear Plants - Production Cost Ranking.

Income Statement Analysis - With the advent of deregulation and competition in US electric markets, utilities, Independent Power Producers (IPP's) and generating companies are now looking more carefully at the revenues and earnings of individual and combinations of nuclear generating units. Today, the practice of preparing detailed income statements at the generation company/ nuclear unit level is becoming more common. These income statements typically include depreciation based on the outstanding asset value of the plants. This annual depreciation expense would reflect the unrecovered cost of the initial capital investment as well as subsequent capital improvements.

To the extent that a generation unit was sold/auctioned and revalued for competitive purposes, the annual depreciation would only reflect the new value of the unit. Below (see Figure 3) is a representative pro forma income statement, prepared at the nuclear unit level, that was used for financial planning purposes.

	2000	2001	2002	2003	2004
ENERGY VALUE					
Generation (MWh x 1,000)	6,503	5,899	6,485	5,899	6,503
($/MWh)	30.2	29.7	28.8	29.9	31.0
Energy Revenue	196,132	175,258	186,776	176,142	201,334
CAPACITY VALUE					
Capacity (MWe)	1,150	1,150	1,150	1,150	1,150
($/KWe)	16.7	19.0	22.7	23.5	24.4
Capacity Revenue	13,047	14,823	17,733	18,382	19,063
OPERATING REVENUE	**209,179**	**190,081**	**204,509**	**194,525**	**220,397**
OPERATING EXPENSES					
Nuclear O&M	38,301	54,347	41,188	58,443	44,292
A&G	11,286	11,703	12,136	12,585	13,051
Nuclear Fuel	31,556	28,984	31,898	29,379	32,421
TOTAL EXPENSE	**81,143**	**95,035**	**85,222**	**100,408**	**89,764**
EBITDA	**128,037**	**95,046**	**119,287**	**94,117**	**130,633**
DEPRECIATION	57,789	4,196	4,552	4,941	5,367
TAXES					
Federal Income	19,114	27,296	36,219	27,899	39,541
State Income	4,428	6,323	8,391	6,463	9,160
Municipal-Property	9,348	5,166	1,438	1,528	1,613
TOTAL OPERATING EXPENSES	**171,822**	**138,016**	**135,821**	**141,239**	**145,445**
OPERATING INCOME	37,357	52,065	68,688	53,285	74,952
INTEREST EXPENSE	1,859	1,373	1,424	1,472	1,518
EARNINGS FOR COMMON	**35,498**	**50,692**	**67,264**	**51,813**	**73,434**

FIG. 3. Typical Nuclear Unit Pro Forma Income Statement.

Return on Investment **-** The Return on Investment of a regulated electric utility is similar to the profits provided to investors of unregulated companies. The Return on Investment is comprised of interest on debt and the return on equity. Return on equity is the profit or dividends paid to the shareholders of the utility as a percentage of the equity portion of outstanding investment. Expressed as a rate, the return, when multiplied by the rate base or net outstanding asset value of the utility, should produce sufficient revenues to pay the interest on debt, provide a profit for the equity owners and, at the same time, continue to allow the utility to attract capital investment. The dollar amount and at times the equivalent (weighted average) rate of return are often referred to as the Weighted Average Cost of Capital. The total amount of money invested for a utility which is allowed, for ratemaking purposes, is its rate base or total capitalization. The capital structure of the utility refers to the proportion or mixture of debt (investment financed through the issuing of bonds) versus equity (investment financed by the issuing of common stock), expressed as a percent of debt financing versus a percent of equity financing. The risks and the costs associated with debt versus equity financing are quite different. This is primarily because the risk and consequently the cost of equity financing are greater than those of bonds. In the financial marketplace, stock investors will demand a premium in compensation for this additional risk. For the case of regulated nuclear utilities, to achieve least cost financing and consequently reasonable electric rates, the ratios of % debt to % equity of total invested capital often exceed 50/50 or even 75/30. In this situation, with long term assured (not guaranteed) returns, the risk associated with debt investment is relatively low compared to that of equity and, as such, so is the rate of return or interest rate. Typically, in the US, the Weighted Average Cost of Capital for regulated utilities is in the order of 10 - 12 % including interest on long term debt in the order of 6-8 % (assured long term returns, low risk to debt investors, relatively low rate of return). To further understand the concept of the Weighted Average Cost of Capital consider the following hypothetical example (see Figure 4 below) of an electric utility with a total capitalization of $300 million which is comprised of $180 million of 8% bonds and $120 million of equity financing with a cost of equity (return on) of 15%.

	Amount Financed ($ Millions)		**Rate of Return** (%)		**Cost** ($ Millions)
Debt	180	X	8.0	=	14.4
Equity	120	X	15.0	=	18.0
Total	300	X	10.8	=	32.4

Weighted Average Cost of Capital
Expressed in $ = $32.4 Million
Expressed as % = 10.8%

FIG. 4. Sample Calculation -Weighted Average Cost of Capital.

From a nuclear power plant perspective, instead of rate base of the company, the net outstanding investment would be the average outstanding asset value of the power plant for the period in question. In this case, the outstanding investment would be the sum of all capital investments including interest during construction, less accumulated depreciation up to the point in time of the calculation. For any given year, the average outstanding investment would be the net outstanding investment at the beginning of the year plus its value at the end of the year divided by two.

For unregulated utilities (typically electric generating companies) fundamentally, the definition of return is the same as above, with the following clarifications. Return on Investment is not as assured. Only to the extent that a company can generate sufficient revenues, through the competitive sale of electricity and other generation products, at market prices, can there be sufficient income to cover all costs including Return of Investment (Annual Depreciation Expense) and interest on debt, as well as sufficient earnings to provide a reasonable rate of return or dividends to the common (equity) shareholders. Also, the risks associated with competitively operating nuclear power plants in unregulated markets are not well understood. Consequently, the risks are relatively high as are the assumed rates of return. In recent economic evaluations for the operation of nuclear power plants it is common to assume the Weighted Average Cost of Capital to be in excess of 15-20 % with typical Book Life assumed to be in the order of 10-20 years.

AN ECONOMIC ANALYSIS

In deregulated markets, the most sensitive variable in evaluating the economics and determining the competitiveness of nuclear power is the price of electricity. One might imagine, what level of electricity price would encourage a utility or owner to invest in a new or existing nuclear plant? Or, what additional level of capital investment is justified in extending the life of a nuclear unit 20 or more years, assuming some level of operational and economic performance and electricity price? In my opinion, an effective analysis of this type can only be accomplished through dynamic computer modeling of all significant nuclear operational, financial and economic variables. The answer to these questions is highly dependant on an individual companies' economic and financial circumstances as well as their assumptions associated with other key operational and financial variables for a particular nuclear unit. These variables, as was discussed previously, include: How much investment will be required to construct a new nuclear unit or extend the life of an existing one; What is the design Electric Rating (MWe); What levels of nuclear operating performance will be assumed (operating capacity factor, refueling outage duration, etc.); How will capital investment be financed (debt Vs equity); What are projections of annual Operating & Maintenance (O&M), Administrative & General (A&G) and Fuel Expense; What book life will be assumed for depreciation of nuclear assets; Will life extension be a consideration; What rates for cost escalation will be assumed; What levels of risk tolerance will be assumed for selling the output of the unit (high risk/reward – energy sold at spot market, low risk/reward – energy sold by power purchase agreements or bilateral contracts) and so on.

For example, hypothetically consider a 1000 MWe nuclear unit with projected performance (cost and operational) at levels at or above the top decile of recent US nuclear industry performance and with the following additional assumptions (see Figure 5 below).

Initial Capital Investment	$2000/MWe
Capital Structure	25 % Debt, 75 % Equity
Book Life for Depreciation	20 years
Annual Escalation Rate	3.7 %
Average Return on Equity	20 %

Figure 5. Nuclear Plant Key Economic Assumptions.

Through computer modeling, the energy price consistent with the above assumptions would be approximately \$70 - \$80/MWh.

An alternate approach to evaluating this situation (see Figure 6 below) would be to determine the range of nuclear performance, investment levels and returns on investment that would support a specified level of energy price. For this approach, energy price is not treated as a controllable variable.

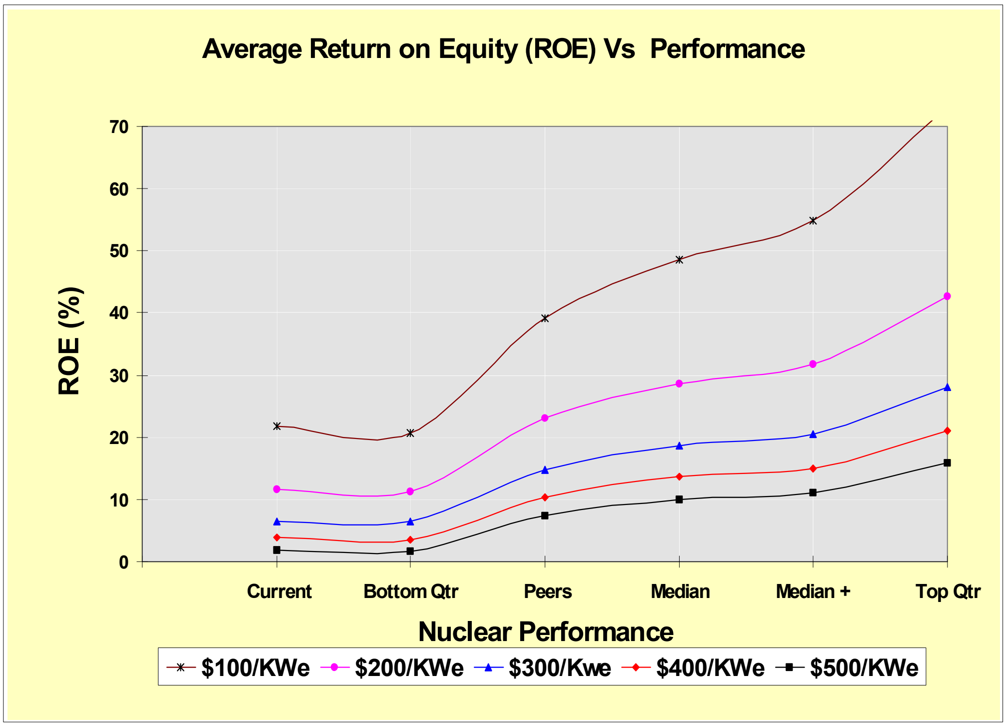

FIG. 6. Nuclear plant investment opportunity, as measured by Return on Equity (ROE) investment, for varying levels of plant performance and investment @ \$30/MWh electricity price.

THE CASE FOR PLANT LIFE EXTENSION

Under deregulation, the economics to justify a nuclear plant life extension decision will require complex evaluation of not only Production Cost but additionally other key economic and financial variables including Return on Investment, Return on Equity, Busbar Cost, Revenues, Net Earnings, etc. Economics should be evaluated on a unit-by-unit basis, not only with respect to current and projected nuclear performance, but additionally by careful comparison to other non-nuclear generation alternatives.

In accordance with US law, nuclear power plants are licensed by the NRC to operate for a period 40 years with an option available to renew their operating license. The License Renewal Rule establishes detailed requirements and a process to renew nuclear plant operating licenses for up to 20 additional years. The focus of this process is to require the utilities to manage the potential adverse effects of aging on the key operational and safety systems and components of nuclear units. In addition, utilities are required to evaluate the potential environmental impact for having extended the license life of a unit. A US licensee may apply to renew their operating license from 20 to 5 years prior to the current license expiration. By the end of 2000, the US NRC approved 20-year extensions to the operating licenses of Baltimore Gas & Electric's Calvert Cliff, 2 units and Duke Power's Oconee, 3 units. Approximately 30 of the 103 US operating units have either applied or have informed the NRC of their intent to apply for license renewal by 2003. By 2015, the initial operating license for 45 of the 103 US units will have expired. The cost associated with applying for and obtaining NRC approval for life extension is estimated to be in the range of $10 million to $40 million. These costs do not include the additional refurbishment costs that may be required as a result of the life extension approval process. Estimates for nuclear plant refurbishment have been published in the $200 million to $400 million range, per unit, and typically will require substantial capital investment.

In considering the economics of nuclear plant life extension, a brief discussion of the accounting of depreciation for electric utilities is helpful. The appropriate accounting of depreciation is particularly important to the electric utility industry in light of the substantial investments in plant operating assets. Consequently, depreciation represents a major portion of the revenue requirements for the utility and implicitly, to a large degree, impacts the electric rates charged to utility customers. As such, there are several governing utility principles which determine how utility assets are depreciated and ultimately how annual depreciation expenses are reflected in electric rates. In determining a utility's "Cost of service", the allowance for annual depreciation not only directly impacts the allowable or recoverable annual expenses, but additionally, depreciation methods have a significant impact on electric rates and revenue requirements which lead to the utility's profits.

Depreciation and amortization expenses, regardless of the methods employed, are calculated on property and plant that is "Used and Useful" in providing public utility service. Only to the extent that a capital investment meets the "Used and Useful" test, can that investment be capitalized and placed in service. And, in consideration of a particular service life, the annual depreciation allowance can be determined, through the use of various depreciation methods, and reflected in utility electric rates. To the extent that the purchase of plant or equipment is useful for a period of time, which is greater than one accounting period or one year, its cost must be apportioned over the period of time for which the investment is "Useful" or provides service to customers. The various depreciation methods translate the cost of plant and equipment to annual depreciation expenses, which are allocated over successive accounting periods. Any item, whose service life covers several accounting periods, should have its cost or service value apportioned over those periods through an appropriate depreciation method.

In the case of US nuclear power plants, under utility regulatory principles, the use of straight-line depreciation of plant assets over the license life of forty years, becomes the compelling approach used by utility regulators in achieving "Just and Reasonable" rates for electric customers. Similarly, under the above principles of utility regulation, it becomes obvious why long service lives at first appear so attractive when evaluating the economics of nuclear generation to other generation alternatives. Annual depreciation, as an expense, is minimized with straight-line depreciation over such extended service lives. This effect is accentuated when considering the possibility of life extension of US nuclear plants with an additional service life of twenty years and anticipated capital refurbishment investments in the range of $200 to $400 million, a fraction of the original construction costs of these plants. Unfortunately, in competitive unregulated environments, the risk associated with value that is based on forecasts and extrapolations made so far into the future (at least twenty years) may become questionable and unacceptable. Under regulated utility principles, electric utilities could, with minimum risk, depend on the recovery of substantial investments with relatively assured returns or profits over extended time periods such as the forty- year license life of a nuclear power plant.

Glossary of Business Literacy Terms

The following glossary includes definitions of financial and economic terms used throughout this paper and its presentation at the International Symposium on Nuclear Power Plant Life Management. In addition, other definitions have been included which are relevant to the finance, accounting and economics of nuclear power generation.

A&G Costs: Administrative and General Costs - Corporate overhead costs covering items such as pensions, benefits, legal, human resources, tuition refunds, transportation, and similar costs.

Accelerated Depreciation: Any depreciation method resulting in greater amounts of depreciation expense in the early years of a plant asset’s life and lesser amounts in later years. Examples are double-declining-balance and sum-of-the-years’-digits methods.

Accumulated Depreciation: The cumulative amount of depreciation recorded against an asset or group of assets during the entire period of time the asset or assets have been owned.

Allowance for Funds used during Construction (AFUDC): A non-cash item representing the estimated composite interest costs of debt and a return on equity funds used to finance construction. The allowance is included in the property accounts; the contra credit is included in income. This portion of the carrying value of property (along with the rest) is included in a utility company’s rate base and is recovered through revenues over its useful life.

Allowed (allowable) rate of return: The rate of return (to be determined to rate base) which the regulatory commission sets to determine electric rates.

Amortization: The general process of allocating acquisition cost of assets to either the periods of benefit as expenses or to inventory accounts as product costs. Called depreciation for plant assets, depletion for wasting assets (natural resources), and amortization for intangibles. Also used for the process of allocating premium or discount on bonds and other liabilities to the periods during which the liability is outstanding.

Amortize: To periodically write off as an expense a share of the cost of an asset, usually an intangible asset.

Asset Management: A process for making resource allocation and risk management decisions at all levels of a business to maximize profitability and value to all stakeholders.

Asset: A property or economic resource owned by an individual or enterprise.

Balance sheet: The financial statement of a firm that lists the assets and liabilities at a point in time.

Benchmarking: Process of identifying best practices by comparing one's own performance to the best in the industry. Comparisons include process performance (cycle time and efficiency), cost measures as well as other indirect measures of performance. Benchmarking strives to improve one's own practices by implementing change. To be more like those of top performers.

Betterment: The replacement of an existing asset portion with an improved or superior asset portion.

Book Costs: Original cost of property, as reflected in utility company records.

Book Depreciation: This represents the amount of money which must be set aside annually to recover the capital cost over the anticipated life of the facility. There are several methods of depreciation; emphasis will be upon the "straight line" method. Using the straight-line system of depreciation, this component is constant each year.

Book Value: The carrying amount for an item in the accounting records. When applied to a plant asset, it is the cost of the asset minus its accumulated depreciation.

Busbar Costs: The total costs associated with supplying electricity at a generating station. Components include Operations and Maintenance costs, Fuel Expense, capital carrying costs, decommissioning costs and Administrative and General (A&G) Costs. These costs are usually expressed in cents per Kilowatt-hour.

Business Literacy: Understanding and awareness of business/financial terminology and the application of this understanding to daily processes.

Business Plan: A document linking overall company strategic goals and objectives to everyday work processes, including major steps about how to achieve them.

Capital Asset: Any item of property except (1) inventories, (2) trade notes and accounts receivable, (3) real property and depreciable property used in a trade or business, (4) copyrights or similar property, and (5) any government obligation within one year and issued at discount.

Capital budgeting: The plan for the coordination of resources and expenditures which will determine which projects a firm should undertake.

Capital Expenditure: An expenditure that increases net assets.

Capital Intensive: A term used to designate a condition in which a relatively large dollar investment is required to produce a dollar of revenue. The electric industry, for example, has an investment of about $4.00 for each dollar of revenue generated annually.

Capital Structure: Mix of different securities issued by a firm.

Capital: Costs associated with an investment in a facility, usually financed, and can be depreciated. Capital return on investment and depreciation are amortized over the life of the investment as an expense.

Capital-intensive Business: One which requires a significantly greater investment in facilities than do other businesses.

Capitalization: Long-term debt, preferred stock, and owners' equity.

Cash Flow Statement: A financial statement consisting of cash receipts and disbursements, with a summary for the organization's net cash position. It reveals the sources and uses of a company's cash.

Cash Flow: The difference between cash receipts and cash disbursements over a specified period of time.

Common Dividends: A payment to common stockholders.

Competition: Freedom of economic choice in buying, selling, or exchange of goods and services.

Competitive Drivers: Factors which may impact, either directly or indirectly, a strategy or business plan. These could include such items as changes in the law, innovation, the entrance of new competitors and mergers or acquisitions within the industry.

Consolidated Financial Statement: Combined balance sheets, income statements, and statements of cash flows of a parent company and its subsidiaries.

Construction Work in Progress (CWIP): Plant not yet operational, which may or may not be included in a utility's rate base.

Cost Management: Controlling and being acutely aware of the expenses incurred in and/or required to operate a business.

Cost of Capital (Net): The return asked, or being asked, by investors for the use of their money, expressed as percentages of the capital funds.

Cost of Capital: The composite rate of cost for debt interest, preferred stock dividends and common stockholder earnings requirements. It is the composite of the cost of the various capital sources used to provide the facilities utilized in supplying utility service.

Cost of Removal: The cost of demolishing, dismantling, tearing down or otherwise removing electric plant, including the cost of transportation and handling.

Cost of Service (often referred to as "Revenue Requirements): Operation and maintenance expenses, depreciation and amortization expenses, and income and other taxes found just and reasonable by the regulatory agency of rate-making purposes plus, in the case of public utilities, and allowance for capital (usually computed by applying a rate of return to the rate base).

Cost-Benefit Analysis: A financial model used to determine if a project will be profitable by comparing the estimated project cost with the estimated project benefit.

Current asset: An asset whose useful life is less than one year, such as cash, securities and accounts receivable.

Debt Capital: Funds secured for a business by borrowing, such as through the dale of bonds.

Debt Expense: All expenses in connection with the issuance and initial sale of evidences of debt, such as fees for drafting mortgages and trust deeds; fees and taxes for issuing or recording evidences of debt; cost of engraving and printing bonds and certificates of indebtedness; fees paid trustees; specific costs of obtaining governmental authority; fees for legal services; fees and commissions paid underwriters, brokers, and salesmen for marketing such evidences of debt; fees and expenses of listing on exchanges; and other like costs.

Debt, Long-Term: Borrowed funds with a maturity (repayment date) occurring far in the future.

Debt-to-Equity Ratio: The total dollar value of business debt financing divided by the total dollar value of equity financing.

Decommissioning fund: For a nuclear power plant, a regular, annual set-aside of funds generated from operations, to support the eventual decommissioning of the plant when it is retired.

Deferred Charges: An expense that has been incurred but whose payment, for whatever reasons, has been put off until some time in the future.

Depletion: Closely related to depreciation, this refers to the actual physical consumption of property (e.g., a coal deposit).

Depreciable Life: For an asset, the time period or units of activity (such as miles driven for a truck) over which depreciable cost is to be allocated. For tax returns, depreciable life may be shorter that estimated service life.

Depreciation: The wearing out or loss of service value of property used in business operations.

Depreciation Expense: The annual allowance for the depreciation of property representing that portion that has been "used up" during the previous twelve months.

Depreciation Reserve: the paper account that represents the accumulation of yearly allowances for depreciation expense. The reserve is viewed as an asset and indicates that funds are (in theory) being set aside.

Deregulation: The relaxation of controls over business operations.

Direct Costs or Expenses: Costs that are easily traced to or associated with a cost object, for example, costs incurred by a department for the sole benefit of the department.

Discount Rate: An interest rate, measured as a percentage, used to convert future dollars into present dollars (discounting) and vice versa (interest compounding), according to standard net present value formulas.

Discounted Cash Flow (DCF): Analysis technique used in business to convert future cash flow estimates to their present (i.e., today's) value, using a discount rate. Related to the term net present value.

Discounted Cash Flows: The present values of a stream of future cash flows from an investment, based on an interest rate that gives a satisfactory return on investment.

Divestiture: The compulsory transfer or disposal of interests (such as stock in a corporation) by government order.

Earnings Before Interest and Taxes (EBIT): Standard measure of business performance; calculated as annual total earnings, before subtracting out tax payments and payments to debt holders. Also known as net operating income.

Earnings Before Interest, Taxes, Depreciation and Amortization (EBITDA): Similar to EBIT; calculated by subtracting only cash expenses from revenues. Depreciation and amortization are not subtracted out, as in the EBIT calculation.

Earnings Per Share: Net income available to shareholders divided by the number of shares of stock outstanding.

Earnings: Annual revenues minus annual operating expenses (including non-cash expenses such as depreciation and amortization).

Economic Life: The time span over which the benefits of an asset are expected to be received. The economic life of a patent, copyright, or franchise may be less that the legal life. Service life.

Economies of Scale: The principle that increased size of operations yields increased efficiency, as well as greater output.

Equity: Financial value of ownership or partial ownership of a company.

Fixed Cost: A cost that remains unchanged in total amount over a wide range of production levels.

Fixed expenses: Expenses that do not vary with levels of production, such as plant costs or salaries.

Fixed O&M Costs: O&M cost categories that are independent of the amount of energy generated by the plant.

Forward Price: Price of a commodity on offer today, at which a buyer can contract for delivery at some specified time in the future. For example, if the forward price of electricity for January 2003 is $75/MWh, a buyer can contract for that price today and be assured of getting electricity at that price on 1/1/2003, regardless of what the "spot" price is on that day.

Functional Depreciation: Loss of service usefulness or obsolescence due to technological advances or social requirements.

General and Administrative Expenses: The general office, accounting, personnel, and credit and collection expenses.

Gross Income: All revenues collected; the starting point for all income tax calculations.

Improvement: An expenditure to extend the useful life of an asset or to improve its performance (rate of output, cost) over that of the original asset. Such expenditures are capitalized as part of the asset's cost. Contrast with maintenance and repair.

Income (profit and loss) Statement (P&L): A financial statement showing a company's net income—the profit after deducting all expenses—over a period. Provides investors and creditors with information that helps predict the amount, timing and uncertainty of future cash flows. Accurate predictions of future cash flows help investors assess the economic value the company and creditors determine the probability of repayment of their claims against the company.
Income: Revenues received from sales and other operations of a business.

Indirect Costs: Costs of production not easily associated with the production of specific goods and services; overhead costs. May be allocated on some arbitrary basis to specific products or departments (A&G costs).

Intangible Asset: A nonphysical, non-current asset such as a copyright, patient, trademark, goodwill, organization costs, capitalized advertising cost, computer programs, licenses for any other preceding, government licenses (e.g., broadcasting or the right to sell liquor), leases, franchises, mailing lists, exploration permits, marketing quotas, and other rights that give a firm an exclusive or preferred position in the marketplace.

Interest: Regular payments (usually semiannually) remitted by bond issuers to bond holders for the use of borrowed money. Annual interest payments will be equal to the face value of the bond times its coupon.

Internal Rate of Return (IRR): Discount rate for which the present value of a company's or project's expected cash inflows equals the present value of the company's or project's cost; this rate gives an NPV of zero.

Liabilities: A firms obligations to pay its creditors sometime in the future.
Life Cycle Management (LCM): Process by which nuclear power plants integrate operations, maintenance, engineering, regulatory and business activities to manage plant condition, optimize operating life, and maximize plant value while maintaining plant safety.

Line of Sight: The ability to view how a companies corporate goals and objectives are being implemented throughout the organization.

Market price: Price at which a security or commodity is traded in the market. Electricity is traded at both the wholesale level and retail level as a commodity in a deregulated environment.

Market Rate of Bond Interest: The current bond interest rate that borrowers are willing to pay and lenders are willing to take for the use of their money.

Merger: A combination of two or more firms in which the assets and liabilities of the selling firm(s) are absorbed by the buying firm. Mergers are usually accomplished by either exchanging stock, cash purchase of assets or payment of debt or by some combination of these methods.

Municipal Ownership: A term applied when a business enterprise is owned and operated by a municipal government.

Natural Monopoly: An activity such as the provision of gas, water, and electrical service characterized by economies of scale wherein cost of service is minimized if a single enterprise is the only seller in the market.

Net Present Value Method: Annual revenue less all expenses including taxes but not book depreciation and return, discounted at an assumed rate of return (cost of money) to determine a present worth of incoming cash flow for comparison with the initial capital expenditure (an outgoing cash flow). Often used to determine a "go or no-go" decision for project implementation.

Net Present Value: Present (i.e., discounted) value of the cumulative future net cash flow generated by a company, plant, or project.

Net Utility Plant: The investment in utility plant less depreciation.

O&M Cost: Operating and Maintenance costs. Those expenses needed to operate and maintain a facility.

Obsolescence: Depreciation caused by technological improvements.

Oligopoly: A market where there is very limited competition.

Operating Margin: Difference between operating revenue per kWh (i.e., market price) and operating cost per kWh; a measure of how much cash can be generated to retire debt and cover related capital costs.

Opportunity Cost: A sacrifice made to gain some benefits; that is, in choosing one course of action, the lost benefit associated with an alternative course of action.

Option Value: Increment in net present value due to the right-not the obligation-to retire a plant before expiration of the original licensed term or to operate during a license renewal term; option value is always positive because an option will be exercised only if future conditions are favorable.

Overhead: The costs associated with support from non-electricity producing organizations.

Owner's Equity: The ownership interest in a business enterprise.

Payback Period: Annual revenue less all expenses including taxes (but not book depreciation or return), divided into the initial capital expenditure to determine the number of years required to equal or pay back the initial capital expenditure.

Physical Depreciation: Loss of service usefulness or life due to wear and tear from use or other causes, such as rust or rot.

Power Pool: A regional organization of electric companies interconnected for the sharing of reserve generating capacity and power production coordination.
Present value: The discounted value of future cash flows.

Price Cap Regulation: A rate-setting process whereby a ceiling is placed on the price of service instead of limiting the allowable rate or return.

Price Earnings Multiples (P/E): Standard family of financial indicators of business performance. The ratio of the stock price (as determined by the market) to earnings as represented by typical accounting measures of income, usually EBIT or EBIDA. Also known as price/earnings ratios. It is Wall Street's valuation of profitability.

Price Earnings Ratio: Market common stock price divided by the annual earnings per share of common stock. The market price used may be a spot price, or an average of closing or the high and low prices for a period and the earnings are for the corresponding period.

Pro Forma Statement: Financial statement prepared on the basis of some assumed future events; usually consists of an income statement, balance sheets and cash flow statement.

Production Cost: Costs assigned directly to the production of electricity. Electric generation production cost equals O&M cost plus fuel expense. It is normally expressed in cents per Kilowatt-hour.

Productivity: Amount of output generated per unit of input. In a power plant capacity factor (i.e., MWh generated per unit of MW capacity) is a measure of productivity.

Profit Margin: Difference between revenue per kWh (market price) and total cost per kWh (includes operating costs, debt payments, taxes and other corporate costs), a measure of cash generated for stockholders.

Profit: Remaining income after business expenses are paid.

Public Service (Utilities) Commission: State regulatory body governing the rates and practices of utilities.

Public Utility District: Political subdivisions which are independent of city and county government and are voted into existence by residents for the specific purpose of rendering a utility service.

Public Utility: A business enterprise rendering a service considered essential to the public and, as such, subject to regulation in the public interest, usually by statutory law.

Rate Base: Value of property upon which a utility is given the opportunity to earn a specified rate of return as established by a regulatory authority.

Rate Case: A proceeding, usually before a regulatory commission, involving the rates to be charged for a public utility service.

Rate of Return on Average Investment: The annual, after-tax income from the sale of an asset's product divided by the investment in the asset.
Rate of Return on Common Stockholder's Equity: Net income after taxes and dividends on preferred stock divided by average common stockholder's equity.

Rate of Return on Total Assets Employed: Net income after taxes, plus interest expense, expressed as a percentage of total assets employed during the period.

Rate of Return: The return earned or allowed to be earned by a utility enterprise calculated as a percentage of its fair value or rate base.

Regulation: Process whereby governmental powers are used to direct or control some phase or unit of economic activity.

Regulatory Agency: Governmental body that regulates enterprises in certain specified industries.

Retained Earnings: Dollar amount of assets furnished by earnings of the company that were not distributed as dividends.

Return Allowance: The rate of return designated by a regulatory commission for testing the reasonableness of rates.

Return on Assets (ROA): Earnings as a percentage of total assets; ratio of net income to total assets.

Return on Equity (ROE): Earnings as a percentage of stockholder equity; ratio of net income to common equity; measures the rate of return on common stockholders' investment. The profit earned for each dollar of shareholders equity.

Return on Investment (ROI): Annual revenue less all expenses including taxes and book depreciation but not return, divided by investment required.

Return on Net Assets: The profit earned on each dollar invested in assets.

Return: Represents the money required annually to compensate security holders for the funds provided as invested capital for the plant facilities. It consists of interest on debt, dividends on preferred stock and earnings on common equity. The return element is variable, being greatest initially and then declining over the years because a fixed cost of money, or rate of return, is applied annually to the net plant (total plant less accumulated depreciation). To adjust for the variability of this component, present worth techniques are employed to obtain an equivalent constant, annual return.

Revenue Requirement: That amount a utility must collect to pay expenses and provide a fair return to investors.

Revenue: Receipts from the sale of goods and services.

Risk Premium: Extra compensation paid to an employee or extra interest paid to a lender, over amounts usually considered to be normal, in return for their undertaking to engage in activities more risky than normal.

Risk: The measure of the variability of the return on investment. For a given amount of return, most people prefer less risk to more risk. Therefore, in rational markets, investments with more risk usually promise, or are expected to yield, a higher rate of return than investments with lower risk. Most people use “risk” and “uncertainty” as synonyms. In technical language, however, these terms have different meanings. “Risk” is used when the probabilities attached to the various outcomes are known, such as the probabilities of heads or tails in the flip of a fair coin. “Uncertainty” refers to an event where the probabilities of the outcome, such as winning or losing a lawsuit, can only be estimated.

Service Life: The period of time a plant asset is used in the production and sale of other assets or services.

Service Obligation: A term used to mean the obligations which are among the duties a public utility is to perform. They usually are considered include the duty: to serve all; to provide adequate service; and to render safe, efficient, and nondiscriminatory service.

Short-term Debt: Bank borrowings or bonds with less than the traditional 20- to 30-year maturities.

Spot Market: Commodity transactions whereby participants make buy-and-sell commitments of relatively short duration, in contrast to the contract market in which transactions are long-term.

Spot Price: Price of a commodity for immediate exchange at a specific point in time.

Station Use (Generating): The kilowatt-hours used internally at an electric generating station for purposes other than sale. Station use includes electric energy supplied from house generators, main generators, the transmission system, and any other sources for this purpose. The quantity of energy used is the difference between the gross generation plus any supply from outside the station and the net output of the station.

Straight-Line Depreciation: A depreciation method that allocates an equal share of the total estimated amount a plant asset will be depreciated during its service life to each accounting period in that life.

Stranded Cost Recovery: Ability of an electric utility to recover stranded costs through surcharges or other means, as allowed by a regulatory authority.

Stranded Costs: Costs incurred in the past that have been rendered non-economic and or "stranded" due to the onset of competition or by other changing economic or business conditions.

Stranded Investment: Net plant investment held by owners of a facility at the time when de-regulation (restructuring) takes place. “Stranded” implies an inability to recover the investment over the original amortization period.

Strategy and Planning: Strategy identifies future business direction and goals. The key objectives in strategy are to employ company strengths to take advantage of business opportunities, while avoiding business threats created by company weaknesses. Planning refers to business planning or the objectives and methods anticipated to implement the strategies.

Sunk Costs: Costs incurred (i.e., funds spent or committed) in the past that cannot be affected by any present or future course of action.

Unbundling of Rates: A pricing structure which charges separately for the individual components of providing various utility services.

Used and Useful Rule: A utility is entitled to earn a return on all its property used and useful in the provision of utility service.

Useful Life: The period of time over which property is depreciated; the length of time that property or equipment is expected to last before replacement.

Valuation: A process by which the value of an asset or resource is assessed.

Variable Costs: Those expenses of a business enterprise which vary with changes in volume of output, such as outlays for fuel to generate electric power.

Variable O&M Costs: O&M cost categories that depend at least partially on the amount of energy generated by the plant, excluding fixed costs that are incurred regardless of whether the resource is operating.

Weighted Average Cost of Capital: A weighted average of the component costs of debt, preferred stock, and common equity. Also called the “composite cost of capital.”

Working Capital: The amount of cash or other liquid assets that a company must have on hand to meet the current costs of operations until such a time as it is reimbursed by its customers. Sometimes it is used in the narrow sense to mean the difference between current and accrued assets and current and accrued liabilities.

REFERENCES

[1] PUBLIC UTILITY FINANCE & ACCOUNTING: A READER, Financial Accounting Institute, 1984.
[2] VALUATION: MEASURING AND MANAGING THE VALUE OF COMPANIES, Copeland, Thomas E., John Wiley & Sons, 1990.
[3] NATIONAL ENERGY POLICY, National Energy Policy Development Group, U.S. Government Printing Office, 2001.
[4] FUNDAMENTAL ACCOUNTING PRINCIPLES, Pyle, William W. and Larson, Kermit D., Richard D. Irwin, Inc., 1981.
[5] PUBLIC UTILITY ACCOUNTING: THEORY AND APPLICATION, Suelflow, James E., MSU Public Utility Studies, 1973.
[6] FINANCING BUSINESS FIRMS, Wert, James E. and Prather, Richard D. Erwin, Inc., 1975.
[7] REVIEW OF SELECTED COST DRIVERS FOR DECISIONS ON CONTINUED OPERATION OF OLDER NUCLEAR REACTORS: Safety Upgrades, Lifetime Extension, Decommissioning, IAEA-TECDOC-1084, Vienna (1999)
[8] FUNCTIONAL PERFORMANCE INDICATORS, Southeastern Electric Exchange, Corporate Performance Section, 1990.
[9] NUCLEAR POWER FINANCIAL INDICATORS FOR A COMPETITIVE MARKET, Bonnie, P., Baughman, J. Jr., Cornett, K., EPRI, 2001.
[10] EUCG NUCLEAR COMMITTEE, NUCLEAR DATABASE INSTRUCTIONS, EUCG Nuclear Committee, January 2001.

OVERVIEW OF THE ROLE OF ECONOMICS IN PLANT LIFE MANGEMENT – LICENSE RENEWAL IN THE U.S.

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Abstract.

License renewal of operating nuclear power plants in the United States has become one of the most successful U.S. nuclear regulatory activities in the past few years. In 1995, the U.S. Nuclear Regulatory Commission (NRC) published a revised rule in 10 CFR Part 54 that provided the requirements for an operating nuclear plant to seek license renewal. Although the rule generated considerable interest, the decision to seek license renewal was and is fundamentally an economic decision. In 1995, many people believed that only a select few operating nuclear plants would pursue license renewal and that most plants would operate for no more than 40 years. As of 2002, the owners of approximately 52% of the U.S. nuclear fleet of 103 operating nuclear plants, have decided to pursue license renewal and more are expected to follow. This change in direction since 1995 can be attributed to the improving economics of U.S. nuclear power plant operation and to the improved regulatory process resulting from the 1995 revision to 10 CFR Part 54.

1. BACKGROUND

In December 1991, the U.S. Nuclear Regulatory Commission (NRC) issued a rule in 10 CFR Part 54 for license renewal for operating nuclear power plants. This rule allowed nuclear plant owners to extend the original 40-year term of their license for additional terms of 20 years each. It required an applicant to identify "aging mechanisms unique to license renewal" that could affect the safety performance of plant components and structures. During follow-up pilot studies by the nuclear industry to support prospective license renewal applications, it became clear the new rule was more complicated and expensive to implement than necessary. As a result of this finding, the NRC agreed to revise the rule.

In 1995, the NRC published a revised rule in 10 CFR Part 54 that provides the requirements for an operating nuclear plant to seek license renewal. Based on the original rule issued in 1991, the estimated cost just to prepare a license renewal application was about $40 million (U.S.). Under the revised rule, the cost to prepare an application was reduced to between $10 and $15 million.

Although the revised rule generated considerable interest, the decision to seek license renewal was and is fundamentally an economic decision. In 1995, many people believed that only a select few operating nuclear plants would pursue license renewal and that most would operate for no more than 40 years. The primary reason for this belief was that the cost of keeping U.S. nuclear plants running did not appear to be competitive with other forms of electricity generation.

By 1998, the economic conditions in the U.S. were changing dramatically. Electric utility deregulation was moving ahead, the need for electricity was growing, and the operating costs for nuclear power plants were declining. Also in 1998, the first two applications for license renewal were submitted to the NRC by Baltimore Gas & Electric for the two-unit Calvert Cliffs nuclear power plant and by Duke Energy for the three-unit Oconee nuclear power plant. The U.S. nuclear industry was somewhat skeptical that the NRC could complete the license renewal process in a timely and predictable manner. This skepticism was due to the protracted and unpredictable process used by the NRC to approve the original operating licenses, especially in the 1980's and 1990's.

In March 2000, the NRC approved the renewal of the 40-year operating licenses for the two-unit Calvert Cliffs nuclear power plant for an additional 20 years. Two months later, the NRC approved the renewal of the operating licenses for the three-unit Oconee nuclear station. These reviews were both completed by the NRC in a timely, predictable, and stable manner.

As of September 2002, the NRC has approved renewal of the operating licenses for 10 nuclear units (capable of producing 8,000 megawatts of electricity) and has applications under review for 16 more units. Several additional nuclear plant operating companies have notified the NRC of their intention to seek license renewal for 28 nuclear units by 2005. This means that the owners of 54 nuclear units, or approximately 52% of the U.S. nuclear fleet of 103 operating nuclear plants, have decided to pursue license renewal and more are expected to follow. This change in direction since 1995 can be attributed to the improved economics of U.S. nuclear power plant operation and to the improved regulatory process resulting from the 1995 revision to 10 CFR Part 54.

2. REGULATORY AND SOCIO-ECONOMIC FACTORS

Without license renewal in U.S., thirty-eight commercial nuclear power plants in the U.S. will reach the end of their initial operating licenses within the next 15 years. The companies that own those plants must decide soon whether to pursue license renewal, or plan to retire those plants and replace them with other generating capacity or electricity purchased on the open market.

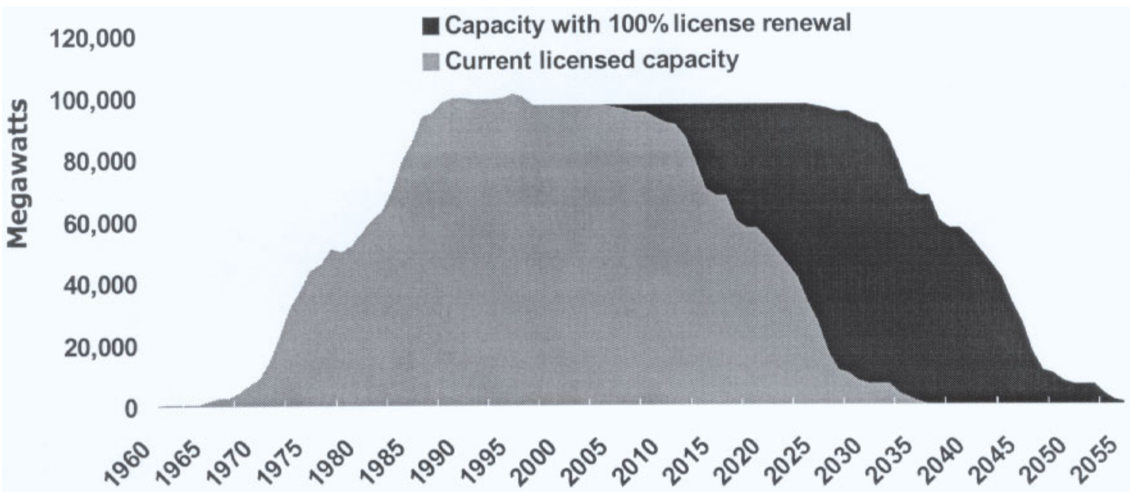
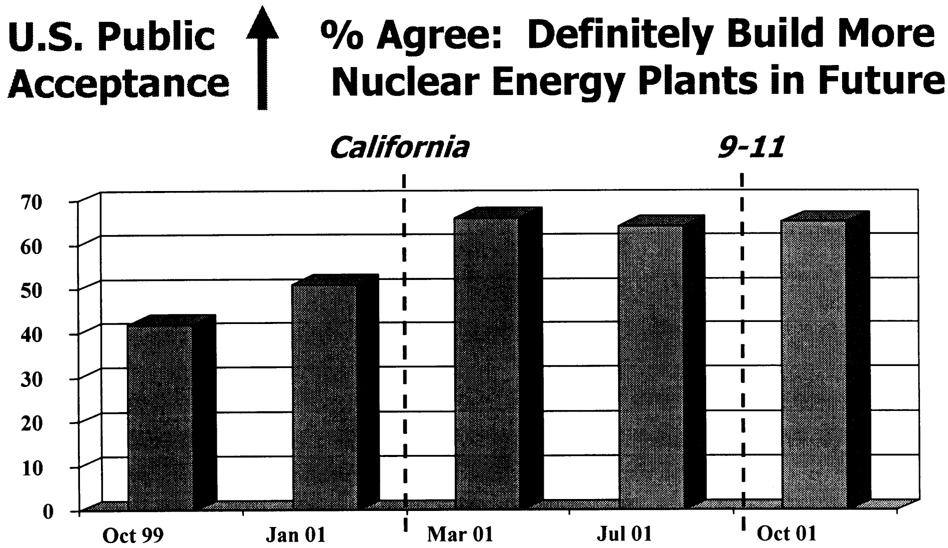


Figure 3. U.S. Nuclear Energy Capacity, With and Without License Renewal.

As shown in Figure 1, the loss of nuclear generated electricity would begin a dramatic trend starting approximately in 2010. However, with an assumption of 100 percent of operating nuclear plants obtaining license renewal, the loss of nuclear generated electricity would be delayed until approximately 2030 [1]. Based on the current regulatory and socio-economic factors, the assumption of near 100 percent license renewal is not unrealistic.

In 1995, when the NRC published the revised license renewal rule, the regulatory environment for license renewal was unpredictable and many nuclear industry representatives were somewhat skeptical that the NRC could complete the license renewal process in a timely and predictable manner. However, based on the success of the Calvert Cliffs and the Oconee license renewals in 2000, the regulatory process has proven to be both stable and predictable. The NRC schedules for approving license renewal applications have been consistently less than 24 months for the 10 completed license renewals and all indications are that the NRC review schedule will continue to be shortened to about 22 months or less due to efficiency gains in the regulatory review process.

In addition, the cost of preparing a license renewal application has seen dramatic improvement. Under the old license renewal rule published in 1991, a company would have had to spend an estimated $40 million just to prepare an application. Today's estimate is roughly one-quarter of that, and some companies believe they can cut the cost further by working cooperatively on elements that are common to plants of similar design.



Bisconti Research Inc.

Figure 2. Change in Public Acceptance of Nuclear Power Plants in the U.S.

Separate from the NRC regulatory environment, the electric utility deregulation process in the U.S. is also creating a potentially more positive environment for license renewal. A deregulated, competitive electric generating business creates a powerful business incentive to renew a nuclear plant's license. Under cost-of-service regulation, a power company's earnings are based on its rate base – its total investment in plant and capital equipment. Because a 40-year-old nuclear unit would be fully depreciated – and thus not part of the rate base – it would have limited earnings potential under cost-of-service regulation. In a deregulated, competitive business, a fully depreciated nuclear plant may be a tremendous asset. It can sell its power at marginal cost, which (for a nuclear unit) is currently very competitive. Such a plant would have a significant profit potential.

The socio-economic situation in the U.S. has also changed in the past few years such that public opinion is much more positive about nuclear power plants. As shown in Figure 2, the public support for building new nuclear power plants has significantly increased since 1999, and has continued to be above 60 percent of the population. According to U.S. Secretary of Energy, Spencer Abraham, in a speech dated February 2002, ". . . lingering public concerns about the safety of nuclear power plants is an ongoing challenge for the nuclear industry. I think the industry has accomplished the most important work to assure the public – it has operated its plants safely, efficiently, and professionally and earned the trust and respect of an increasing proportion of the public. Recent polls indicate that 65% of the public believes in the use of nuclear power [2]."

According to a February 2002 U.S. national survey, 60 percent of Americans think nuclear plants are safe. The poll also found that 66 percent of Americans favor the use of nuclear energy as one of the primary ways to provide electricity – the highest level in any survey asking this question since 1983 [3].

The current regulatory and socio-economic factors are favorable for license renewal of nuclear power plants in the U.S. These positive factors are related to safety, capacity, and environmental factors that also create favorable economic conditions for license renewal.

3. SAFETY AND CAPACITY FACTORS

Safety remains the top priority for operating nuclear power plants and the safety record of U.S. plants has continued to improve in conjunction with the economic improvements. It is now widely agreed that the top performing plants from a safety viewpoint are also the top performing plants from an economic viewpoint. This close relationship between safety and economics is likely the reason that the positive public perception of nuclear power plant operation in the U.S., and license renewal of nuclear power plants, is seeing an improving trend.

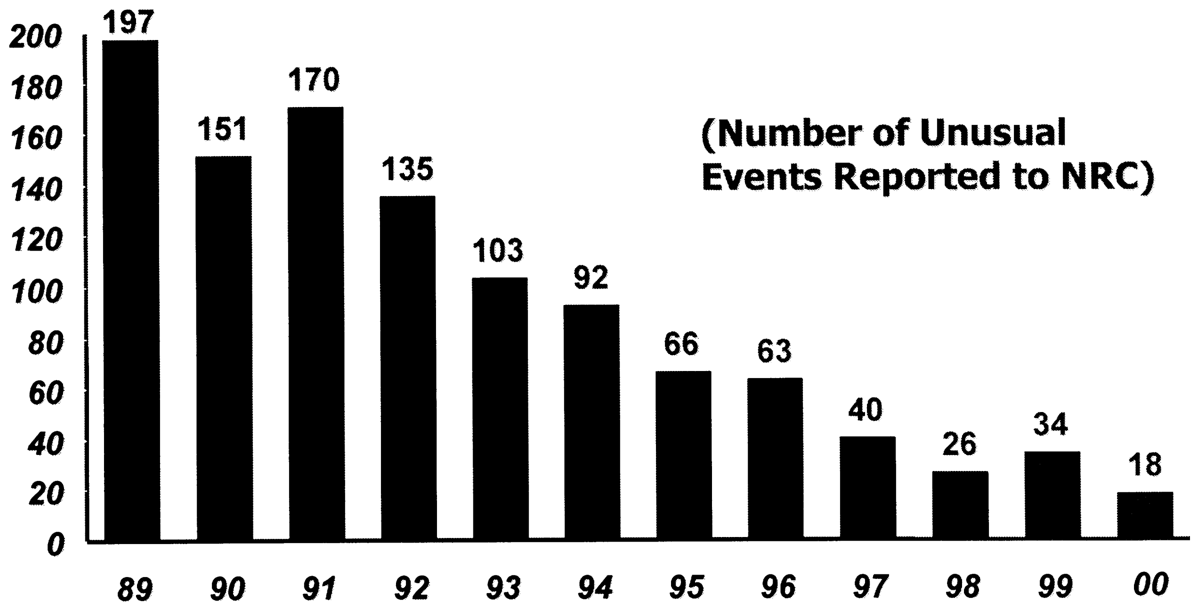


Figure 3. Number of Unusual Events Report to the NRC, 1989-2000.

World Association of Nuclear Operators (WANO) data shows an improving trend in safety records for U.S. nuclear plants. Since 1989, the U.S. industry has monitored the performance of three important, redundant standby safety systems used to respond to unusual situations. The latest figures show that key safety systems are highly capable of performing their safety functions if called upon [4].

NRC data on numbers of safety events also shows an improving trend since 1989. Figure 3 is a graph of the number of safety events reported to the NRC from 1989 through 2000 [1]. This steady trend of improvement corresponds with improving public acceptance of nuclear power.

Safety is also measured in terms of the industrial safety accident rate. That rate has plummeted from 2.1 lost-time accidents per 200,000 worker-hours in 1980 to 0.24 in 2001. By comparison, the accident rate for the U.S. manufacturing sector was 4 per 200,000 worker-hours in 2000 – the last year for which figures are available from the Bureau of Labor Statistics. The industrial safety rate is a "useful measure of the culture of nuclear safety and the attitude of plant management and staff to safe operation across the board," according to Ralph Beedle, chief nuclear officer and senior vice president of the Nuclear Energy Institute. "Not only is it safer to work in a nuclear power plant than in general manufacturing, but it's safer working in a nuclear power plant than in finance, insurance, and real estate." Figure 4 provides a graphical view of industrial safety accident rate from 1992 to 2000 [4].

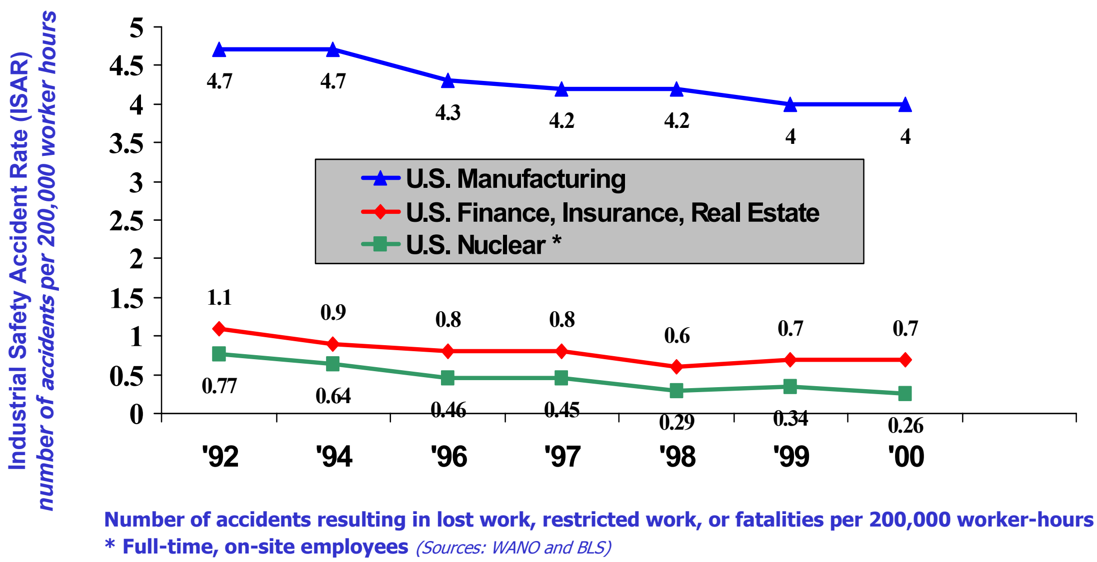


Figure 4. U.S. Industrial Safety Accident Rate (ISAR), 1992 – 2000.

As stated earlier, improving safety performance has been paralleled by improving economic performance of U.S. nuclear power plants. Economics will continue to drive the decision to pursue license renewal for the U.S. nuclear power plants. For nuclear units with strong safety and performance records, license renewal is a good business decision when compared to the cost of building new generating capacity. Building a new gas-fired combined-cycle plant is estimated to cost approximately $500-600 per kilowatt. For a coal-fired power plant with state-of-the-art emission control systems, the cost would be approximately $1,200 per kilowatt. The cost of license renewal (approximately $10 to $15 million) is extremely attractive to strong performing nuclear companies.

One of the major factors in strong performance is capacity factor. In 1990, the average capacity factor for U.S. nuclear plants was about 70 percent. By the end of 2000, the average capacity factor had increased to about 90 percent. As shown in Figure 5, the increase in capacity factor from 1990 to 2000 was equivalent to building 22 one thousand megawatt power plants and provided the means for meeting 22 percent of growth in U.S. electricity demand during the 1990s. In 2001, that figure continued to increase to an average capacity factor of 90.7 percent (see Figure 6).

The two largest nuclear utilities in the U.S., Entergy and Exelon, have done even better in capacity factor improvement. In 2001, Entergy's nuclear fleet of nine units had an average capacity factor of 96 percent and Exelon's nuclear fleet of 17 units had an average capacity factor of 94.4 percent [5]. These performance records are translating into improved economic status for nuclear plants, which provides further support for making a decision to seek license renewal.

Due in part to the capacity factor improvements, the production cost at U.S. nuclear power plants are now the lowest of any major expandable electricity source according to Resource Data International. In 2000, the latest year for which data are available, estimated production costs at nuclear plants averaged 1.76 cents per kilowatt-hour (see Figure 7). In 1999, nuclear plant production costs fell below those of coal plants for the first time in more than a decade [6].

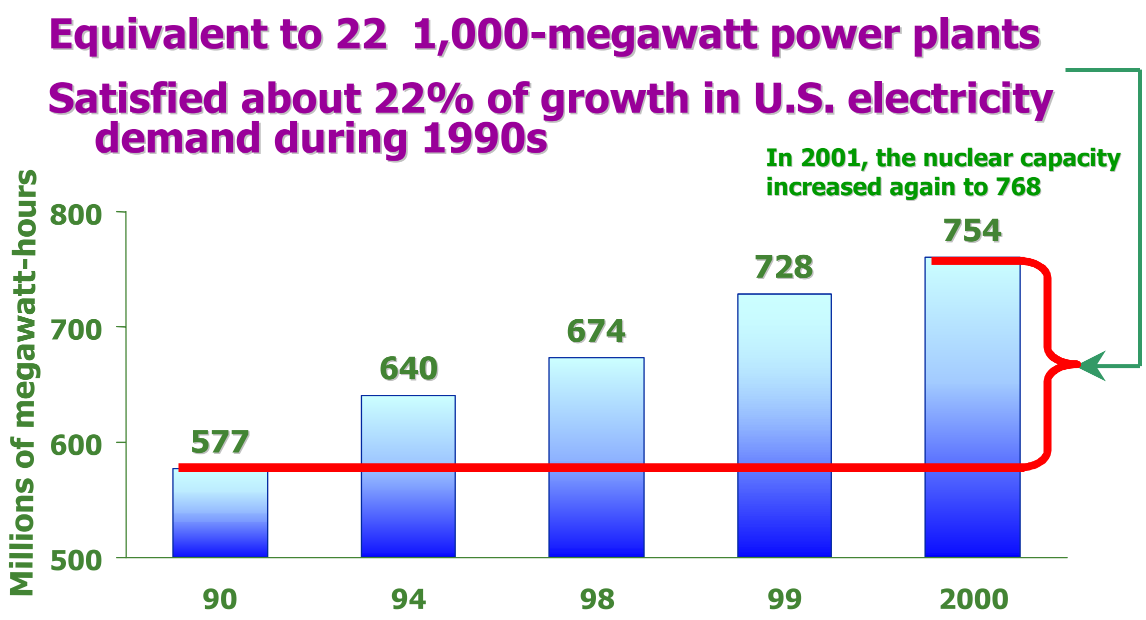


Figure 5. Total Output of U.S. Nuclear Plants, 1990 – 2000.

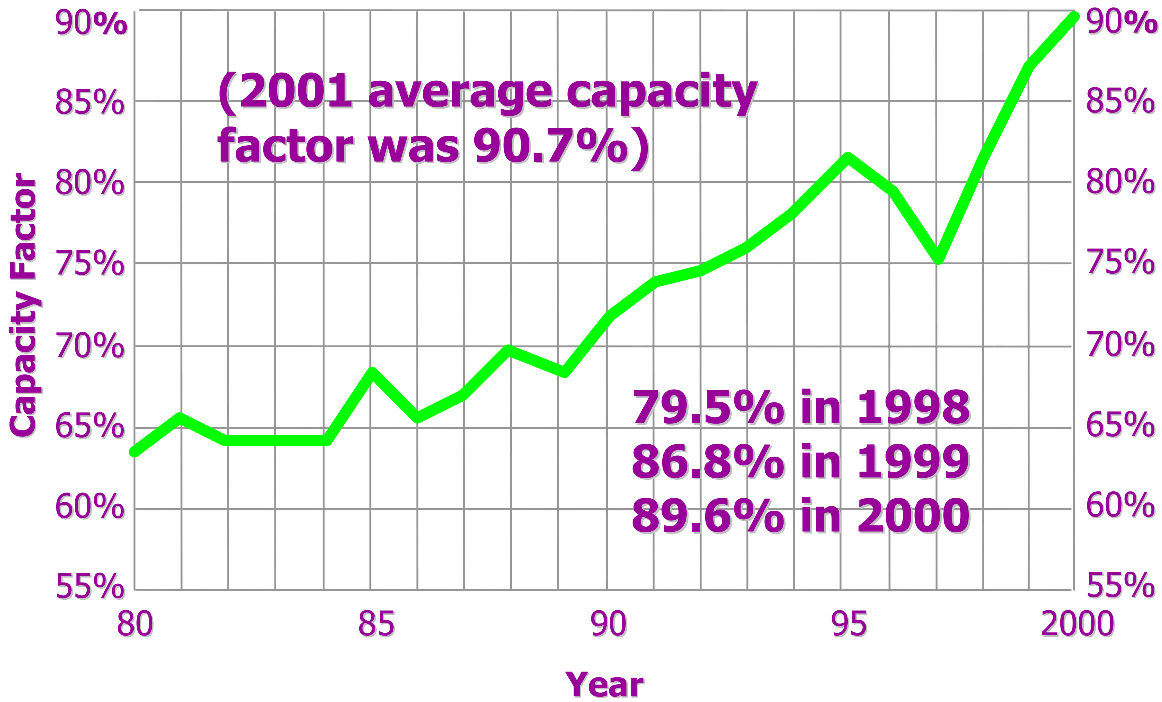


Figure 6. U.S. Nuclear Power Plant Capacity Factors, 1980 – 2000.

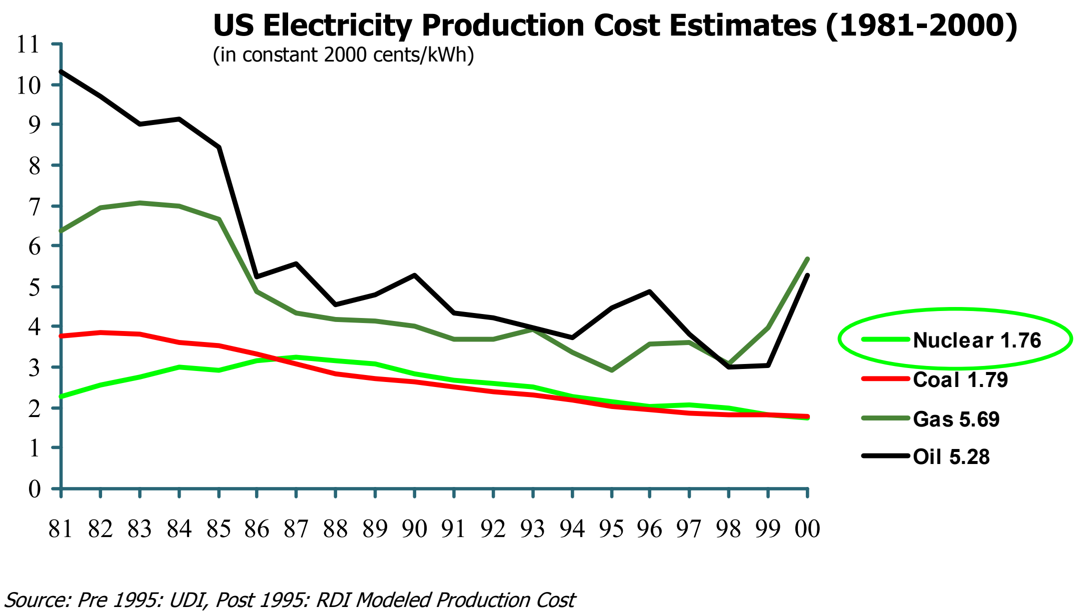


Figure 7. U.S. Electricity Production Cost Estimates, 1981 –

In summary, the current safety and capacity factors are favorable for license renewal of nuclear power plants in the U.S. These positive factors are related to positive regulatory, socio-economic, and environmental factors that also create favorable economic conditions for license renewal.

4. ENVIRONMENTAL FACTORS

Electricity produced by nuclear power plants is a major contributor to clear air in the U.S. Nuclear power plants in the eastern U.S. make it possible for many states to meet the requirements of the Clean Air Act. Since the mid-1970's, nuclear energy enabled the U.S. to avoid emitting over 80 million tons of sulfur dioxide and about 40 million tons of nitrogen oxides [2].

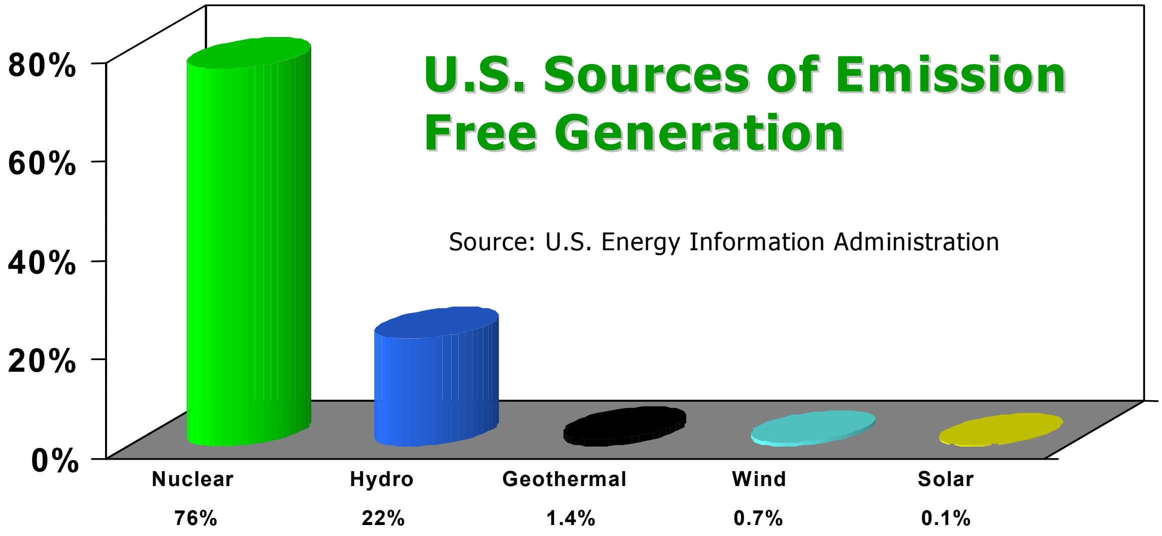


Figure 8. U.S. Sources of Emission Free Generation.

Nuclear power is currently the single largest contributor to the U.S. voluntary program to reduce carbon emissions. The capacity factor gains made in the past decade have amounted to about half of the voluntary carbon reductions achieved so far by all U.S. industries [7]. As shown in Figure 8, the latest data from the U.S. Energy Information Administration indicates that nuclear power plants constitute 76 percent of the emission free sources of electricity in the U.S., with hydo power plants a distant second at 22 percent.

In summary, the current environmental factors are favorable for license renewal of nuclear power plants in the U.S. The positive environmental factors, when combined with the positive regulatory, socio-economic, safety, and capacity factors, help to create favorable economic conditions for license renewal.

5. CONCLUSION

The ultimate decision to pursue license renewal in the U.S. is primarily an economic decision. The U.S. currently has 103 operating nuclear power plants, which produce 20 percent of U.S. electricity. The U.S. nuclear power program is the largest in the world and the best performing at about 91 percent capacity factor in 2001. In its 2001 forecast of U.S. nuclear capacity in 2020, the Energy Information Administration (EIA) assumed that few U.S. reactor licenses would be renewed. Thanks to the success of license renewal, the recent EIA forecast for U.S. nuclear capacity in 2020 is 23 percent higher than the 2001 forecast [8]. If the current positive economic trends, supported by the positive regulatory, socio-economic, safety, capacity, and environmental factors for the U.S. nuclear plants continue, license renewal for most of the 103 operating plants is a distinct possibility.

ACKNOWLEDGEMENTS

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REFERENCES

[1] "Nuclear Power Plant License Renewal," Nuclear Energy Institute Status Report, February 2002

[2] U.S. Secretary of Energy, Spencer Abraham, speech at the Global Nuclear Energy Summit, The Cosmos Club, Washington, D.C., February 14, 2002

[3] "Americans Confident About Nuclear Plant Safety, Majority Continues to Favor the Use of Nuclear Energy," Nuclear Energy Insight, February 2002

[4] "U.S. Plants Maintain Unwavering Safety Focus," Nuclear Energy Insight, March 2002

[5] "Nuclear Plants Excel in 2001," Nuclear Energy Insight, February 2002

[6] "Nuclear Energy's Low Costs," Nuclear Energy Insight, June 2002

[7] "Nuclear: The Stars are Aligning," speech at the Southern Legislative Conference by C. Randy Hutchinson, Senior Vice President, Entergy Corporation, July 17, 2001

[8] "Energy Forecast Reflects Success of License Renewal," Nuclear Energy Insight, April 2002

[9] "Nuclear Energy 2002: Solid Value…Significant Upside," Nuclear Energy Institute Annual Briefing for the Financial Community, February 1, 2002

Session 3
PLANT LIFE MANAGEMENT (PLIM)

Sub-Session 3.2

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PROSPECTS OF NUCLEAR POWER DEVELOPMENT IN BELARUS

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Abstract

The paper describes the state of power in the Republic of Belarus, different scenarios of electric energy development up to 2020, including the putting into operation of NPP in the energy system. The results of preliminary investigations on the problem of NPP creation are given, such as siting for NPP location and disposal of radioactive waste, the choice of the reactor type, a sociological monitoring of public opinion concerning NPP construction in Belarus.

1. INTRODUCTION

After the Second World War Belarus being a member of the Soviet Union was undergone sufficient changes in economy and social structure. Within the period of several ten years the Republic turned from mainly agricultural region into an industrial one with powerful energy-intensive industries. From 1960 to 1990 the energy consumption was increased by a factor of 20. Naturally it required the development of appropriate energy base.

Not possessing domestic fuel and power resources (the share of own fuel resources was about 8 %), Belarus was oriented to nuclear power. In 1983 the construction of the first phase (2000 MW) of Minsk central heating and power plant was begun, then the construction of Belarus NPP with the total capacity of 6000 MW was planned in Vitebsk region. The accident at the Chernobyl NPP stopped the Nuclear Power Program in Belarus.

2. MODERN SITUATION IN BELARUS ENERGY SECTOR

Having become an independent republic, Belarus has got the energy-intensive economy not having any considerable domestic fuel and energy resources. The Republic is forced to import 90% of fuel and 25% of electric energy, the deficit of peak electric capacity reached 40%. Last years the energy consumption is reduced because of the drop in production and now Belarus is forced to buy about 85% fuel resources, mainly in Russia. In this connection in October 1992 the Council of Ministers approved “State Energy Program in the Republic of Belarus till 2010” and adopted decision of its realization. The Power Program envisaged the development of nuclear power in the Republic. In accordance with assignment of the Council of Ministers Belarus Academy of Sciences in cooperation with other Institutions had prepared the draft of the State Program of Nuclear Power Development. The researches having been carried out showed what under the conditions of severe limitations on delivery of organic fuel to the Republic the scenarios combining steam-gas and nuclear plants are the more perspective, economical and ecologically pure directions of the development.

The necessity of the works on the feasibility studies of NPP construction in Belarus was confirmed in "Main Directions of Power Policy of RB for period to 2010" and approved by the Council of Ministers in 1996 and "National strategy of sustainable development of RB"(1997).

Last years the growth of economic indicators is observed in Belarus and the proper energy base must reinforce this. The steady power is necessary for the development of the national economy, which is not subjected to the influence of the fuel market policy. The level of the dependence of the Republic on fuel and electric energy import has to be decreased.

At present, natural gas (≈85%) is prevailing in the structure of fuel-energy resources consumption. Russia supplies gas to Belarus by the price of USD 30 per1000 m^3 what is lower than European prices. However, and under these conditions the share of the payment for primary energy resources reaches 60% or about USD 2 bill. in the volume of Belarus import. These are comparable with the value of an annual state budget. Prices for gas will be sufficiently raised in Russia for the nearest future. So, according to the assessments of Russian specialists a wholesale price for natural gas will be raised by a factor of 3.7 in Russia by 2005 in comparison with 2001, which makes Belarus production ultimately non-competitive in the world market.

Electric energy plays a special role in the energy supply system in the Republic. The forecasted energy consumption will be about 45 bill. kW·h in Belarus by 2015. At present the demands of the Republic in electric energy are satisfied by 70-80% due to generation at own power electric plants operating mainly on import fuel, and by 20-30% due to electric energy import generated at Smolensk (Russia) and Ignalina (Lithuania) NPPs. But the share of electric energy import from Russia will be decreased by 2015 in connection with the forecasted growth of its internal demands and will not exceed 5 bill. kW·h a year. Ignalina NPP will be shut down by that time. Therefore, the main part of demands in electric energy (35-40 bill. kW·h) must be covered at the expense of own generation. But already now about 60% of energy equipment has worn out, its working capacity is maintained due to repairs, volumes of which increase every year. Approximately 3.8 mill. kW from 7.8 mill. kW of the installed capacities will be in operating state in Belarus by 2015. With due account of the growth of electric energy demands about 6 mill. kW of new capacities are needed to be put into operation. Thus, the problem on energy supply is one of the most important problems for Belarus. As well as for many countries of the Central- and South Eastern Europe having the limited energy resources, one of the possible ways for solving the power problem for Belarus is nuclear power.

3. NUCLEAR POWER OPTION: METHODS AND RESULTS

By present according to the Decision of the Government of Belarus the preliminary investigations on studying the possibility and availability of including the energy sources on nuclear fuel in the electric energy structure of the Republic have been carried out. The following directions have implemented these investigations:

- feasibility study of various scenarios of power development in Belarus up to 2020;
- selection of sites for NPP construction;
- selection of a reactor type for NPP with the increased safety;
- assessment of ways of RAW storage and management;
- monitoring of the public opinion about the nuclear power development in Belarus.

The analysis of possible ways of electric energy development in Belarus has been carried out with the help of the WASP-III plus (Wien Automatic System Planning Package) program more widely applied for energy planning and the database RTDB (Reference Technology Database), and CSDB (Country Specific Database), developed in the framework of IAEA DECADES project.

When selecting scenarios of generating sources system development, technologies of electric energy generation and fuel types available at present have been taken into consideration first of all. The following has been included in the number of the technologies considered:

- steam turbines;
- combined cycles;
- gas turbines;
- solar energy converters;
- wind plants;
- nuclear power plants (NPP).

Taking into account the fact what the putting into operation of units on coal in the energy system results in the greatest expenditures of electric energy generation, as well as the fact the capital specific expenditures of wind plants and solar converters sufficiently exceed expedient limits from the economic point of view, traditional technologies of electric energy generation and fuel types have been considered in the scenarios of energy generating sources system development (Table 1).

TableI. Scenarios of development of generating sources

Scenario 1	Steam turbines (ST)	Natural gas
	Steam-gas plants (SGP)	Natural gas
	Gas turbines (GT)	Natural gas
Scenario 2	Steam turbines (ST)	Natural gas
	Steam turbines (ST)	Nuclear fuel
	Steam-gas plants (SGP)	Natural gas
	Gas turbines (GT)	Natural gas

The power unit with the capacity of 640 MW has been considered as the unit-candidate on nuclear fuel for replacement of the obsolete equipment, here the more pessimistic technical and economic characteristics for NPP have been chosen.

More than 10000 different variants of putting into operation of units have been calculated for each scenario, and such variants have been chosen which ensure the minimum total cost of electric energy generation in the system of generating sources. The results of the calculations have shown that Scenario 2 with using natural gas and nuclear fuel is the more optimum. The optimum is NPP with the capacity of 2500 MW (4 units of 640 MW each) being put into operation in the energy system in Belarus from 2010 to 2020, as well as new steam-gas and gas-turbine plants with the capacity of approximately 5300 MW. Average annual expenses for construction of all 4 units are about 250 mill. USD. Realization of Scenario 2 will give the possibility to decrease the cost price of electric energy generation by ≈20% and to save ≈500 mill. USD a year in the import of natural gas.

Selection of points and sites for NPP construction is a complicated task, which needs detailed researches and site engineering of different profile. Within the period of 1969-1985 various institutions of the former USSR carried out feasibility studies on possible sites for construction of the Belarus NPP on the territory of the Republic of Belarus. In 1992-1994 the works on studying the possibility of NPP siting were recommenced. On the basis of the information accumulated on population of the large towns, forbidden territories, airways, oil and gas pipelines and other factors the map of the refused territories has been worked out which includes about 50% of the territory of the Republic of Belarus. Six relatively competitive sites have been chosen out from 54 possible points for NPP siting, such as:

- in Vitebsk region - Dubrovensky and Shumilinsky;
- in Mogilev region - and Bykhovsky;
- in Gomel region - Rogachevsky;
- in Grodno region - Skidelsky.

Because of the lack of funds given in 1995-1996 the further site engineering was carried out on three competitive sites, the more favourable one from six mentioned above, such as Dubrovensky, Shklovsko-Goretsky and Bykhovsky. In parallel the works on assessment of NPP effect on environment in these sites were carried out. The carried out complex of works on selection of sites for NPP construction has shown that there is the possibility of nuclear power plants siting on the territory of the republic.

The analysis of nuclear power development in the world shows new reactors plants of the increased safety and economy have been designed. Both the most advanced achievements of the science and engineering are realized in them, and also traditional technical decisions demonstrated their value in the process of long-duration operation. The level of safety of power units corresponds to the international demands and recommendations of the IAEA. Such projects are being created in Russia, USA, France, Germany, Japan and other countries. The following projects are the better designed and recognized in the world nuclear power:

- PWR of American firm WESTINGHOUSE;
- PWR N4 of French firm FRAMATOME;
- PWR KONVOI of German firm SIEMENS.

Up to now leading Russian design institutions have prepared a number of new projects of reactor plants and power units of NPP of a new generation with the increased safety of large and middle capacities: NPP-91, NPP-92 (NG WWER-1000), NPP with NG WWER - 640 reactor.

As a result of comparison of foreign and Russian projects of NPP as to feasibility characteristics, safety and reliability as well as to the degree of completion for construction Russian projects of NPP-92 and NG WWER - 640 are more preferable. A final choice of the project is advisable to be done as to the results of the international tender.

The project of the Concept of the radioactive waste management has been developed, sites for storing low and medium active waste have been determined, as well as regions for high active waste disposal. The results of preliminary study have shown the geological and hydro geological peculiarities of the territory of the Republic of Belarus permit the location of depositories of various types for radioactive waste. In particular, sites for placement of depositories for low and medium active short-lived waste of partially put deeper into ground can be selected in the region of the supposed construction of NPP.

At present Belarus possesses a definite production, engineering and scientific potential, which can be used for the realization of the Nuclear Power Development Program. Construction organizations have the experience on construction of large power units. Therefore, a considerable share of construction works can be done by own efforts. Electric power and radio-electronics industries can be involved in production of instrumentation, control and monitoring systems, sets and chips to them. The very skilled staff of designers is in BelNIPIEnergoprom, BelTEI, Energosetproject. Scientific staff from the National Academy of Sciences and branch Science and Research Institutes is able to carry out works on scientific support for construction and operation of NPP.

One of the key problems associated with the development of nuclear power in is the overcoming the so-called "Chernobyl syndrome" in the public acceptance. As it is known, Belarus has been suffered from Chernobyl accident most of other countries. The portion of Belarus is about 58% all territory having been contaminated. There are 27 towns and 2736 settlements with more than 2 million inhabitants on the contaminated territory. Naturally, the attitude of considerable part of the population of Belarus to the possibility of NPP construction has been formed in the light of the accident. Within the period of 1995-1998 three sociological monitoring of public opinion were conducted regarding of the prospects of the energy development and, in particular, nuclear power in Belarus. The results of last sociological monitoring, where besides population poll the poll of experts (specialists of a high scientific qualification) was done, as well as mass media professionals, are presented in Fig. 1. As it is in Fig. 1 such categories of the population as mass media professionals and the experts have opposite opinion on the issue of possible construction of NPP. One third of the population of Belarus is unalterable oppose to construction of NPP in the territory of Belarus, while the rest ones are potentially ready to support it at observing a number of conditions, associated with ensuring the safety of population and accident-free operation of NPP.

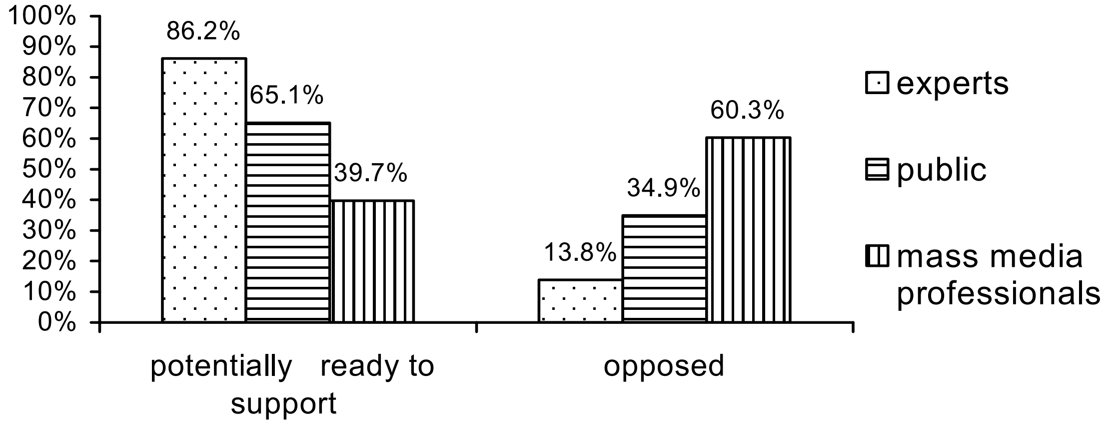

FIG.1. Results of sociological monitoring of public opinion on NPP construction in Belarus.

4. CONCLUSION

The results of the implemented works are the following:

- economically optimum scenario of electric energy development in Belarus is the variant with introduction of power sources on natural gas and nuclear fuel in the energy system of Belarus;
- at introduction of NPP with the capacity of 2.5 mill. kW in the energy system of Belarus the deficit of electric capacity of the energy system of Belarus will be practically eliminated to 2020, and an annual economy in the import of natural gas will be 500 mill. USD;
- suitable sites have been selected in the territory of Belarus, which answer the modern requirements imposing for NPP siting, though the required volume of geologic-seismic and other kinds of investigations has not been completed yet. NPP with the capacity of up to 4000 Mw can be located in each of these sites;
- all considered advanced reactors of PWR type (Westinghouse, Framatome and Siemens firms) and Russian reactors NG WWER-640, with the advanced safety systems are similar as to safety performances. Concerning economic parameters the Russian reactors are more preferable. But the final choice of a concrete NPP project must be determined by the international tender with due account of engineering and commercial conditions and possible patterns of financing;
- geological and hydro geological peculiarities of Belarus territory admit the construction of storage facilities of various types for radioactive waste;
- almost 35% of the population of Belarus and more than 60% of the mass-media professionals are unalterable oppose to NPP construction in Belarus.

The researches having been carried out do not reveal any objective obstacles in NPP construction in the Republic of Belarus. But one should note the “Chernobyl syndrome” still plays an important role in the problem of nuclear power development in the republic. During the next coming years the set of works are to be completed:

- Selection of the main and the reserve sites for NPP;
- The type of NPP is to be finally defined;
- Conceptual design of storage facilities and points for long duration storing and disposal of RAW;
- Analysis of the possible sources of funding;
- Evaluation of the investments requirements;
- Scientific and research works on safe development of nuclear power
- Public information on the problems connected with the power policy of Belarus.

POWER UPRATING OF NUCLEAR POWER PLANTS – SOLUTIONS, EXPERIENCE AND ECONOMICAL ASPECTS

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Abstract

Due to opening of market for electricity the prices for electricity decrease very rapid. Therefore the costs for power generation must decrease also. One efficient way to reach this goal is to increase electric power output of operating plants by minimal commercial efforts. The following paper gives a short overview for power increase realized in Framatome ANP nuclear power plants and nuclear power plants of different manufacturers also realized by Framatome ANP.

1. INTRODUCTION

The possibilities of power increase of NPP's become more and more interesting related to the question, how to operate a plant more efficient and profitable, what is essential in the times of increased competition on the international energy market. Another important aspect is the balance of available power capacity in a country or in an energy supply network of some countries. If for example plants are foreseen for closing or the completion of plants still under construction is not yet finished, but the power demand is increasing, then power up rating, also called power increase, of the already existing power plants is an achievable solution.

On this background the power increase can help to guarantee a stable electrical power supply for industry and private consumers.

2. GOALS OF POWER UPRATE

Which results will be reached by power increase?

Increase of plant economy due to optimal cost-benefit relation for the rest of plant lifetime.

- Optimal use of available reactor core power by increase of efficiency of the secondary side systems and reduction of the house load consumption (self needs)

- Use of available design margins of the nuclear island, the secondary heat sink (turbine, preheaters etc.) and cooling water systems

- Increase of core power considering plant design margins and/or possible improvements of design features

- Improvement of the heat transfer from primary to secondary side (Steam generator improvement modification or exchange)

3. POSSIBILITIES OF POWER UPRATE

The main components and systems to be assessed in frame of power up rating activities in PWR plant are summarized in figure1.

The possibilities of electrical power output increase can be divided in two groups of measures:

1. Increase of core output (acc. German practice nuclear licensing necessary):

    - Increase of ΔT (core inlet to core outlet temperature) with unchanged constant average coolant temperature, using core design margins
    - Increase of average coolant temperature using core design margins
    - Increase of core power beyond design margins, new core design and licensing would be necessary

2. “Green Megawatt” from conventional island (acc. German practice nuclear licensing not necessary):

    - Improvement/Modification of turbine systems
    - Check and improvement of heat balance (thermal efficiency), for instance implementation of POWERSEP, see figure 2
    - Increase effectiveness of steam generators
    - Reduce self consumption

Both groups of measures for power up rate, increase of core output as well as "green Megawatt", require the following necessary preconditions in the frame of I&C:

- sufficient preciseness of measurements (usually present in Framatome ANP plants)
- sufficient measurement ranges
- sufficient possibilities for adaptation of limit values in Reactor Protection System, limitation systems and control systems

Depending on the individual conditions of each plant a tailor made optimized concept according to the customers needs for power increase can be developed. For taking decision about power output increase the following aspects have to be assessed:

- max. possible gain by optimization/modification of the conventional island
- max. possible gain by core power increase (new core configuration) without decreasing rest life time by brittle fracture risk and in view of possibility for life time extension

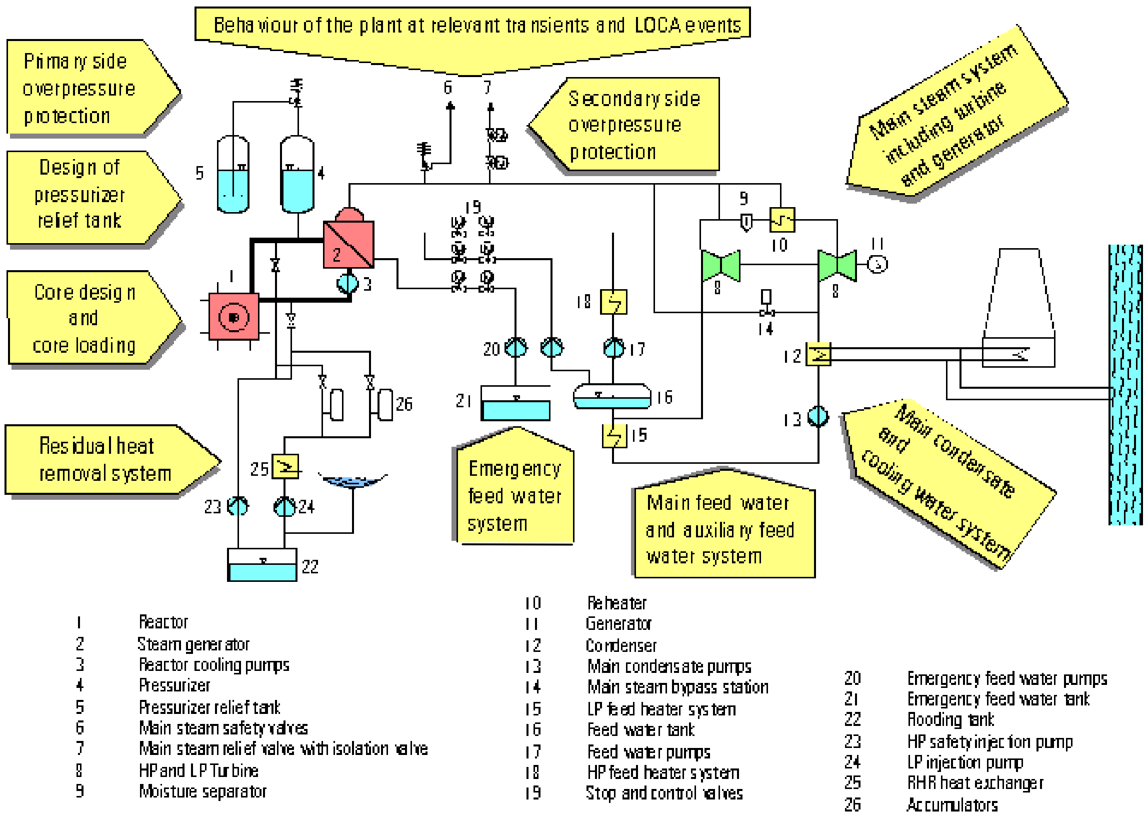

FIG.1. Main components and systems to be assessed for power up rate in PWR plants.

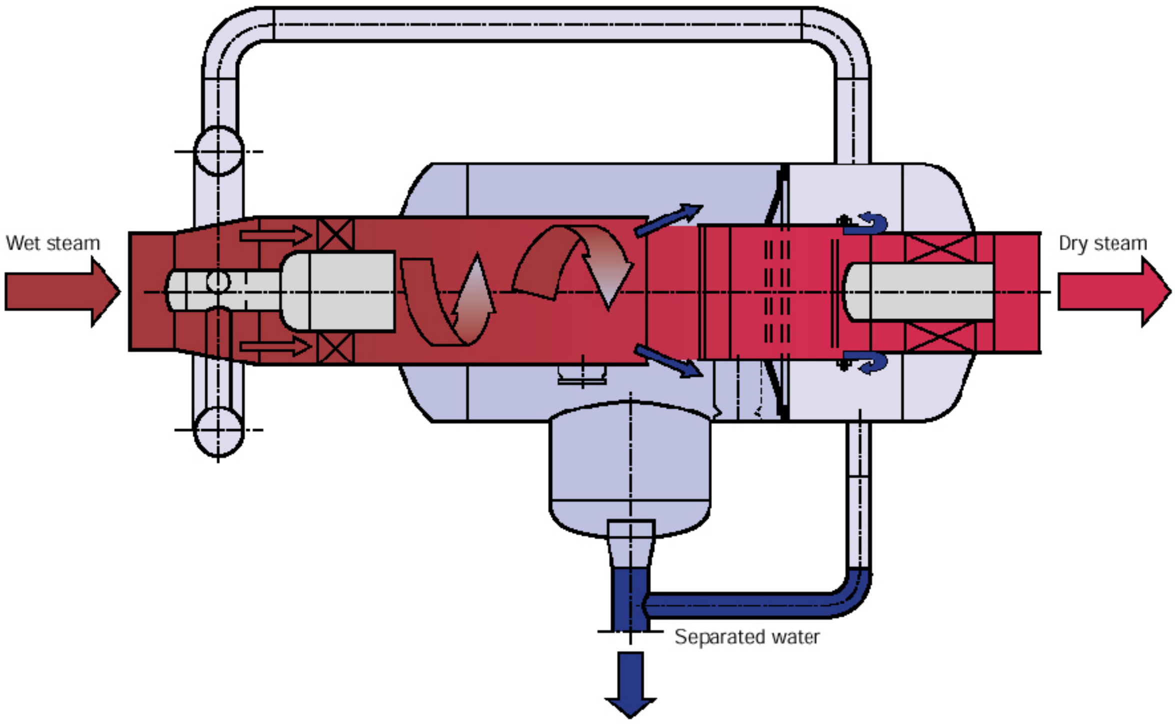

FIG.2. Principle working scheme of the POWERSEP [1].

with respect to the following criteria:

- time for planning, engineering, licensing, fabrication, commissioning activities
- costs for planning, engineering, licensing, fabrication, commissioning activities
- possible schemes for completion (turnkey, consortium, single orders with coordination by the utility etc.)
- amortization time (with respect to rest life time and energy prices).

4. CONCLUSION

It is possible to gain the power increase by core measures only or by "green megawatt" only, but in most of the cases the combination of both leads to maximum electrical power output. However, a carefully check of economics is necessary. By this method the combination of measures realizing the best benefit at the current energy market conditions will become apparent.

REFERENCES

[1] Hans-Joachim Frank, Framatome ANP GmbH; Wilhelm Bruckmann, Balcke-Duerr Service GmbH, Advanced Nuclear Power Journal, January 2002, "POWERSEP pre-separator on secondary side increases plant output"

[2] Susanne Wisniewski, Framatome ANP GmbH; NUSIM March 2002, “Possibilities for increasing electrical power output in PWR nuclear power plants”

BN-350 NPP REGULATORY ASPECTS OF DECOMMISSIONING

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Abstract

The fast breeder reactor BN-350 was commissioned in 1973. In 1999 the Government of the Republic of Kazakhstan adopted Decree on the Decommissioning of Reactor. Since the decision on the decommissioning was accepted before end of scheduled service life (2003), to this moment "The Decommissioning Plan" was not worked out. As the shut downed reactor continues to remain a source of nuclear and radiation hazard, one have to take measures on putting the reactor to safe status, and thus "Plan of priority measures on BN-350 reactor decommissioning" was developed. It includes following activities:

- Measures on BN-350 decommissioning Project development;
- Measures on provision the reactor safety during transition period;
- Measures on spent fuel disposal for a long-term storage;
- Measures on sodium drainage and utilization.

This paper describes the current situation in Kazakhstan with regard to the decommissioning of the BN-350 reactor and some aspects, which the Kazakhstan Atomic Energy Committee (as a regulatory body) encountered for the regulation of decommissioning activities.

1. INTRODUCTION

Decommissioning is the final phase in the life cycle of a nuclear facility [1]. According to norms [2] of nuclear regulation the shutdown reactor plant is considered to be operated until the fuel is not removed out of its boundaries. Based on this requirement, the decommissioning may be determined as a complex of administrative and technical measures and actions carrying out with the goal to achieve non nuclear status of reactor plant and reduction of its radiation hazard to the level, established by the legislation, or completes elimination of radiation hazard. The achievement of the declared goal ensures the possibility to cancellation of some or all the measures of regulating control in respect of the facility. These measures include unloading of fuel and its disposal for long term storage, deactivation, dismantling and removal of radioactive materials, waste, elements and structures. They are carrying out in order to reduce the level of radiation hazard and realized on the basis of preliminary planning and estimation to ensure the safety during the period of decommissioning.

The decommissioning is defined by national peculiarities and for each particular facility depends on technical, economical, political and other aspects. Reactor plant is considered to be decommissioned when its final condition is achieved, that is determined by the Decommissioning Project and confirmed by the regulatory body. This phase should be provided by appropriate regulations and guides for planning, safe implementation and completion of decommissioning activities. There are certain special aspects, which can be encountered during decommissioning regulations by the regulatory body.

2. HISTORY OF THE PLANT

Experimental-industrial BN-350 reactor facility – fast neutron sodium-cooled reactor [3] – is a constituent of the Mangyshlak Atomic Energy Complex (MAEC) located near Aktau city in the part of the eastern Caspian Sea shore belonging to the Republic of Kazakhstan. It was designed and built for electricity generation and seawater desalination for the Aktau region. Planned thermal power of the reactor – 1000 MWth, electrical power – 350 MWe.

The BN-350 reactor was commissioned in 1973 and operated for its design life of 20 years.

In 1993 on the basis of estimation of actual reactor condition, qualified personnel availability and taking into consideration significant progress in fulfillment of measures by safety enhancement it was concluded about possibility of extension of BN-350 reactor facility lifetime until 2003. Thereafter it operated on the basis of annual licenses of the regulator body - the Kazakhstan Atomic Energy Committee (KAEC) and positive conclusion of its safety level from General Designer (VNIPIET, St. Petersburg), Chief Designer (OKBM, Nizhni Novgorod) and Research Manager (FEI, Obninsk) of reactor facility.

In March 1998, however, the reactor was shutdown for annual maintenance, but it was not allowed to restart, since KAEC would not grant a further license until a set of safety improvements had been implemented.

In April 1999, the Government of the Republic of Kazakhstan adopted the Decree on the Decommissioning of BN-350 Reactor. This Decree establishes the conception of the reactor plant decommissioning. The conception envisages three stages of decommissioning.
The first stage of decommissioning aims at putting the installation into a state of long-term safe enclosure. The main goal is an achievement of nuclear-and radiation-safety condition and industrial safety level. The completion criteria for the stage are as follows:

- Spent fuel is removed and placed in long term storage;
- Radioactive liquid metal coolant is drained from the reactor and processed;
- Liquid and solid radioactive wastes are reprocessed and long-term stored;
- Systems and equipment, which are decommissioned at the moment of the reactor safe store, are disassembled;
- Radiation monitoring of the reactor building and environment is provided.

The completion criteria of the second stage are as follows:

- 50 years is up;
- A decision about beginning of works by realization of dismantling and burial design is accepted.

The goal of the third stage is partial or total dismantling of equipment, buildings and structure and burial.

3. MAJOR REASONS FOR DECOMMISSION OF BN-350 REACTOR FACILITY

Financial problems concerned with further reactor operating

Analysis of possibilities of reactor safety enhancement shows that for attainment of the acceptable safety level it requires about three years, during of which the reactor will stay and it will be necessary to bear expenses by reactor keeping and remodelling. Main components of costs on this period are follows:

- Operating costs of reactor manufacture;
- Costs by carrying out of modernization, including:
  - replacement of equipment that operating time is up;
  - contract payment for work fulfilment;
  - reactor safety accordance to international standard and enhancement of safety culture.

As a whole continuation of the reactor operation isn't compensated at the expense of fuel economy and will be unprofitable.

Technical problems

The reactor facility was created till appearance basic documents by safety and has considerable quantity of requirement departures. On the basis of facility analysis on accordance of present-day requirements, one compiled list of departures and worked out complex of measures by highest possible facility accordance to necessary at that time safety level.

However a number of substantial measures by enhancement of reactor safety remained unfulfilled, in particular:

- It was not carried out remodelling of the reactor control and protection system;
- It was not realized project of a seismic reserve diesel electric power station;
- It decreased the level of operative safety of personnel availability.

Because of practical impossibility to realize follow-on under non-payment Russian organizations didn't approve engineering solution of BN-350 operating prolongation. Since the approved engineering solution is absent, KAEC refused to give permission on further reactor operating.

IAEA experts in the frame of IAEA Operational Safety Review Team (OSART) mission, that realized expertise of reactor operating safety in 1998, also came to conclusion that financial and material resources received by plant during last years are not enough for maintenance of needed minimum reactor safety level.
At this conjuncture, taking into consideration financial and technical problems of further reactor BN-350 operating, in view of plant conclusion of unsecuring acceptable safety level during reactor operating and OSART mission recommendations [4] the Government of the Republic of Kazakhstan adopted the Resolution № 456 "On decommission of reactor BN-350 in Aktau city Mangistau region".

4. ORGANIZATION OF DECOMMISSION PROCESS

KAEC is a part of the Ministry of energy and mineral resources of the Republic of Kazakhstan and has special executive and supervision duties in the sphere of nuclear weapon non-proliferation regime and regulation of safe use of the atomic energy in the Republic of Kazakhstan.

Some main duties of KAEC in the Decommissioning Project are as follows:

- Supervision of project implementation by licensing and engagement of state licensed institutions only for decommissioning works;
- Consideration, approval and authorization of technical documentation of enterprises and institutions implementing the project within its competency.

During the development and implementation of the project all participants must act according to the laws and regulations of the Republic of Kazakhstan and to IAEA recommendations. In a case of absence of the regulatory documentation BN-350 decommissioning works shall be performed in accordance with the standards existing for operational facilities after they have been approved by KAEC.

KAEC shall arrange development of necessary regulatory documentation if it is required.

The following project reporting is required by KAEC:

- Obligatory submission of all project documentation to KAEC for approval;
- Submission of regular reports on works fulfilled during the reported period by institutions, licensed for activities related to atomic energy use, to KAEC.

In compliance with [2] approved by KAEC all activities related to designing (developing), manufacturing and operation of BN-350 systems and components, important for safety, during facility decommissioning period shall be regulated by the special standards and rules approved for use in the atomic power engineering by state supervising and control bodies.

The principal regulatory documents are on "Basic list of normative legal documents related to nuclear energy use in Republic of Kazakhstan", approved by KAEC. In addition to this list it was compiled the list of documents in order to comply with all of the standard technical documents for decommissioning the BN-350. This list is regularly analyzed, reviewed and updated to capture the ongoing changes. This list is attached in full to the BN-350 reactor safety analysis decommissioning report. The international standards, rules and guides are recommended for use and reference in organizing and fulfilling activities related to BN-350 decommissioning.

Pursuant to the Ministry order Joint Stock Company "KATEP" (KATEP) is authorized to represent the state interests in the BN-350 Decommissioning Project. KATEP being the general customer defines the responsible subcontractors for each stage of the work, as well as their functions and responsibility and provides responsibility for the Ministry.

Pursuant to the given order KATEP has developed "The scheme of management the activity on BN-350 decommissioning", that is the basis defining both the organizational structure and procedure for interaction of BN-350 decommissioning organizations-participants.

According to "the Scheme..." the main participants of the project are as follows:

- General Contractor - Mangyshlak nuclear power plant (MAEC), BN-350 site, Aktau City, Republic of Kazakhstan;
- General Designer - the state unitary institution, Russian design and research institute of complex power-engineering technology (VNIPIET), St-Petersburg, Russian Federation;
- Research manager - the state research center of Russian Federation Physics-power-engineering institute (GNC RF FEI), Russian Federation, Obninsk City.

The other main contractors (heading institutions) of Bn-350 decommissioning project are as follows:

- Nuclear Technology Safety Center (NTSC), Almaty, Republic of Kazakhstan;
- National Nuclear Center of the Republic of Kazakhstan (NNC RK), Kurchatov City, Republic of Kazakhstan;
- Argonne National Laboratory, USA

As the work volume is very large and BN-350 decommissioning is a very durable process the essential number of another contractors including foreign ones (RAOTEKH, MosNPO "Radon", etc.) will be engaged into the decommissioning activity.

KATEP provides the following administrative and managerial functions in BN-350 decommissioning activity:

- Defines the institutions participating in BN-350 decommissioning activity;
- Establishes duties and responsibilities for participants of BN-350 decommissioning process;
- Through daughter organization or directly signs the agreements with contractors for whole complex of work at the stage of BN-350 decommissioning and namely:
    - with General Contractor for on whole complex of BN-350 decommissioning work;
    - with general Designer on development of the Project and provision of field supervision over the activity on the Project;
    - with research Manager on the scientific maintenance of both design and project realization.
- Provides both control of fulfillment and acceptance of contract work;
- Follows the timely submittance-acceptance of the work (fulfillment of work schedule);
- Organizes the analysis and expert reviewing (coordination) of the results of work;
- Submits the reports to Ministry of energy and mineral resources on fulfillment of both the work schedule and costs (for half a year);
- Develops the cost projects and provides the funding of work.

All the participants engaged into the activity on decommissioning of BN-350 shall unconditionally recognize the right of KATEP as the institution fulfilling the function of a General Customer.

The responsibilities of the project participants are as follows:

VNIPIET is responsible for:

- Observance of the approved procedure for designing;
- Quality of project documentation;
- Terms of the work fulfillment.

MAEC is responsible for:

- Quality control of project documentation;
- Observance of terms for expert review and coordination of project documentation with General Contractor.

Contractors (subcontractors) - engineering-design, design, building, erecting, setting organizations as well as the equipment manufacturers participating in BN-350 decommissioning activity are responsible for:

- Quality of work (supply, services) fulfilled;
- Observance of work ((supply, services)) schedule in accordance with the agreement commitments;
- Safety of the work under fulfillment.

Detail commitments of counterparts fulfilling of the work are listed in the contracts.

As the decision on the reactor decommissioning was adopted before the end of scheduled operation (2003), the plan to decommission the BN-350 reactor had not yet been developed. To determine the activities required for ensuring reactor safety and in preparation for decommissioning, the Ministry of energy and mineral resources of the Republic of Kazakhstan developed and approved, a “Plan of priority measures on BN-350 reactor decommissioning”. This plan has the status of managerial and ruling document and defines the activity on provision the safety of BN-350 and preparation to decommissioning within the period until the “Project of BN-350 Decommissioning” is approved. Actions provided for the plan (first edition) include the following:

- Development of BN-350 Reactor Decommissioning Project;
- Accident prevention during the period of transition;
- Unloading nuclear fuel from the reactor and draining the coolant from the heat exchange circuits.

In the case of BN-350 decommissioning the Decommissioning Project is being equated to what in the EU is called Decommissioning Plan. Because such a document has not been produced in Kazakhstan before, and because even the process of nuclear decommissioning (and its licensing) is new, both the customer organization and to the government as a whole, and taking into account the BN-350 uniqueness and high potential hazard as well as the technological complicity of the forthcoming work on reactor plant decommissioning, it was judged necessary to product a specification for the technical specification itself, and to establish the responsibilities and inter-relations of the main parties, and to set out the subsequent stages in the regulatory and approval process. This process has resulted in two “Special technical requirements” (STR) documents:

- Special technical requirements «General provisions on BN-350 reactor plant Decommissioning Project development";
- Special technical requirements on development the BN-350 Decommissioning Project.

Special technical requirements «General provisions on BN-350 reactor plant Decommissioning Project development" cover the development and realization of BN-350 reactor plant Decommissioning Project (hereinafter referred to as Project) and are its integral part. STR define and establish the project stages and the main requirements for technical tasks, Decommissioning Project, as well as define the procedure for coordination and approval of project documentation and basic lows and standards, to which the project should correspond. Observing the main provisions and meeting the requirements of STR is mandatory for all legal and physical entities, which take part in project development and realization.

STR "General provisions on project development…" defines two main stages of project development:
- Development of Decommissioning Project;
- Development of project documentation.

Special technical requirements for project development define the BN-350 Decommissioning Project development procedure, establish the project stages, as well as the project documentation composition and contents and procedure for coordination and approval.

As soon as the first edition of the "Plan of priority measures..." provided for the measures which were to be completed until 2001 and most part of the work provided for by the Plan was completed in January 2001 KATEP has developed the second edition of the Plan - "Alteration no 2. Plan of priority measures on BN-350 decommissioning in Aktau City (for 2001-2003)". The forthcoming work on BN-350 decommissioning are grouped in new edition of the Plan into four chapters:

- Measures on BN-350 Decommissioning Project development;
- Measures on provision the reactor safety within the transitional period;
- Measures on sodium drainage and utilization;
- Measures on spent fuel disposal for long term storage.

Measures on BN-350 Decommissioning Project development includes:

- Development of "General provisions" of Decommissioning Project;
- Development of the project itself;
- Development of working documentation for the project.

«Basic provisions» of BN-350 Decommissioning Project are planned for development till the end of 2002.

To attract the international society to the fulfillment of the works on the decommissioning of the BN-350 reactor the decision was made to develop "The Decommissioning Plan of the transfer the BN-350 reactor into a state of long-term safe enclosure". The development of this Plan is being conducted under the State Department of USA with the technical support of USA National Laboratories, EC and other international organizations (including the Russian ones).

5. PRINCIPAL ACTIVITIES

Since 1999 until present the works have been intensively conducted according to the "Plan of priority measures on BN-350 reactor decommissioning".
Main task of the stage of setting BN-350 reactor in safe long-term storage state in the field of handling with spent nuclear fuel (SNF) is removal of SNF from the reactor site.
By now the following works have been made:

- For the purpose of preparation of SNF for transportation and storage, and assurance of favorable conditions for temporary store of fuel in the reactor cooling ponds before transportation, nuclear fuel was packed in sealed casings filled with inert gas;
- According to decision made by Government of RK, the complex "Baikal-1" located on the site of National Nuclear Center (Kurchatov city) in Eastern-Kazakhstan region was selected as the place for long-term storage of SNF of BN-350 reactor. SNF will be transported to the place in transport packages by railway transport.

The second main task is removal of liquid metal coolant from heat-exchange circuits and equipment, bringing of it into explosion-and fire-safe condition and placing of products of processing for long-term storage.

In order to decrease radiation activity of sodium of the primary circuit and, as a result, to decrease dose burden on personnel at processing of sodium and handling with products of processing, it is planned to carry out purification of sodium of the primary circuit from radio nuclides of cesium.

By now the following works have been made:

- It was completed the assembling of the installation for purification of sodium using cesium traps on the basis RVC installed in the primary circuit;
- It was approved the technical requirements to the project of drainage of sodium of the primary circuit.

The next main task is the processing (conditioning) and storing of radioactive wastes accumulated in the process of operation and formed at the decommissioning process.

By now it was approved the technical requirements to the project of the technological installation of the processing of the liquid radioactive waste. It will be used ion-selective purification method (with the purpose of reducing the volume of radioactive wastes) with further cementing of radioactive deposits and placing of cement compound in concrete containers.

6. CONCLUSION

Information stated above allows to conclude that the process of decommissioning is very complex. Often there are problems, the solution of which is still to be found even in the developed countries.

As the experience has showed, the chosen decommissioning strategy has proved to be efficient. In spite of the absence of the Decommissioning Project a big amount of work has been intensively conducted in a short period of time (1999-2002).

KAEC faces the real interest of foreign organizations and namely IAEA in all the works BN-350 were conducted safely and on a high level of quality in accordance with the recommendations of this authoritative organization.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, The Regulatory Process for the Decommissioning of Nuclear Facilities, Safety Series No. 105, Vienna (1990).

[2] PNAE G-1-011-89 General Conditions of NPP Safety Provision (OPB-88).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of liquid metal cooled fast reactor technology, IAEA-TECDOC-1083, Vienna (1999).

[4] Report of the OSART mission to the BN-350 NPP Republic of Kazakhstan, 30 November to 11 December 1998, IAEA-NSNI/OSART/98/101.

ECONOMIC ANALYSIS OF NUCLEAR POWER PLANT FOR DECISION MAKING IN THAILAND

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Abstract

Electricity demand in Thailand from now up to the year 2011 will rise more than 87 %. EGAT predicts that total demand will increase from about 16,126 MWe now to about 22,282 MWe in year 2006. Thailand has a diversified energy resource base, consisting of natural gas, oil, lignite, and hydropower. This paper will present the methodology to select power plant, from coal-fired, oil-fired, combined cycle, and nuclear power plants. By using economic analysis for cost benefit analysis (CBA) of the power generation cost of power plant with consideration of the net present worth (NPW) and interest rate return (IRR), it shows that nuclear power plant will offer the lowest IRR, and the highest IRR is combined cycle power plant. While the NPW for the best worthwhile projects and suitable investment is imported coal-fired power plant and the runner-up is nuclear power plant at 4-6 years construction period.

1. INTRODUCTION

According to National Economic and Social Development's forecast, electricity demand in Thailand from now up to the year 2011 will rise more than 87 %. So, the Eighth-Ninth National Economic and Social Development Plans (NESDP)(1997-2006) has launched the main energy resources, imported oil, coal, imported coal, natural gas and hydro. From the Tenth NESDP up (2007-) may launch the energy option more, such as liquid natural gas and nuclear.

Due to the favorable economic growth rate of Thailand since 1987, the expansion in industries, the construction of several industrial estates, including infrastructure, and the growth in business and commercial sectors, there was a high growth in power generation.

The rate of electricity demand in the country has risen at more than 8 % annually in the past two years, about 1,200 MWe every year. This mean the EGAT has to increase its generating capacity by at least 1,200 MWe each year. At this rate, the EGAT predicts that total demand will increase from about 16,126 MWe now to about 22,282 MWe in year 2006. Table 1 shows the forecasting electricity demand in Thailand and the electricity generating capacity of EGAT.

Table 1. Forecasting Electricity Demand and Generating Capacity [1,2]

Year	Demand (MWe)	Increasing Demand(%)	Generating (MWe)
2000	14,918.3	8.79	22,269.0
2001	16,126.4	8.10	22,034.8
2002	17,308.0	6.75	24,779.8
2004	19,610.9	6.65	27,258.8
2006	22,282.5	6.81	29,405.3

Although Thailand has reserved lignite and natural gas enough for more than two centuries, we have found that the energy resources are inadequate and expected to be imported for over 60%. So nuclear energy is necessary and suitable for the alternative sources of energy.

2. Energy resources

The most important raw material in electric industry is energy resources. Hydro, natural gas, oil, coal, lignite, and nuclear are energy resources now, and in the future, some resources such as solar, wind, geothermal, and fusion energy will be developed by engineering methods to produce electricity. Thailand has limited energy resources. Table 2 shows the reserve of Thailand energy resources. EGAT, therefore, they are considered as follows:

Table 2. Thailand's Energy Resources Reserve

Resources	Reserves	Used	Remaining Reserves
Hydro (MWe)	12,734	2,886.27	9,847.73
Natural Gas (TCF)	26.5	14.1	12.4
Oil & Condensate(million barrels)	1,150	-	498
Lignite (million tons) [1]	1,917.2	227.2	1,690

[1] As at June 30, 2002.

2.1. Hydro-electric power

The hydro-electric potential in Thailand is estimated at about 12,734 MWe[1] [3]. Up to 2001 only 2,886.27 MWe of hydro power capacity was used and during development is about 1,000 MWe [4]. The remaining potential of about 8,847.73 MWe is not easy to develop due to opposition from environmentalists. However, multipurpose hydro projects are still useful for future development.

Normally, hydroelectric power plants could produce more electricity than average of water inlet to reservoir because of the cheaper electric equipment cost. So, hydroelectric power is suitable to generate electricity at peak load. Because of water in reservoir is limited, a hydroelectric power plant can produce electric only when the water level reached the minimum requirement level. Normally, water will both be used to produce electricity and serve agriculture.

2.2. Natural gas

The natural gas reserve in Thailand is estimated to be about 26.5 TCF.(4,872 million barrels)[2] comprising 23 fields off-shore and 3 fields no-shore. It is expected to have a commercial recoverable reserve of 14.1 TCF., and 12.4 TCF. is proven reserve. Up to 2000, natural gas reserves have been developed for use at 1,971 MCF. a day [4] and it will increase in the future. So this resource is limited. The minimal natural gas resources used by EGAT is about 345,314.67 MCF. a year.

Natural gas is a clean-fuel and has less environmental effect. But the problems are storage and transportation because gas was a high volume but a low mass. Natural gas can be liquidized to Liquefied Natural Gas (LNG). It is not only used to produce electricity but also raw material of plastic.

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[1] As at Sept, 1997.
[2] Natural Resources Department, as at January 1, 1998.

2.3. Oil

The crude oil reserves in Thailand are small. The known reserves of crude oil and condensate have been estimated at about 1,150 million barrels. It is expected to have a recoverable reserve of 498 million barrels. The minimal crude oil resources used by EGAT is about 781.56 million litres a year.

As fuel, oil can be used in steam, diesel, and gas turbine power plants. Normally, gas turbine and diesel power plants will be used to generate electricity at peak load.

2.4. Lignite

There are various lignite resources scattered in Thailand. The total geological reserve is estimated at about 2,330 million tons. Up to 2001, lignite reserves have been developed for use at 22.1 million tons a year. The minimal lignite resources used by EGAT are about 15.24 million tons a year. The significant reserves located in the north and the other is located in the south.

Lignite is a bad coal, with high moisture and high oxygen, which generates carbon dioxide (CO_2) and water when it burns. Lignite fuel is controlled by environmental impact on air pollution such as sulphur dioxide (SO_2), dust, etc. The lignite price trend is likely to increase in the future.

Table 3 shows the system-installed capacity of Thailand.

Table 3. Thailand's System Installed Capacity (as at September 30, 2001) [4]

Source	Installed Capacity (MWe)	Percentage (%)
EGAT 's power plants	15,000.40	68.08
Thermal	6,255.00	28.39
Combined cycle	5,074.60	23.03
Hydro	2,886.27	13.10
Gas turbine	778.00	3.53
Diesel	6.00	0.03
Renewable energy	0.53	-
Purchase from	7,034.40	31.92
Domestic private (CC [a], Gas [b])	6,694.40	30.38
Neighbouring (Hydro, Laos) [c]	340.00	1.54
Total	**22,034.80**	**100.00**

[a] Combined cycle
[b] Gas turbine
[c] Purchase hydro-electric power from Laos

2.5. Nuclear

Nuclear energy is an alternative source of energy that can used for power generation. Presently, nuclear power plants are improved in safety, efficiency and environmental effect. Nuclear and thermal power plants are alike, only boiler of thermal power plant which changes to reactor of nuclear power plant. Nuclear power plant is suitable for base load and it will stop to replace fuel rod only one month per year. Total investment cost of nuclear power plant is higher than the others but the fuel cost is very cheap and stable.

3. ECONOMIC ANALYSIS STUDY OF NUCLEAR POWER PLANT

3.1. Alternative power plants

Under this study, using the data and assumption of the following alternative power plants to consider for the selection:

(a) Nuclear power plant.
(b) Coal-fired power plant.
(c) Oil-fired power plant.
(d) Combined cycle power plant.

3.2. Component of total nuclear power plant costs

There are three components of total nuclear power plant costs

3.2.1. Power generation costs

The power generation costs are the main cost of total nuclear power plant costs which have three basic elements, capital investment costs, nuclear fuel cycle costs, operation and maintenance costs [6].

3.2.1.1. Total Capital investment costs

The total capital investment costs are the total cost of nuclear power plant project and bringing it to commercial operation. The capital investment costs are divide into four parts.

(a) Base costs : include costs associated with the equipment, buildings and structures, installation, and materials (direct cost) as well as the engineering, construction and management service (indirect cost).
(b) Supplementary costs : include costs for spare parts, contingencies and insurance.
(c) Financial costs : include escalation, interest and fees.
(d) Owner's costs : include the owner's capital investment and services costs, escalation of owner's costs, and financing of owner's costs.

3.2.1.2. Nuclear fuel cycle costs [7]

Nuclear fuel cycle costs or nuclear fuel costs mean those costs that must be recovered in the cost of energy generation. These include the costs of nuclear materials, fuel fabrication, transportation, spent fuel intermediate storage, chemical reprocessing, associated with waste management, including storage and final disposal of wastes as well as any credits realized through the sale and use of uranium, plutonium, heavy water, and other materials.

The fuel cycle of a nuclear power plant can be divided into two main stages comprising the following activities.

(a) Front-end activities: from the exploration and mining of uranium ore to the delivery of fabricated fuel elements to the reactor site.
(b) Back-end activities: beginning with shipping of spent fuel to away-from-reactor storage, spent fuel reprocessing and waste management, and ending with final disposal of reprocessing wastes or the spent fuel itself.

The cost of front-end and back-end activities must include all step of transportation, insurance, tax, and interest.

3.2.1.3. Operation and maintenance costs

The operation and maintenance (O&M) costs of power plant include all non-fuel costs and are generally divided into two components.

(a) Fixed costs : determined by the size and type of plant and the mode of operation(load following or base load operation) and are independent of the energy production.

(b) Variable costs : vary directly with energy production and depend on the number of operating hours. The number of similar units at a particular site has a strong influence on the O&M costs are affected by items such as plant staffing and outside support services, consumables, nuclear liability insurance and taxes, interim replacements, inspections, decommissioning, and administration and maintenance crews. In addition, the costs of general and administrative support functions and of providing working capital for plant O&M are included.

3.2.2. Infrastructure development costs

The infrastructure development costs are very important for the first nuclear power plant project. Normally, the infrastructure development costs are usually not included in the nuclear power generation cost. There are many tasks and activities involving certain expenses, which are needed for the implementation of a nuclear power project and a nuclear power programme. Such activities are:

(a) Development of national infrastructures (governmental, regulatory, industrial, education).
(b) Manpower development at all levels, except the training of operations staff.
(c) Regulatory and licensing costs.
(d) National participation promotion.
(e) Planning studies.
(f) Scientific research and development.
(g) Technology transfer.

These costs are difficult to evaluate and express; they can produce benefits by promoting the country's overall development. The usually accepted procedure is to assume that infrastructure development costs and the benefits resulting from them compensate each other.

3.2.3. Other costs

These costs are generally divided into two components:

3.2.3.1. Non-financial costs

These costs mean those costs that affect capability of power plant, such as load factor, net power rate, and plant economic life.

3.2.3.2. Depreciation costs

Power plants and their associated electric generation equipment, like all production equipments, decrease in worth over time as they wear out physically or are replanted by newer or more economic facilities.

The depreciation costs are the capital invested in a production facility, such as the initial capital invested in a power plant, must be recovered in some systematic fashion from the revenues it generates during its operating life [8]. These costs will affect the project cost and the interest rate return (IRR) of projects, such as interest rate, price escalation, and discounted rate.

4. COST-BENEFIT ANALYSIS (CBA)

The pure altruist would consider the net gains to those he or she cares for and might totally disregard gain or loss. If, then we substitute " benefit " for gain or advantage, and " cost " for loss or disadvantage we can immediately establish an elementary but useful proposition. This is that cost-benefit analysis(CBA) merely formalizes a common-sense concept of rationality.

The CBA set out to answer whether a number of investment projects should be undertaken and, if investment funds are limited, which one, two, or more, among these specific projects that would otherwise qualify for admission, should be selected. In choosing projects he has to ascertain which ones best satisfy the interest and objectives of the nation.

The main reason for doing social cost-benefit analysis in project selection is to subject project choice to a consistent set of general objectives of national policy. Before one project is chosen rather than another, the choice has consequences for employment, output, consumption, saving, foreign exchange earning, income distribution and other things of relevance to national objectives. CBA can be conceptualized as follows:

(a) A practical approach of assessing the desirability of projects where it is important to take a long view and a wide view(i.e. it implies the enumeration and evaluation of all the relevant costs and benefits.
(b) The process of identification, estimation, and evaluation of the net benefits associated with alternatives for achieving defined public goals.

CBA is intended to test for desirability of a project and to select infrastructure investments in particular and public expenditure in general [8]. Industrial sector is another important area of its application. Not only large but small scale projects can also be handled. It is particularly applicable in:

(a) Projects which have a long period of operation.
(b) Projects which have a far reaching nature of their positive and negative effects.
(c) Countries where market prices have been distorted by heavy reliance on protective trade policies, and all the costs and benefits included in the financial analysis are not true economic costs and benefits. Therefore, private profits do not reflect the actual contribution of the projects to the country as a whole.
(d) Countries where people are kept unemployed because minimum wage legislation and union pressure make the abundant labor to expensive, and because subsidized interest rates, concessionaire taxes on imported capital equipment and accelerated depreciation allowances make scarce capital too cheap.
(e) Projects involving some public goods.
(f) Projects being implemented in market situations which cannot be assumed to have perfect competition, as is the case in the real world, particularly for the developing countries.

4.1. Aspects of cost-benefit analysis

The means by which projects are selected can be approached from three principal viewpoints, reflecting alternatives and hence criteria by which projects are judged [9].

4.1.1. Financial analysis

The financial approach is chosen to appraise projects by considering their contribution to profits or some other measure of their objectives, which are project-specific rather than national. This kind of analysis is sometime referred to as " commercial " or " private " analysis though it can be applied to non-commercial or public projects as well. The essential feature of such appraisals is that costs and benefits are valued at market prices. For this type of analysis is valued at market prices. For this type of analysis the following assumptions must be made.

(a) All costs and benefits included in the financial analysis are true economic costs and benefits.
(b) The absence of any external costs and benefits.
(c) The absence of any public goods.
(d) The existence of perfect competition in all markets.

However, in the real world these strict conditions are not normally fulfilled. There always exist, for instance, transfer payments, externalities, public goods produced by the projects, and various forms of price distortions plus government interventions. Moreover, when the wider aims of economic policy are taken into account, the differences between financial and economic appraisal become even more pronounced. This indicates the need and importance of economic analysis.

4.1.2. Economic analysis

The economic impacts of a project are referred to as effects on the national economic development objectives which may be measured as change in GDP etc. These effects lead to changes in the value of the outputs of goods and services and consequent changes in national economic efficiency.

The economic aspects of project analysis require a determination of the likelihood that a proposed project would contribute significantly to the development of total economy and that its contribution will be great enough to justify using the scarce resources it will need. The point of view taken in the economic analysis is that of the society as a whole.

In economic analysis, the project is assessed by measuring the contribution of its net discounted benefit to national income. This requires that project inputs are identified and valued at their real opportunity costs which is the true economic cost to the economy, and outputs by their contribution to real income. This in turn requires the identification of a wider set of costs and benefits, known as externalities, more than the financial analysis; it also necessitates the valuation of all costs and benefits at shadow rather than market prices. Often market price distortion cannot be removed through basic economic policy changes because of powerful political forces with a vested interest in status quo. Under such circumstances, one way of improving economic efficiency and equity is to make economic decisions on the basis of shadow prices that reflect the true value to the country of its resources. These shadow prices may be " national parameters " (e.g. the shadow price of foreign exchange) or they may be specific to the given sector, region and/or project (e.g. the shadow wage rate for labor).

The financial and economic analysis is thus complementary. The financial analysis takes the viewpoint of the individual participants and the economic analysis that of the society. As in financial analysis, the relevant costs and benefits in the economic appraisal must be set out in a cash flow scheme. Standard discount techniques are then applied in the same manner. The difference between the two types of analysis are the necessary adjustments made to the structure of the cash flow. They relate to the identification of the relevant costs and benefits to be included and, once they have been quantified, the method of their valuation. The later includes the problem of valuing over time, or the appropriate choice of a discount rate.

4.1.3. Social cost-benefit analysis

CBA presenting only the economic costs and benefits which will be given a dollar valuation cannot be taken as encompassing all the effects of a project. The project produce some " social impacts " which are the effects on the distribution of income as well as on the psychological, social and physical well-being of the individuals affected by the project. The incorporation of this social aspect into the analysis amends the cost-benefit analysis and the overall approach of analysis carried out in this way is called " social CBA ". Such an approach involves some forms of interpersonal comparisons of income either between different income groups, or by comparing average incomes today with expected future level. The distribution between economic and social appraisal is an importance, since the latter involves the use of value judgments concerning weights to be applied to various income groups, while such value judgments are, or should be , absent from the economic analysis.

Despite serious efforts in this field of study, a number of problems remain. In particular, research is still underway to find the most appropriate methods of estimating the parameters necessary for social analysis in a way that reduces judgments to a minimum value.

The discussion on the various aspects of CBA would be incomplete if the " environmental impacts " of projects are not discussed here. These results in changes in our physical and biological surroundings as they are perceived to affect the quality of life, their valuation, and sometimes even identification, is very difficult. Therefore, these are normally included in the CBA in qualitative form.

4.2. Economic analysis methods

The most common criteria on which to base, may be classified into two main groups[8]:

(a) Criteria which consider the expenses without taking into account the time of their occurrence(which constitutes their main disadvantage); they are base on :
  - total net cash flow per monetary unit disbursed
  - average annual net cash flow per monetary unit disbursed
  - payback or capital recovery time.

(b) Criteria which consider the time associated with the expenses, using discounting procedure to equalize the amounts of money at different moments of time, they are based on :
  - Present Worth Value (PW)
  - Internal Rate of Return (IRR)
  - Benefit-Cost ratio (B/C).

4.2.1. Criteria based on present worth value

Of the criteria based on present value, the following three will be discussed:

4.2.1.1 Maximum net present worth (NPW)

The net cash flow at time t of an investment is equal to the different between the cash flow of the expected benefits, B_t , and the cash flow of the expected expenditures, C_t .

If the discount rate is equal for all future periods of time, the net present value is given by the following formula:

$$NPW = \sum_{t=T_B} \frac{(B_t-C_t)}{(1+d)^t}$$

Where
T_B is the bid difference date,
T_L is the end of the economic life,
d is the constant discount rate.

This formula assumes end-of-period payments which are discounted to the beginning of the first period. It is clear that investments with higher NPW are preferable. The discount rate may be real or nominal.

4.2.1.2. Minimum present worth of total plant costs (PWTPC)

The present worth of the total plant costs(PWTPC) can be express by the formula:

$$PWTPC = \sum_{t=T_B}^{T_L} \frac{C_t}{(1+d)^t}$$

Where
C_t is the cash flow of expected costs.

The lowest present worth of costs should be chosen.

4.2.1.3 Levelized discounted electricity generation costs

The minimum present value of total plant costs is divided by the present value of electric energy generation to obtain the levelized discounted electricity generation costs in terms of costs per energy generated(in Kilowatt-Hours, KW.h).

The levelized discounted electricity generation costs(C_g) can be expressed by the formula:

$$C_g = \frac{\sum_{t=T_B}^{T_L} \frac{C_t}{(1+d)^t}}{\sum_{t=T_B}^{T_L} \frac{E_t}{(1+d)^t}}$$

Where
C_t is the total plant costs at year t ,
E_t is the energy produced(KW. h) year t ,

The minimum values of C_g should be selected. These costs can be viewed as the rate for each unit of electric energy which must be charged to recover exactly the total plant costs, including capital investment costs, fuel cycle costs and O&M costs.

4.2.2. Criteria based on internal rate of return(IRR)

The internal rate of return, r , an investment with benefit and cost streams B_t and C_t , respectively, is defined as the discount rate at which the net present value becomes zero. The appropriate discount rate, r , can be obtained be the following equation:

$$\sum_{t=T_B}^{T_L} \frac{(B_t-C_t)}{(1+r)^t} = 0$$

It will be of economic interest to commit only those investment projects whose rate of return, r , is greater than the capital cost.

4.2.3. Criteria based on benefit-cost ratio(B/C)

This is defined as ratio of the present value of the time stream of benefits to the present value of the total plant costs, that is:

$$B/C = \frac{\sum_{t=T_B}^{T_L} \frac{B_t}{(1+d)^t}}{\sum_{t=T_B}^{T_L} \frac{C_t}{(1+d)^t}}$$

The decision rule is to accept a project for which the ratio exceeds or equals to unity, and to reject it otherwise.

5. DATA AND ASSUMPTION

The alternatives are considered construction costs, capital investment cost, and power generation costs

Criterion by using the following data and assumption:

(a) Coal-fired 1,000 MWe , FGD
(b) Oil-fired 1,000 MWe
(c) Combined Cycle 600 MWe
(d) Nuclear[6]
  (i) Reference Plant 1,139 MWe , PWR Plant
  (ii) Reference Cost Date 1986
  (iii) Reference Commercial Operation Date 1993
  (iv) Case Study:

— Reactor Type	PWR
— Gross Capacity	1,000 MWe
— Date of Estimate	1997
— Base Date (Constant Money)	1997
— Construction Time (Years)	7
— Commercial Operating Date	2005
— Discount Rate (%)	5, 7, 10, 12,15
— Plant Load Factor (%)	80
— Plant Economic Life(Years)	25 & 30
— Exchange Rate (THB/USD)	40, 42, 45

(v) Interest Rate (%) & Escalation Rate (%)

(e) Lifetime Levelized Cost.

6. CAPITAL INVESTMENT COST ESTIMATE METHODS

We use 1,139 MWe PWR as reference plant to calculate the capital investment cost of nuclear power plant in Thailand and then we breakdown cost by using size adjustment to 1,000 MWe, material/equipment and labor ratio adjustment, time adjustment, and location adjustment which are suitable for situation in economic and social of Thailand. The later includes decommissioning cost. To compare the capital investment cost and power generation cost, we calculate by using levelized cost, with thermal power plants such as 1,000 MWe coal-fired unit with flue gas desulphurization(FGD), 1,000 MWe oil-fired, and 600 MWe combined cycle. From the base year 1997, the alternatives power plant will be commissioning in the year 2005.

The calculation of capital investment cost can be separated into foreign money and local money by using constant money and discount to the reference year.

7. NUCLEAR FUEL CYCLE COST ESTIMATE METHODS

"Once-Through" fuel cycle assumption is used for reference plant. Spent fuel are considered as nuclear waste which will deposit in the suitable site. Nuclear fuel cycle cost include costs for uranium ore, conversion, enrichment, fabrication, transportation, and waste disposal. The calculation of this nuclear fuel cycle cost can be expressed by the formula:

$$FCST\ (n) \quad = \quad (Prc90).(60\ \%) + (Prcs90).(40\ \%).(1 + Es\ \%)^{n}$$

Where

FCST is the fuel cost at year 1990 + n

Prc90 is the fuel cost at year 1990 = 7.0 mills/KW.h

Prcs90 is the fuel cost with escalation at year 1990

Es is the escalation per year

The costs of each cycle are divided into two parts:

(a) Front-End: from uranium ore purchase to fabrication is about 48.2 %.

(b) Back-End: from spent fuel storage to permanent waste disposal is about 51.8 %.

The cost shares of the different fuel cycle:

Uranium	11.4 %
Conversion	1.7 %
Enrichment	19.8 %
Fabrication	13.0 %
Other	2.3 %
Total Front-End	**48.2 %**
Interim storage of spent fuel	13.0 %
Reprocessing	36.4 %
Final waste disposal	10.4 %
Credits of U+Pu	-10.4 %
Other	2.4 %
Total Back-End	**51.8 %**
Total Fuel Cycle	**100.0 %**

The reference component costs of each nuclear fuel cycle are as follow[a] :

Uranium	3.52 \$/kg U_3O_8 (10.50 \$/lb U_3O_8)
Conversion (U_3O_8 to UF_6)	12 \$/kg U
Enrichment	91 \$/SWU (Separative Work Unit)
Fabrication	200 \$/kg
Reprocessing	1,000 \$/kg HM (Heavy Metal)
Waste disposal	250 \$/kg
Plutonium credit	20 \$/kg Pu
Uranium credit	170 \$/kg U

[a] UI 1997

8. ALTERNATIVES POWER PLANT COST ESTIMATION

Base on data and assumptions above, the cost estimation are as follows:

8.1. Total Capital investment cost estimate

The estimation of total capital investment cost of nuclear power plant in Thailand in two cases will offer the highest cost , and the lowest cost is combined cycle power plant and the runner - up is imported coal-fired and imported oil-fired power plant. Table 4 shows the total capital investment cost of the alternative power plants at 25, 30 years plant economic life.

8.2. Construction cost estimate

The estimation of construction cost of nuclear power plant in Thailand is about 1,573.83 \$/kWe, while the thermal power plants such as coal-fired with FGD is about 740 \$/kWe, oil-fired is about 650.4 \$/kWe, and combined cycle is about 554.4 \$/kWe.This estimation cost do not include the interest during construction(IDC) and land. The cost of shorter construction period is cheaper than the longer construction period. Table 5 shows the construction cost of alternative power plants.

Table 4. Total Capital investment cost

Power Plants	Total Capital Investment Costs(million USD)[a]		
	Foreign Money	Local Money	Total
Plant Economic Life 25 years :			
Nuclear 1,000 MWe	925.41	212.70	1,138.11
Coal-fired 1,000 MWe	493.55	157.08	650.63
Combined Cycle 600 MWe	265.15	96.04	361.19
Oil-fired 1,000 MWe	433.06	138.56	571.62
Plant Economic Life 30 years :			
Nuclear 1,000 MWe	1,156.77	265.88	1,422.65
Coal-fired 1,000 MWe	592.26	188.50	780.76
Combined Cycle 600 MWe	371.21	134.46	505.67
Oil-fired 1,000 MWe	519.67	166.27	685.94

[a] Exchange Rate : 1 USD = 40 Bahts
commercial operation in the year 2005

Table 5. Construction cost

Power Plants	Construction Costs ($ / kWe)[a]
Nuclear 1,000 MWe	1,573.83
Coal-fired (FGD) 1,000 MWe	740.00
Combined Cycle 600 MWe	554.40
Oil-fired 1,000 MWe	650.40

[a] Exclude IDC(interest during construction) and land

Table 6. Power Generation cost of nuclear power plant [a]

mills/ kW.h

Construction Period (Year)	Cost	Discount Rate 5 %	7 %	10 %	12 %	15 %
4	- Capital Cost	13.55	16.92	22.69	26.85	33.49
	- O&M Cost	5.19	4.99	4.75	4.61	4.45
	- Fuel Cost	9.39	9.34	9.27	9.23	9.18
	Generation Cost	**28.13**	**31.25**	**36.71**	**40.69**	**47.12**
5	- Capital Cost	13.63	17.17	22.98	27.27	34.18
	- O&M Cost	5.19	4.99	4.75	4.61	4.45
	- Fuel Cost	9.39	9.34	9.27	9.23	9.18
	Generation Cost	**28.21**	**31.50**	**37.00**	**41.11**	**47.81**
6	- Capital Cost	13.96	17.67	24.13	28.92	36.79
	- O&M Cost	5.19	4.99	4.75	4.61	4.45
	- Fuel Cost	9.40	9.34	9.26	9.22	9.17
	Generation Cost	**28.55**	**32.00**	**38.14**	**42.75**	**50.41**
7	- Capital Cost	14.45	18.67	25.85	31.41	40.78
	- O&M Cost	5.19	4.99	4.75	4.61	4.45
	- Fuel Cost	9.40	9.34	9.26	9.22	9.18
	Generation Cost	**29.04**	**33.00**	**39.86**	**45.24**	**54.41**
8	- Capital Cost	14.68	19.00	26.73	32.73	42.99
	- O&M Cost	5.19	4.99	4.75	4.61	4.45
	- Fuel Cost	9.40	9.34	9.26	9.22	9.18
	Generation Cost	**29.27**	**33.33**	**40.74**	**46.56**	**56.62**
9	- Capital Cost	15.10	19.77	28.32	35.10	47.01
	- O&M Cost	5.19	4.99	4.75	4.61	4.45
	- Fuel Cost	9.40	9.34	9.26	9.22	9.17
	Generation Cost	**29.69**	**34.10**	**42.33**	**48.93**	**60.62**

[a] Present value of all costs are discounted to year 1997
Rate of Exchange = 40 Bahts / USD
Plant Economic life = 30 Years

Table 7. Power Generation cost of power plants with 5 % discount rate [a]

mills/kW.h

	Plant Economic Life					
	25 Years			30 Years		
Exchange Rate (1 USD : xx THB)	40	42	45	40	42	45
Nuclear Power Plant (Construction Cost = 1,574 USD/kWe)						
- Capital Cost	14.91	15.52	16.42	14.45	15.04	15.92
- O&M Cost	4.93	4.93	4.93	5.19	5.19	5.19
- Fuel Cost	9.33	9.79	10.49	9.40	9.87	10.57
Generation Cost	**29.17**	**30.24**	**31.84**	**29.04**	**30.10**	**31.68**
Combined Cycle Power Plant (Construction Cost = 554 USD/kWe)						
- Capital Cost	6.44	6.68	7.03	6.14	6.37	6.70
- O&M Cost	1.91	1.91	1.91	2.01	2.01	2.01
- Fuel Cost	22.91	24.05	25.77	23.16	24.32	26.06
Generation Cost	**31.26**	**32.64**	**34.71**	**31.31**	**32.70**	**34.77**
New Thermal Coal-Fired Power Plant (Construction Cost = 740 USD/kWe)						
- Capital Cost	7.12	7.39	7.79	7.44	7.72	8.14
- O&M Cost	2.73	2.73	2.73	2.88	2.88	2.88
- Fuel Cost	16.75	17.58	18.82	16.22	17.03	18.23
Generation Cost	**26.60**	**27.70**	**29.34**	**26.54**	**27.63**	**29.25**
New Thermal Oil-Fired Power Plant (Construction Cost = 650 USD/kWe)						
- Capital Cost	6.26	6.49	6.85	6.53	6.78	7.15
- O&M Cost	2.00	2.00	2.00	2.11	2.11	2.11
- Fuel Cost	22.87	24.01	25.71	22.73	23.86	25.55
Generation Cost	**31.13**	**32.50**	**25.71**	**31.37**	**32.75**	**34.81**

[a] Present value of all costs are discounted to year 1997

Table 8. Power Generation cost of power plants with 7 % discount rate [a]

mills/kW.h

	Plant Economic Life					
	25 Years			30 Years		
Exchange Rate (1 USD : xx THB)	40	42	45	40	42	45
Nuclear Power Plant	(Construction Cost = 1,574 USD/kWe)					
- Capital Cost	19.14	19.91	21.07	18.67	19.32	20.45
- O&M Cost	4.80	4.80	4.80	4.99	4.99	4.99
- Fuel Cost	9.29	9.75	10.45	9.34	9.81	10.51
Generation Cost	**33.23**	**34.46**	**36.32**	**33.00**	**34.12**	**35.95**
Combined Cycle Power Plant (Construction Cost = 554 USD/kWe)						
- Capital Cost	7.39	7.66	8.07	7.12	7.41	7.80
- O&M Cost	1.86	1.86	1.86	1.94	1.94	1.94
- Fuel Cost	22.73	23.90	25.61	22.94	24.11	25.83
Generation Cost	**31.98**	**33.42**	**35.54**	**32.00**	**33.46**	**35.57**
New Thermal Coal-Fired Power Plant (Construction Cost = 740 USD/kWe)						
- Capital Cost	8.90	9.23	9.73	9.24	9.56	10.08
- O&M Cost	2.66	2.66	2.66	2.77	2.77	2.77
- Fuel Cost	16.62	17.44	18.67	16.24	17.02	18.23
Generation Cost	**28.18**	**29.33**	**31.06**	**28.25**	**29.35**	**31.08**
New Thermal Oil-Fired Power Plant (Construction Cost = 650 USD/kWe)						
- Capital Cost	7.81	8.10	8.55	8.09	8.39	8.85
- O&M Cost	1.95	1.95	1.95	2.03	2.03	2.03
- Fuel Cost	22.74	23.87	25.57	22.63	23.74	25.43
Generation Cost	**32.50**	**33.92**	**36.07**	**32.75**	**34.16**	**36.31**

[a] Present value of all costs are discounted to year 1997
Construction Period = 7 Years

Table 9. Power Generation cost of power plants with 12 % discount rate [a]

mills/kW.h

	Plant Economic Life					
	25 Years			30 Years		
Exchange Rate (1 USD : xx THB)	40	42	45	40	42	45
Nuclear Power Plant	(Construction Cost = 1,574 USD/kWe)					
- Capital Cost	31.95	33.24	35.17	31.41	32.67	34.57
- O&M Cost	4.52	4.52	4.52	4.61	4.61	4.61
- Fuel Cost	9.20	9.66	10.35	9.23	9.68	10.38
Generation Cost	**45.67**	**47.42**	**50.04**	**45.25**	**46.96**	**49.56**
Combined Cycle Power Plant	(Construction Cost = 554 USD/kWe)					
- Capital Cost	10.27	10.58	11.14	10.12	10.42	10.97
- O&M Cost	1.75	1.75	1.75	1.79	1.79	1.79
- Fuel Cost	22.50	23.58	25.26	22.59	23.68	25.37
Generation Cost	**34.52**	**35.91**	**38.15**	**34.50**	**35.89**	**38.13**
New Thermal Coal-Fired Power Plant	(Construction Cost = 740 USD/kWe)					
- Capital Cost	14.27	14.84	15.66	14.50	15.08	15.91
- O&M Cost	2.51	2.51	2.51	2.56	2.56	2.56
- Fuel Cost	16.38	17.21	18.43	16.19	17.01	18.21
Generation Cost	**33.16**	**34.56**	**36.60**	**33.25**	**34.65**	**36.68**
New Thermal Oil-Fired Power Plant	(Construction Cost = 650 USD/kWe)					
- Capital Cost	12.53	13.04	13.75	12.73	13.25	13.97
- O&M Cost	1.84	1.84	1.84	1.88	1.88	1.88
- Fuel Cost	22.47	23.65	25.33	22.39	23.56	25.24
Generation Cost	**36.84**	**38.53**	**40.92**	**37.00**	**38.69**	**41.09**

[a] Present value of all costs are discounted to year 1997
Construction Period = 7 Years

8.3. Power Generation cost estimate

To calculate the comparison of the power generation cost by lifetime levelized energy cost at the exchange rate 40, 42, and 45 Bahts per Dollar with 25 and 30 years plant economic life, from 5 -12 % discount rate. In all case (except 5% discount rate), show that the power generation cost of nuclear fuel is higher than the power generation cost from fossil fuel. From lowest to highest power generation cost are coal-fired, combined cycle, oil-fired, and nuclear respectively. To compare with the other thermal power plants, the power generation cost of nuclear will be more expensive about 1.27-1.37 times.

The capital cost of nuclear power plant is highest and more expensive than the other thermal power plants about 2.3-3 times. While the O&M cost of nuclear power plant is more expensive than the other thermal power plants about 1.8-2.6 times. The advantage of nuclear power plant over the other thermal power plants is only the fuel cost which is cheaper than the other thermal power plants about 1.8-2.5 times.

Table 6 shows the power generation cost of nuclear power plant at 4-9 years construction period and with 5-15 % discount rate. Table 7, 8, 9 shows the power generation cost of alternative power plants with the discount rate of 5, 7, and 12 % respectively.

9. ALTERNATIVE POWER PLANTS ECONOMIC ANALYSIS

The economic parameter will affect exchange rate and discount rate calculation. To assess the economic analysis of cost and benefit of Electricity Generating Authority of Thailand (EGAT) project, real interest rate for discount rate (social discount rate) will be calculated. By the year 1992-1998, the social discount rate of Thailand is estimated at about 7.59 %. For studying power generating cost of alternative power plants in this paper, we will use social discount rate at about 12 % and discount rate at about 5, 7, 9, 10 and 12 %. For local currency (LC) and foreign currency (FC) is about 14.60 and 7.53 % respectively.

To calculation the total capital investment cost of nuclear power plant in Thailand we use 1,139 MWe PWR for reference plant. Then we breakdown cost by using size adjustment to 1,000 MWe, material/equipment and labor ratio adjustment, time adjustment, and local adjustment which are suitable for the situation in Thailand. To compare the total capital investment cost and power generation cost, we calculate by using lifetime-levelized cost, with thermal power plants such as coal-fired (1,000 MWe), oil-fired (1,000 MWe), and combined cycle (600 MWe).

10. SENSITIVITY ANALYSIS

By using the different discount rate, which will affect to the comparison of cost, for 5-7 % discount rate show that the power generation cost of nuclear power plant will have an advantage over other thermal power plants. While the higher discount rate 10-15%, nuclear power plant will not have an advantage over other thermal power plants for the power generation cost except the fossil fuel cost will be risen and the reserves of fossil fuel are limited or that country will have another nuclear power plant projects to go on.

For the case of Thailand, by studying many cases show that the power generation cost of coal-fired power plant is the lowest. By using matrix for 5,7,10,12,and 15 % discount rate, 4-9 years construction period, 25-30 years plant economic life, 40 Bahts/USD exchange rate, show that the power generation cost of nuclear power plant will be 28.13-60.62 mills/kWh.

The power generation cost of nuclear power plant will have an advantage over combined cycle oil-fired power plants at 5 % discount rate and 4-9 years construction period with the cost of 28.13-29.69 mills/kWh.

While at 7 % discount rate and 4-6 years construction period, the power generation cost of nuclear power plant will be 31.29-32.03 mills/kWh have an advantage over combined cycle and oil-fired power plants. While the higher discount rate 10,12,15%, nuclear power plant will not have an advantage over other thermal power plants.

11. CONCLUSION

The main factors used for power generating cost calculation of nuclear power plant are capital investment cost, nuclear fuel cycle cost, operation and maintenance cost, and infrastructure cost. Consequently, the parameter which indicating the performance of power plant and power generation cost is load factor, net power rating, and plant economic life. Another variable group are interest rate, escalation rate, and discount rate.

The overhead and operation cost are always changed due to the economic or other variants of interest rate, and out of schedule operation or the changing of fuel cost. In order to compare each type of power plant, we had to use present worth value analytical technique to calculate the levelized energy cost (mills/kWh) by giving present worth value of average power generation cost equal to present worth value of total cost of the project and operation of power plant.

From the comparison of power generation cost per kWh by using discount rate of 12 %, we found that the power generation cost from nuclear fuel is higher than fossil fuel at about 1.27-1.37 times. The higher or lower discount rate will affect the comparison results. For the sensitivity analysis in comparing the power plants of Thailand case study, we calculate by using discount rate of 5, 7, 10, and 15 %, in 4-9 years construction period and for 25-30 years plant economic life. At a discount rate of 5 % it shows that the power generation cost of nuclear power plant is about 28.13 - 29.69 mills/kWh cheaper than combined cycle and oil-fired power plants. For the 7 % discount rate and 4-6 years construction period we have found that the power generation cost of nuclear power plant will be about 31.29-32.02 mills/kWh cheaper than oil-fired and combined cycle power plants.

The other factors affect capital investment and power generation cost are escalation of materials/equipment and labour , local exchange rate , local inflation rate , and the selection of account system for estimation which is different from other countries. Types of nuclear power plant and vendor will also affect capital investment.

By using economic analysis for cost benefit analysis (CBA) of power generation cost of nuclear power plant with consideration of the net present worth (NPW) and interest rate return (IRR), it shows that nuclear power plant will offer the lowest IRR at about 8.62-9.89 %, and the highest IRR is combined cycle power plant about 15.54 % and the runner - up is imported coal-fired and imported oil-fired power plant as show in the table 8. While the NPW for the best worthwhile projects and suitable investment is imported coal-fired power plant about 449.24 M$ and the runner-up is nuclear power plant at 7 % discount rate and 4-6 years construction period about 281.56, 273.66, 241.66 M$ respectively, as show in the table 9.

Finally, we would like to draw your attention to the fact that there are many advantages of nuclear energy such as economic advantages as a result of price stability, ecological advantages with respect to the greenhouse effect and carbon dioxide production, political advantages in terms of energy independence, and industrial advantages in terms of benefits to the domestic economy. It can only be developed in a favourable context in the coming years if it is accepted by the citizens.

Table 8. Conclusion of Economic Analysis (Case 1) [a]

Power Plants	Generation Cost(mills/kW.h)	NPW(M$)	IRR(%)
Nuclear	45.25	-223.53	8.62
Combined Cycle	34.50	32.19	15.54
Coal-fired	33.25	81.47	14.58
Oil-fired	37.00	-15.19	11.40

[a] Discount Rate 12 % , Construction Period : 7 years
Exchange Rate : 1 USD = 40 Bahts
Plant Economic Life = 30 Years

Table 9. Conclusion of Economic Analysis (Case 2) [a]

Power Plants	Generation Cost(mills/kW.h)	NPW(M$)	IRR(%)
Nuclear :			
Construction Period:			
4 Years	31.25	281.56	9.89
5 Years	31.50	273.66	9.75
6 Years	32.00	241.66	9.24
7 Years	33.00	194.12	8.62
Combined Cycle	32.00	144.68	15.54
Coal-fired	28.25	449.24	14.58
Oil-fired	32.75	203.39	11.40

[a] Discount Rate 7 % , Exchange Rate : 1 USD = 40 Bahts

REFERENCES

[1] ELECTRICITY GENERATING AUTHORITY OF THAILAND, EGAT Power Development Plan(1990-2006), Vol. 1, Systems Planning Department, Bangkok, Thailand(1990).

[2] PUBLIC RELATIONS DIVISION, Data and Information, EGAT, Thailand(2002), http://pr.egat.or.th.

[3] ELECTRICITY GENERATING AUTHORITY OF THAILAND, Annual Report 1997, Bangruay, Thailand(1997).

[4] ELECTRICITY GENERATING AUTHORITY OF THAILAND, Annual Report 2001, Bangruay, Thailand(2001).

[5] PTTP Public Company Limited, Annual Report 2000, Natural Gas Business, (2000).

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Economic Evaluation of Bids for Nuclear Power Plants: A Guide Book, Technical Reports Series No. 269, IAEA, Vienna(1986).

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Guidebook on the Introduction of Nuclear Power, Technical Reports Series No. 217 , IAEA, Vienna(1982).

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, Expansion Planning for Electrical Generation System : A Guide Book, Technical Reports Series No. 241 , IAEA, Vienna(1984).

[9] FROST M.J, How to Use Cost-Benefit Analysis in a Project Appraisal, Gower Press, Hampshire (1975).

APPLICATION OF THE COMBINED CYCLE LWR-GAS TURBINE FOR LIFE EXTENSION, SAFETY UPGRADE AND IMPROVING ECONOMY OF OPERATING VVER-440 REACTORS

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Abstract

Economic characteristics of new technology of lifetime extension for nuclear power unit are considered in the report. The option is examined which involves decreasing in reactor capacity and compensating for power losses at the expense of erection of steam and gas topping plant with using steam from its heat recovery steam generator for loading the turbine of nuclear power unit up to nominal power level. It is shown that realization of the option will allow to cover capital investments in the plant during 5 years approximately. Besides, the technology will permit to extend NPP life time while enhance its safety and technical indices and to improve the reliability of power supply in case of gas supply troubles or reactor emergency shutdowns owing to temporary forcing one or other component.

1. INTRODUCTION

Steady rising of the rate of power generation to replace fossil fuel and release gas and oil resources for export is the most important line of Russian nuclear power development. However, economy reform in Russia lessens the value of nuclear power depreciation funds. As a result, the alternatives of creating new generation capacities of lesser capital intensity had been grown in importance, including:

– increase in NPP capacity factor;
– completion of "frozen" construction projects;
– life time extension of operating power units.

Analysis indicates that reactor operation at reduced power level (at "sparing" mode) is a very practicable and effective means to extend lifetime of a power unit (up to 15 years) with securing its safety reliably [1, 2]. Reduction in reactor capacity allows to improve NPP nuclear safety at the expense of parameter margins and to extend operation time of reactor vessel (owing to decrease in neutron flux), steam generators and pipelines (owing to reducing their power rating, coolant velocity and so on), which are critical components for determining NPP life time.

With the purpose to extend lifetime of nuclear power unit it is proposed to use new technology, namely, combined-cycle (steam and gas) topping plant at operating nuclear power plants. For life time extension and safety improvement the reactor thermal power decreases with maintaining a power generation level of NPP turbine. At that total capacity and efficiency of the combined cycle with NPP increase considerably. The NPP turbines can be used after reactor decommissioning.

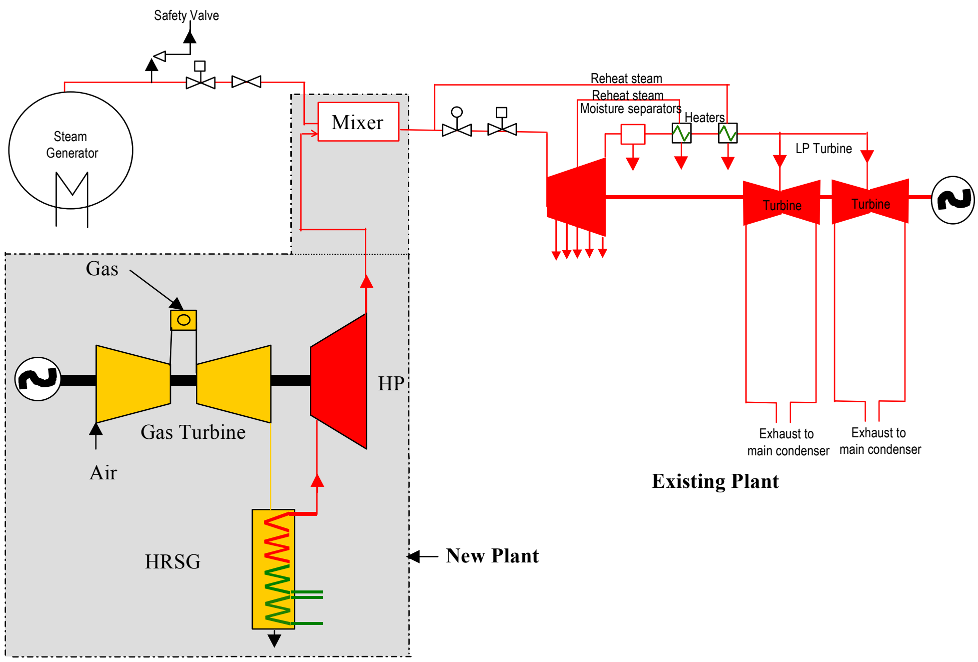
Consideration of applying the proposed technology of lifetime extension using the power unit based on VVER-440 reactors as an example is a subject of specific current interest. At present more than 20 power units with VVER-440 reactors are operated in Russia, Hungary, Bulgaria, Slovakia, Czechia, Finland. For a number of the power units their operation term comes to the end, and their parameters incompletely meet present-day safety requirements.

In previous works [1, 2] under considering the options of life time extension for VVER-440 reactor with reducing its power and applying gas turbine topping plant to compensate for power losses the engineering issues had received primary attention. In the present work economic problems are considered in more detail from viewpoint of owner of the power installation.

2. PERFORMANCE

Main performance and thermal schemes for the proposed option of life time extension for VVER-440 reactor with reducing its capacity and building gas turbine topping plant to compensate for power losses are given in [1, 2].

The schematic circuit of the nuclear power plant with steam and gas topping plant (SGTP) is presented in Figure 1.



Nuclear Reactor - Gas Turbine Combined Cycle for VVER-440/213

FIG. 1.

Superheated steam of high pressure (e.g., 160 bar, 550°C) is generated in the heat recovery steam generator (HRSG) of the gas turbine plant (GTP) and then comes to the topping plants backpressure turbine of high pressure (HPT). The HPT can operate at the shaft common with the GTP (this preferred alternative scheme is shown in Figure 1) or with individual electric generator. The principal condition – minimal modernization of NPP main turbine - should be observed. Steam from the HPT is exhausted to the mixer, where it is mixed with saturated steam from NPP steam generator and enters the NPP turbine inlet. It is possible to use various thermal schemes for topping. The steam and gas topping plant (SGP) with backpressure turbine at supercritical parameters appears to have especial promise; at that efficiency of combined-cycle NSGP may be as high as 48%.

The turbines of Russian production GTG-110 (SPU "MASHPROEKT" and SJC "Ryibinskiye motory") and GTE-170P (OSJC "Aviadvigatel" and OSJC "Permskie motory") can be used as gas turbine plants. The most promising NSGP turbines of foreign production are the turbines ABB G24, Siemens V94.2 and GE, Nuovo Pignone GT 9001FA.

Performance specification of combined-cycle power units are characterized by Table 1.

Table 1. NSGP Main Performance

Performance	NPP with VVER-440	NSGP with VVER-440		
		4×GTE-170P	4×GT24	2×GTG-110*
Reactor thermal power, MW	1324	662	662	662
Unit net power, MWe	414	1275	1160	507
Power of steam and gas topping plant, MWe	—	861	756	229
Net efficiency, %	30.115	48.3	49.3	37.9

* — variant with incomplete loading of NPP turbines.

Safety of the nuclear power unit with steam and gas topping plant is not reduced at least, because the gas turbine does not effect on a design-basis accidents progression, and the concerns associated with fuel gas are solved at the expense of choosing a proper distance between the steam and gas topping plant and NPP (~1 km). At the same time the NPP safety is improved owing to a reduction in the probability of emergency conditions occurrence due to essential widening the limits of safe operation.

Consequences of design-basis accidents are decrease considerably since fuel parameters are reduced and amplitudes of thermo-mechanical processes diminished noticeably.

3. ECONOMIC PREREQUISITES AND ALTERNATIVES

Extension of lifetime of nuclear power units is especially efficient in connection with the peculiarities of cost structure of the electricity produced by these power units. In Table 2 the data on SGP and VVER taken from [3] are given.

Table 2. Electricity Cost Structure, US cent/kWh

Costs	**NPP**	**SGP**
Capital investments	3.0	1.0
Operation & maintenance	1.0	0.5
Fuel	0.5	2.5
TOTAL	**4.5**	**4.0**

Main contribution into cost of the electricity produced by NPP is the capital component including repayment of credit, depreciation deductions, deductions into the fund of NPP decommissioning and so on. After completion of design life time of nuclear units and also after payments for credits in case of operation prolongation the capital costs decrease down to zero almost and the value of electricity production costs (fuel + operation + maintenance) is smaller than the cost of electricity produced by substitution SGP by a factor of 2 -3.

The following procedure for life time extension of nuclear power unit are considered by the example of the VVER-440 unit.

Towards the end of nuclear unit life time it changes over operation in "sparing" mode at a lower power level. Thus, the life time is extended up to 15 years, for example, and at the same time NPP safety can be improved. For compensating for the power losses produced by the unit the steam and gas topping plant with the steam output corresponding to a nominal capacity of the steam turbine is constructed. At that a total capacity of the object increases up to 1275 MW at the expense of gas turbine part and the power unit turns to category of NPPs with the capacity more than 1000 MW. After completion of additional operation term various variants are possible, in particular, the reactor can be finally shut down and it's decommissioning can be started. At that the steam and gas topping plants can be completed so that the capacity of nuclear unit steam turbines would be kept at nominal level and their operation could be continued.

For the variants with reduced power of the steam and gas topping plant the steam turbine will be under loaded and the economic benefit will be smaller in comparison with the base variant.

4. STUDY RESULTS

Economic estimates of the variants being considered were performed according to world market prices [3].

It is assumed that the construction of a steam and gas topping plant will be financed at the expense of credit for a period of 15 years (period of life time extension for nuclear power unit) with 10% annual interest rate. Repayments of the credit are performed in the form of uniform annual payments from annual profit of the nuclear unit with steam and gas topping plant under consideration; the profit is determined as difference between annual income and expenditures. While determining a net profit the payments for credit, profit tax (30%) and property tax are deducted from the annual profit.

The annual expenditures include those for the reactor and those for the steam and gas topping plant. It is assumed that in case of a reduction in the reactor power by 50% the annual expenditures for reactor will decrease in comparison with nominal power because of the proportional reduction in fuel costs. It is assumed also that annual O&M (without fuel) costs as regard to the steam and gas topping plant constitute half of the costs for the SGP which the same electric power in connection with using the steam turbine of VVER-440 power unit.

Capital investments into the construction of a steam and gas topping plant are taken according to the specific cost equal to US$ 450/kW.

Income from electricity sale was determined according to the price US cent 4.5/(kWh) [3].

Investment payback period is taken as a measure of economic efficiency for the projects.

Results of the economic estimates are given in Table 3.

The payback of investments into construction of the steam and gas topping plant is valued at 5 years (bottom line of the Table 3).

So considerable economic efficiency of the project both results from eliminating such constituents as payments for credit, depreciation deductions, deductions into the fund of NPP decommissioning, etc. from the expenditures for the nuclear power unit at the completion of its design life time and because of more full use of steam turbine capacity. Besides, extension of nuclear power unit lifetime enables to postpone for a while such a complicated and expensive procedure as nuclear power unit decommissioning.

Table 3. Expenditures for Power Unit, US$ mln

Index Name	Variant with maintaining the steam turbine capacity at a nominal level
GTP mix	4×GTE-170P
Unit capacity (nuclear power unit + GT topping plant), MW	1275
Capital investment in the steam and gas topping plant:	475
Annual expenditures, including: reactor steam and gas topping plant	240 40 200
Income:	400
Profit	160
Investment pay-back period, years	5

5. CONCLUSION

Extension of NPP lifetime is profitable economically because of the expenditures for nuclear unit is smaller by a factor 2-3 than at normal operation provided its operation after completion of its design lifetime.

In case of life time extension by means of the reduction in a reactor capacity it is effective economically for compensating for the power losses to construct the gas turbine topping plants for the nuclear power units with the purpose to use steam from their boilers-utilizers for additional loading steam turbines of the nuclear power units up to a nominal value. At that the payback period of the investments into the construction of steam and gas topping plant constitutes (in case of reactor life time extension up to 15 years) about 5 years.

Moreover, joint use of the nuclear power unit and the steam and gas topping plant will permit to improve reliability of power supply owing to temporary forcing one or other component in case of gas supply troubles or reactor emergency shutdowns.

REFERENCES

[1] Y.N. Kuznetsov, F.D. Lisitsa, V.G. Smirnov et al. Application of Steam and Gas Topping Plants for Life Time Extension, Safety Enhancement and Economics Improvement of Operating VVER-440 Reactors. ICONE-7053. The 7th Int. Conf. on Nucl. Eng. (ICONE-7). Tokyo, Japan, April 19-23, 1999.

[2] Y.N. Kuznetsov, F.D. Lisitsa, V.G. Smirnov. Combining Nuclear and Gas for Economy and Life Extension. Nucl. Eng. Int., March 2000, p. 34.

[3] J.G. Delene, J. Sheffield, K.A. Williams et al. An Assessment of the Economics of Future Electric Power Generation Options and the Implications for Fusion. ORNL/TM-1999-243.

Session 4
TECHNOLOGICAL AND OPERATIONAL ASPECTS OF PLIM

Sub-Session 4.1

Chairpersons

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PLANT CONDITION ASSESSMENTS AS A REQUIREMENT BEFORE MAJOR INVESTMENT IN LIFE EXTENSION FOR A CANDU NUCLEAR POWER PLANT

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Abstract

The life extension of a CANDU-6 reactor beyond its original design life requires the replacement of the 380 pressure tubes and calandria tubes in the reactor. Before making that major investment, one has to get assurance that it is technically and economically viable to extend Gentilly-2 NPP for an additional 20 years over its original design life of 30 years. Therefore, the condition of the plant has to be determined, and future corrections, where applicable, must be identified. Two processes were carried out: the Critical SSC studies, and the Condition Assessment studies. They are briefly described, and it is shown that the participation of the Plant System Engineers, although not always easy to obtain, was of prime importance.

1. INTRODUCTION

Gentilly-2 NPP is the only nuclear power plant in the province of Québec, one of the 10 provinces of Canada. It is operated entirely in French, as the province of Québec is mainly French speaking, in the North American ocean where English is spoken. Gentilly-2 is owned and operated by Hydro-Québec, a property of the Québec government. In addition to the activities of electricity transmission and distribution, the electricity production at Hydro-Québec comes from 51 hydroelectric plants, 4 thermal plants, 3 windmill units and 1 nuclear plant. The installed capacity is 32 249 MW, for an annual production of about 150 TWh.

Hydro-Québec, with Gentilly-2, was the first company to begin the construction of a nuclear plant of the family of the **CANDU**-6, which is an original design of a ***CAN***adian type nuclear reactor, in which the neutrons are moderated by ***D***euterium (heavy water), and which is fuelled by unenriched ***U***ranium. It is continuously refuelled at power, on a weekly basis. The Gentilly-2 gross capacity is 675 MWe. Its commercial operation began in October 1983, and its mean life annual capacity factor is 80% of full power. It was designed for a 30 years production life, with the limiting factor being the replaceable components of the reactor, namely the 380 pressure tubes/calandria tubes. So Gentilly-2 is performing very well, but is approaching the end of its original design life.

2. THE SEARCH OF LIFE EXTENSION

2.1. Schedule

A feasibility study carried out from 1998 to 2000 [1, 2] revealed that the Gentilly-2 life extension could be feasible technically, and could represent an interesting business opportunity for Hydro-Québec. A pre-project , called phase 1, was approved with a 40 M $Can budget over 4 years (from January 2001 to December 2004). Its purposes are to:

- Obtain assurance that it is technically and economically viable to extend Gentilly-2 for another 20 years beyond the original design life;
- Identify the detailed work to be done during the refurbishment period planned in 2008-2009;
- Define the overall cost and the general schedule of the refurbishment phase;
- Ensure an adequate licensing strategy to restart after refurbishment;
- Complete all the Environmental Impact Studies required to obtain the government authorizations.

If the conclusions of that pre-project are favourable, phase 2, which will consist of the detailed engineering work plus procurement, will last from very early in 2005 to about April 2008. The major work would be performed under phase 3, presently expected to take 18 months in 2008/09. With the work performed in phases 2 and 3, the life of Gentilly-2 would be extended 20 years beyond 2013.

2.2. The basic known refurbishment

Since the beginning of phase 1, it is well known that some components will have to be refurbished in the perspective of a life extension.

- The reactor core major components are the 380 fuel channels. Each consists of a pressure tube (containing 12 individual uranium fuel bundles), surrounded by a calandria tube. There is one feeder pipe at the inlet end and one at the outlet end, for each pressure tube. As stated, all 380 pressure tubes, all 380 calandria tubes and a portion of the inlet and outlet feeders will have to be replaced, due to ageing mechanisms inherent to their environment and well known since the beginning. This will be the heart of the activities.
- Due to growing problems of spare parts availability, the 2 main control computers will have to be changed. They are essential for:
    - Continuous global and local reactor power control
    - Primary Heat Transport pressure and inventory control
    - Secondary pressure control and steam generator level control
    - Other secondary controls of prime importance
- Each year, 20 to 30 tubes of the turbine condensers have to be plugged on a preventive or corrective basis. So far, about 2 % have been plugged, with a prediction of 3% in 2008, at the actual rate. That situation is not acceptable for the life extension, and a decision has to be taken about the appropriate time to retube the condensers.
- The actual GE Turbine Control and Electro-hydraulic Governor will be very soon obsolete, due to the growing difficulty of obtaining spare parts. So it is known that it will have to be changed, if the decision is taken to proceed with life extension.

2.3. The health status of the plant

Since the above facts represent by themselves an investment of a few hundred million dollars, the general condition of the Gentilly-2 Systems, Structures and Components (SSC) has to be scrutinized. For that reason, an extensive campaign is underway during phase 1, in order to:

- Give reasonable assurance that Gentilly-2 will be able to reach its life extension target after the completion of phase 3, with an acceptable capacity factor;
- Identify clearly what will be the list of the work activities that will have to be carried out during the 4 major activities previously identified. (Those which will be required to satisfy the Canadian regulatory considerations are investigated under another approach).

Two processes have been used to perform this global assessment:

- The Plant Life Management Studies for approximately 10 critical SSC's or families of SSC's (PLiM Studies);
- The Condition Assessment Studies for the other SSC's having a lower impact on plant safety or production.

2.4. The PLiM studies

That work began modestly a few years before the feasibility study. Those PliM studies were performed on SSC that were judged critical on the basis of the fact that:

- they are not replaceable or easily replaceable, such as:
  - The Concrete Containment Building;
  - The Calandria (containing the heavy water moderator);
  - The 4 Steam Generators;
- or their failure could have a severe impact on safety, on production or on the operating budget, such as:
  - The electrical motors;
  - The major pumps, heat exchangers and pressure vessels;
  - The nuclear and conventional piping;
  - The electrical cables;
  - The turbine-generator;
  - The main civil structures.

The objectives of the PliM studies are to get strong assurance that the critical SSC's studied will be able to meet their serviceability not only for their original design life (30 years), but also for the expected life extension (an additional 20 years). By providing an assessment of the current condition of the SSC, each study provides separate sets of recommendations with respect to life achievement and then for life extension. By this way, specific recommendations are obtained from the authors regarding what has to be done in addition to the current good practices, whether or not Hydro-Québec decides to invest in Gentilly-2 life extension. The PliM studies reviewed in depth the history of each critical SSC by:

- Identifying clearly the components and/or the region to be studied;
- Revisiting the design requirements and the performance expectations;

The following table shows, for each critical SSC, what was studied, and what expert performed the analysis.

Type of critical SSC	Individual PLiM reports/Content	Author
Concrete containment building	Base slab, perimeter wall, ring beam, upper dome, spent fuel transfer bay, prestressing system, mild reinforcing steel, liner system and sealant	AECL
Reactor Structure	Calandria vessel, end shields, endshield support and embedment rings	AECL
Steam Generators	All internal structures and the external supports [excludes any attached piping (heat transport system, feedwater, steam supply, etc.)]	AECL/ B&W
Electrical motors	PHT, Main BFP, Main CEP, CCW, Moderator, ECC Recovery, RCW, PHT Feed, Aux BF, SDC, RSW, Emergency WS, D2O Recovery, Vault LAC, Moderator Pony, AC ECC MV, DC T/G Lube Oil, DC ECC MV	AECL
	Electrical containment penetration	
	Medium voltage (6,9 KV) motor terminations	
Nuclear systems piping and supports	Moderator systems and auxiliaries	AECL
	Primary Heat Transport systems and auxiliaries	
	Emergency Core Cooling, End Shield Cooling, Dousing, Spent Fuel Bay and Purification, Emergency Water Supply, Liquid Injection Shutdown, Liquid Zone Control, D2O Supply, D2O Vapor Recovery and D2O Cleanup Systems	
Conventional systems piping and supports	Main Steam, Boiler Level Control and Boiler Blow-down Piping	AECL
	Separator/Reheater Drains, Turbine Steam and Steam Drains	
	Condensate, Boiler Feedwater, Extraction Steam, Heater Drains, Condenser Steam Discharge and Auxiliary Steam	
	Raw Cooling Water, Raw Water Supply, Re-circulating Cooling Water, Condenser Cooling Water	
Large pressure vessels, heat exchangers and pumps	Shutdown Coolers, PHT Purification Interchangers, PHT Purification Cooler, Moderator HX, Emergency Core Cooling HX	AECL
	Pumps: PHT, Shut-Down Cooling, Moderator Circulation, Emergency Core Cooling, Condenser Extraction Pumps, Boiler Feed Pumps, Emergency Water Supply	
	Pressure vessels: Pressurizer, Degasser Condenser, Emergency Core Cooling Gas/Water Tanks	
	Moisture Separator Reheater, HP feedwater heaters, LP feedwater heaters, Deaerator & Deaerator Storage Tank, PHT Purification Ion Exchange Columns, Moderator Purification HX, Vapor Recovery Dryers, End Shield Coolers, Drain Coolers, Recirculated Cooling Water HX	
	Pumps: Condenser Cooling Water, Raw Service Water, Raw Cooling Water, Recirculation Cooling Water, Condensate Extraction, Boiler Feed, Fuelling Machine D_2O Supply	
Electrical cables	Control, instrumentation, low, medium & high voltage power, lighting, communication & grounding	IREQ
Turbine-generator and auxiliaries	Turbine, Generator, Turbine Generator Alignment, Main Steam Valves, & auxiliary systems: EHC Control, EHC Hydraulic, Seal Oil, Hydrogen Cooling, Stator Winding Cooling, Lubricating Oil, Steam Seal	GE
Civil structures (other than the concrete containment)	Pump house	AECL
	Turbine foundation	
	Reactor Building internal structures	
	Reactor Building internal liner & penetrations	
	Solid Radioactive Waste Management Facilities	
	Spent fuel pool	
	Spent Resin Tanks, Liquid Waste Storage Tanks	

- Reviewing the design, procurement, manufacturing, construction and commissioning information;
- Reviewing the operational history (operating manual, chemistry practices and data, transients, special events, modifications) and the inspections and maintenance performed.

(The methodology for the turbine-generator and the cables studies was slightly different.)

After an identification of the ageing degradation mechanisms that could have an impact on the SSC, an assessment is made to determine if those mechanisms are or could affect the SSC, based on historical facts. That analysis leads to conclusions and finally to recommendations. The electrical cables and the turbine-generator were studied under a different approach, but with the same purpose of getting sets of recommendations for life achievement and for life extension.

2.4.1. The turbine-generator PliM study

For the turbine-generator, the author made a comprehensive revision of all the Technical Information notices applicable to Gentilly-2 that were sent by the manufacturer to the Gentilly-2 system engineers. Those notices were always taken into consideration, but not necessarily implemented, for technical reasons which unfortunately were not systematically documented. They were written to cover various problems identified through the years on similar equipment around the world. By analysing events involving components in a fleet of turbine-generators similar to that of Gentilly-2, an evaluation is obtained not only of the ageing mechanisms and their consequences, but also of the obsolescence aspect and of lessons learned from bad practices.

One of the major problems encountered at Gentilly-2 was slow turbine slab movement. Due to an alkali-aggregate reaction (AAR) in the concrete, that slab has undergone a slight but continuous expansion since the start of operation. With the consequence that sometimes its rotors had to be realigned, a total of 0.548” (1.39 cm) of coupling spacers were added between the rotors, and the covers were shimmed to the floor. These inelegant solutions were applied successfully, but as the base slab is still moving, Hydro-Quebec concrete experts were requested to study that phenomenon and to estimate the expected displacement for the next 30 years. Those results were then analysed by the turbine manufacturer. He concluded that the predicted movements are viable, but only if major work is carried out during the refurbishment outage, such as machining the sole plates to compensate for past movements. In that way, the full contact of the turbine covers with the sole plates will be restored, and the unit can be restarted without shims.

Other big jobs will have to be performed on this SSC. The two low pressure rotors will have to be changed for life extension. The generator rotor and stator will have to be rewinded. The electro-hydraulic governor, obsolete in a few years, will have to be changed (as stated earlier). Many other small jobs will also have to be carried out. If there is a refurbishment project outage, the turbine refurbishment will probably be the second biggest job, after the pressure/calandria tube replacement in the reactor core.

2.4.2. The concrete containment building PliM study

For the other PliM studies, the recommendations are numerous but generally of smaller magnitude than those for the turbine. They generally reinforce some good operation and maintenance practices, or recommend additional inspections and analyses to confirm expectations, or recommend to modify some maintenance practices, etc. Requested modifications or replacements are small in number.

The Gentilly-2 concrete containment structure was the first study to be performed, as it was early identified as an irreplaceable component. The main ageing mechanisms susceptible to impact that SSC are the freeze-thaw cycles, the concrete shrinkage and creep, frequent high internal pressure leak tests, and the AAR. Minor signs of ageing degradations have been identified, such as tiny surface cracks, but these are not significant with respect to safety and can be managed easily. However, an uncertainty has existed regarding the depth of those cracks, which could bring high humidity to the imbedded steel armature, thus causing corrosion. In 1999, some concrete test samples were extracted, up to the imbedded steel structure. That campaign demonstrated without doubt that there is no corrosion of steel armature, that the cracks are mainly on the surface, and that the humidity, one of the major conditions for AAR to occur and grow, is low enough not to represent an obstacle for life extension. In addition to this, a predictive and follow-up program is under development by Hydro-Québec concrete experts. A proven computer program is under adaptation for the Gentilly-2 structure, whereby the future movements of the structure under the local ageing stressors will be simulated. In that way, it will be possible to demonstrate the good health of that important, unique and irreplaceable SSC.

2.4.3. The calandria PliM study

The calandria, which contains the 380 pressure tubes/calandria tubes to be replaced, is a horizontal cylindrical vessel made of austenitic stainless steel. It contains the low pressure and low temperature heavy water moderator. There is one radiological end shield at each extremity, where demineralized light water circulates. The rest of the external portion of the calandria is submerged in circulating demineralized light water. Some concerns were raised due to the presence of oxygen and hydrogen in those systems, with the potential for very localized corrosion, but the review of the Gentilly-2 historical chemistry data showed that they were maintained within specifications. Nevertheless, some specific inspections are recommended as well as a few additional specific analyses, although it is recognized that the risks are low due to very good chemistry records. It was concluded that life extension is achievable. The recommendations will be implemented during phases 2 and 3.

2.4.4. The Steam Generators PliM study

The four steam generators were studied in depth using the PliM approach. Not only are they costly components, but also they cannot be replaced easily in the Gentilly-2 design. Globally, the current life cycle management program, consisting of internal inspections, cleaning, and continuous very tight chemistry control based on the All Volatile Treatment, was judged adequate for life extension prognostic. Of course, some additional recommendations came from the study, such as suggesting additional inspections and taking steps to eliminate too frequent turbine condenser leaks. After nearly 20 years of operation, it is remarkable to note that only 15 tubes out a total of 14 168 tubes (0.1%) were plugged (four (4) after extraction for metallurgic analysis, and eleven (11) on a preventive basis following non destructive examination): Gentilly-2 has never encountered a steam generator leaking tube. Based on the manufacturer’s information, Babcock and Wilcox, up to 10% of all the tubes can be plugged before the effectiveness of the Gentilly-2 Steam Generators is reduced.

2.4.5. The electrical cables PliM study

For the electrical cables, a clear identification of the insulation type families was retrieved. An evaluation of the current environmental conditions was performed and analysed based on the international knowledge regarding the ageing mechanisms that affect the different types of cable insulation used at Gentilly-2. With some recommendations, the report concludes that life extension under the normal operating conditions should not be a problem during the extension period for any type of insulation used at Gentilly-2. However, for environmental qualification aspects, a regular sampling and testing program is recommended at a frequency of every five years for those which are under the most severe constraints.

2.4.6. The other PliM studies

In general, the other PliM studies came to the conclusion that the life extension prognostic to 2033 is good, provided that some recommendations are implemented. Some of those are due to a lack of inspection, or concern additional studies to be done to demonstrate clearly, as an example, that the fatigue cycles expected until 2033 will be acceptable. Some requested additional inspections are planned during the ongoing phase 1, but these will be continued in phase 2 and completed in phase 3.

2.4.7. Gentilly-2 system engineers involvement with the PliM studies

Although those studies were carried out by external experts, from the owner-operator's point of view the involvement of the Gentilly-2 System Engineers was seen as essential. A key factor to the success of any evaluation of this kind is the close involvement of the site experts who have the systems knowledge. At Gentilly-2, as the turnover in the engineering group has been very small through the years, many system engineers have been the technical guardians of their systems for 10, 15 years, and there are even some who did the initial commissioning. So they have a unique expertise that was precious in obtaining assurance that the authors of the various PliM studies had a good understanding of all the commissioning and operating history of the different critical SSC's. In counterpart, for the few younger system engineers, it was a good occasion to revise those historical facts, and to learn more on potential future problems that will have to be taken care of if Gentilly-2 proceeds with life extension. In spite of their general interest in the project of life extension, it was not always easy to get the system engineers' collaboration because of their very high basic duty burden. However excellent cooperation was obtained from the authors, who made great efforts to minimize the additional burden on system engineers. For example, they came to the site and retrieved by themselves the documentation they needed (drawings, reports, etc.). However, the authors always carried out initial interviews with system engineers to get historical details.

Also, the system engineers revised and commented each report. That way, they were able to give assurance that:

- the facts reported in the reports are accurate and complete;
- they are now all aware of the potential ageing mechanisms and consequences on their systems;
- and they are in agreement with the conclusions and recommendations.

That last aspect is very important because, as we wrote earlier, they are the technical guardians of the design of the systems, and everything done on their systems must get their preliminary acceptance: modifications, equipment replacements, tests, operating procedures, etc. For example, if any inspection must be performed, they will be responsible to initiate it, to coordinate it and to ensure that the work to be done answers the initial request. They will close the loop by analysing the results and reporting them.

2.5. The Condition Assessment studies

The PliM studies give a good assurance that life extension of Gentilly-2 until 2033 is now feasible and will not be constrained by any critical SSC major flaw, provided that additional considerations are taken into account. However, those studies alone do not give the guarantee that other component failures will not impact severely on the overall future capacity factor. So, it became evident that another process must be established to get a more complete picture of how healthy are the SSC's that were not judged critical, but are important for safety or production.

2.5.1. The first screening

A first screening was carried out using the overall Gentilly-2 "Universal Subject Index" (USI) list. An engineer having a good general knowledge of all the important systems required to ensure safe and reliable operation of the plant visited about 5400 USI's. Those USI's without equipment, or that were judged not essential for safety or for production were rejected, with identification of the reasons. There were about 1200 remaining USI's, which were grouped by systems.

In parallel and totally independently, another engineer with similar general knowledge reviewed all operating flowsheets and culled the systems which, based on his judgment, are required for safety and production. The two lists were compared and the rare discrepancies were reconciled after discussion between the two engineers. That way, the process identified 72 systems to be investigated.

2.5.2. The system engineers

Early in the process, it was decided to get to the greatest extent the cooperation of the system engineers to perform those system condition assessments, which we called in French "Bilans De Santé" (BDS). We proposed to them a process using a common template to be used for the reports. This produced a similar approach through all the reports which have been prepared. With the involvement of the Gentilly-2 management, and after intense discussions with system engineers, despite their normal heavy burden, a great majority accepted to write those BDS's, as long as common types of components were evaluated globally by experts.

This approach would ensure that the analyses, the conclusions and the recommendations concerning those common pieces of equipment, called in English "Commodities", would be more uniform for all systems, and that Gentilly-2 system engineers would profit from very good expertise in those different fields. So it was decided to associate with AECL for them to perform those Commodities, which we called in French "génériques". This resulted in two kinds of Condition Assessments:

- The Systems Condition Assessments (in French "Bilans de santé des systèmes" or BDSS);
- The Commodities Condition Assessments (in French "Bilans de santé des génériques" or BDSG).

2.5.3. The BDSS

In the process, the author is first requested to collect, report and analyse all the relevant documentation related to:

- Design/construction/operation:
  Design manuals, construction reports, flowsheets, history docket, training and operating manuals, etc.
- Maintenance/operation reports:
  Technical reports, event reports, normal preventive maintenance tasks, all past work orders, etc.
- Others:
  PliM studies, interviews with the system engineer (in some cases, a non system engineer did the analysis), return of operating experience, etc.

Subsequently, the author has to describe the system to be analysed:

- A short system description, with a very simplified diagram
- Interfaces with other systems
- Statement of the active and passive functions
- Description of the major modifications/corrections performed on the system since and including initial commissioning (physical, chemistry, maintenance, operation)
- Study of the preventive/corrective maintenance historical information. All past work orders (WO) relative to the system are classified into 5 categories (corrective functions, corrective general, preventive, modification, general). They are then analysed in order to highlight any trends. For example, a list of the quantity of WO's by year and by category can reveal some slow deterioration.
- Brief study of the chemical history data (done by a chemical expert familiar with ageing mechanisms) and of the operating history significant facts.

As the purpose of the BDSS is to come to very specific conclusions and recommendations, all the important pieces of equipment of the systems have to be analysed. First, by computer, the equipment list assigned to a BDSS is split in two to separate the equipment which will have to be analysed under the Commodities (BDSG) from that which will have to be under the BDSS. Then, for all remaining pieces of equipment in the latter, the author has to screen them, one by one, in order to determine if it is acceptable or not to let it run to failure. Six criteria are used:

- Critical (is the equipment essential to any of the system functions related to safety or production?)
- Available (is there any redundancy or good historical availability?)
- Possibility of isolation
- Relatively easy access (for repair/maintenance)
- Repair/replacement costs (not too significant)
- Availability of spare parts

The final decision to keep important equipment to be analysed will be based on those criteria and on the author's best judgment.

If the repair/replacement costs could be significant, a second step is performed to look at the actual preventive maintenance program to judge whether the current program is adequate with regards to the functions and the ageing mechanisms. This activity, called the second level screening, identifies the equipment which can be off-ramped without further analysis.

Finally, through a list of ageing mechanisms, the author identifies which ageing mechanisms could have an impact on each piece of equipment not off-ramped. He then briefly analyses each applicable ageing mechanism with regards to the functions of the equipment in the system. After also briefly discussing obsolescence potential problems, the author concludes and states very specific recommendations classified under five categories:

- To be implemented during the refurbishment outage;
- To be implemented during normal outages;
- Inspections required to support conclusions of the analysis;
- Additional maintenance/inspections required for life extension, in order to take care of the identified ageing mechanisms;
- Business opportunities.

2.5.4. The BDSG

A similar pattern is used to carry out the BDSG studies, with the exception that they are initially orientated to type of equipment instead of being basically system oriented. That similarity is not surprising, because when the decision was taken to go on a system approach for BDS's, those who were responsible for developing the general process of the BDSS's and the BDSG's had many discussions in order to clearly understand the purpose of the two types of studies and the available tools to do them. AECL experts did all the BDSG studies:

- 12 BDSG's in Instrumentation and control
- 20 in Process and mechanic (valves, piping, civil structures, reactor building penetrations)
- 3 in Electricity

At some point in the process, there was an attempt to obtain information from the system engineers through written questionnaires. This was not a great success. Instead, many face-to-face interviews were organised, with definitively a better return. All the reports are or were reviewed by system engineers. The recommendations are classified under the same categories as for the BDSS's.

3. CONCLUSION

A comprehensive program has been completed, which demonstrates that Gentilly-2 can technically achieve a 20-year life extension to 2033. That confidence was obtained from two processes, the PliM studies and the BDS studies. A small team of four engineers was required to coordinate from the owner's point of view. But that confidence was also achieved with the essential close participation of the system engineers in those two processes, thus assuring that they continue to get an integrated picture for each system.

Of course, some major pieces of equipment will have to be refurbished, but this is feasible. However the costs will be high. A first phase of cost evaluations is being completed. The Pre-Project Team is confident to be able to demonstrate to Hydro-Québec upper management that the investment in the Gentilly-2 refurbishment project will be a better business opportunity than replacing that nuclear production by a fossil fuel thermal plant.

However, there are some major clouds in the Gentilly-2 future sky. One of them is the uncertainty coming from our Canadian Regulator. Even if the current safety level is judged adequate, if as a condition for life extension, the requests to retrofit Gentilly-2 are too extensive, a mature, safe and healthy plant, free of greenhouse gas emission, will be killed.

REFERENCES

[1] MICHEL H. ROSS, MARC AUBRAY, Should a Gas and Hydro Utility Invest in a nuclear PLEx? Paper presented at PliM+PLEx, Madrid (1999).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Cost Drivers for the Assessment of Nuclear Power Plant Life Extension, IAEA-TECDOC-1309, Vienna (2002).

FRENCH RPV PTS ASSESSMENT:

AN OVERVIEW OF EDF RESEARCH & DEVELOPMENT IN PROGRESS ON THERMALHYDRAULIC, MATERIALS AND MECHANICAL ASPECTS

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Abstract

A significant extensive Research & Development work is conducted by Electricité de France (EDF) related to the structural integrity re-assessment of the French 900 and 1300 MWe reactor pressure vessels in order to increase their lifetime. Within the framework of this programme, numerous developments have been implemented or are in progress related to the methodology to assess flaws during a pressurized thermal shock (PTS) event.

The paper contains three aspects : a short description of the specific French approach for RPV PTS assessment, a presentation of recent improvements on thermalhydraulic, materials and mechanical aspects, and finally an overview of the present R&D programme on thermalhydraulic, materials and mechanical aspects.

Regarding the last aspect on present R&D programme, several projects in progress will be shortly described. This overview includes the redefinition of some significant thermalhydraulic transients based on some new three-dimensional CFD computations (focused at the present time on small break LOCA transient), the assessment of vessel materials properties, and the improvement of the RPV PTS structural integrity assessment including several themes such as warm pre-stress (WPS), crack arrest, constraint effect ….

Keywords: RPV, PTS, Thermalhydraulic, Materials, Fracture Mechanics, LOCA, Warm pre-stress, Crack arrest

1. INTRODUCTION AND OBJECTIVES

The innocuity of flaws, such as surface or embedded flaws, has to be demonstrated in a RPV structural integrity assessment, particularly in case of severe overcooling pressurized thermal shocks (PTS). Several approaches are used around the world, according to the reference codes and standards applied in the RPV PTS assessment (ASME, RCCM, KTA ...). These approaches can be very different on several aspects, as flaw size, flaw location, role of cladding, safety margins.

The French approach for RPV PTS assessment is defined in the RCCM code [1]. This approach is very specific, by taking notably into account some realistic flaws (located in the cladding or in the base metal) and the role of cladding in the structural integrity analysis (both on thermal and mechanical aspects) [2-9]. The RPV structural integrity assessment is mainly based on the use of simplified methods ('engineering approach') instead of a more sophisticated approach such as three-dimensional finite element elastic-plastic computations [6].

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2. A FEW WORDS ABOUT THE FRENCH RPV PTS ASSESSMENT METHODOLOGY

2.1 A specific RPV PTS assessment methodology

As previously indicated in the introduction, the French RPV PTS assessment methodology is very specific, with the main following characteristics :

- the taking into account of realistic flaws (size and location), based on manufacturing and in-service inspections, located in the cladding or in the base metal of the vessel (figure 1)
- a deterministic approach based on the computation of the stress intensity factor at the crack tip and its comparison to the material fracture toughness (K_{Ic} for the base metal, K_{Jc} for the cladding)

- the use of a reference fracture toughness curve (K_{Ic}) – indexed on the RT_{NDT} – for the base metal fracture toughness properties
- a set of transients to take into account in the assessment, whose the leading are the small break LOCA (SBLOCA) and LOCA transients
- some safety coefficients depending on the occurence of postulated transients
- the taking into consideration of the stainless steel cladding in the structural integrity assessment, both on thermal and mechanical computations
- the use of a generic simplified approach based on 1D or 2D finite element computations, instead of more sophisticated analyses (such as three-dimensional finite element computations)

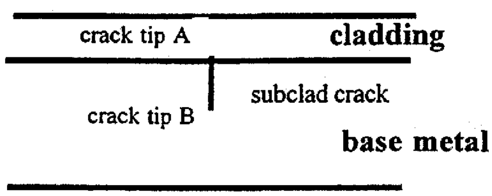
This methodology is described in the French RCCM code [1]. More details can be found in recent papers [6-9].

2.2 An engineering structural integrity assessment based on linear elastic analyses with additional plasticity correction

The French RPV PTS assessment methodogy is <u>deterministic</u>, based on the computation of the elastic stress intensity factor K_I at the crack tips A & B (for an embedded flaw) and its comparison to the material fracture toughness K_{Ic} (for the crack tip in base metal), including some safety factors :

- <u>For the subclad flaws assessment</u>, an additional plasticity correction □ (K_{cp} = □ K_I) is then applied to the pure elastic stress intensity factor K_I in order to take into account plasticity at the crack tips and cladding yielding due to the PTS event (the mechanical behaviour of a subclad flaw is very sensitive to the cladding yielding)
- <u>For embedded flaws located in the cladding</u>, no additional plasticity is required, the assessment is based only on a purely elastic approach ($K_{cp} = K_I$)

The main elements of this approach will be described in more detail in the section 5 of the paper.



These engineering RPV PTS structural integrity assessment methods have been validated by numerous comparisons with two-dimensional and three-dimensional elastic-plastic finite element calculations, taking into account numerous geometries (crack size and location) and loadings. Whatever the configuration (crack size and location, thermal-mechanical loading), these comparisons show that the engineering approach is always conservative compared to some reference elastic-plastic approaches, with reasonable margins and a sufficient level of accuracy.

Some additional informations related to this French RPV PTS assessment approach and the corresponding validation are available in papers [6-7][10].

3. RECENT IMPROVEMENTS ABOUT THE FRENCH METHODOLOGY: REDEFINITION OF SOME SIGNIFICANT THERMAL TRANSIENTS

The previous RPV PTS assessments have shown that the most severe loading conditions were the small break loss of coolant accidents (2" and 3" SBLOCA) due to the pressurized injection of cold water into the downcomer of the RPV. Due to the excessive conservatism of previous thermalhydraulic analysis, an up-to date analysis has been recently carried out by EDF and Framatome for the 3’’ SBLOCA in order to improve the calculated distribution of fluid temperature in the downcomer and consequently improve the safety margins regarding brittle failure of the vessel [11]. Considering the conservative definition of the temperature profile of these leading transients (mostly 2″ and 3 ″ breaks), additional margins are to be gained through a more accurate thermalhydraulics approach, closer to the physical phenomenon involved.

The thermalhydraulic analysis has been performed in two steps [11] : firstly, by running the system code CATHARE for global parameters (absolute pressure, mass flow rates..), secondly by applying a 3D CFD code currently used by FRAMATOME (STAR CD) to the complex determination of fluid-fluid mixing conditions in the downcomer and heat transfer with RPV structures. Lastly, through a fast fracture analysis, the safety margins have been calculated for different locations of subclad defects in the downcomer and compared to the required criteria in the corresponding category of situations.

This section deals with the numerical thermalhydraulic study carried out to obtain the complex distribution of the fluid temperatures on inner RPV wall, followed by the fast fracture mechanics evaluation. As an illustration of the improvement due to this up-to-date methodology, the gain in terms of safety margins is presented for the CP0 RPV, chosen as they are the oldest plants with the highest EOL RT_{NDT} values of the 900 MWe series.

3.1 Thermalhydraulic analysis

The accidental conditions identified as being the most severe are the small break loss of coolant accidents (SBLOCA), during which cold water is injected from the water tank and accumulators into the cold legs and falls directly into the downcomer of the RPV. This thermal shock under pressure belongs to the well known PTS transients. A complex thermalhydraulic situation occurs leading to a non-symmetrical distribution of fluid temperatures and non-linear fluid heat transfer coefficients.

3.1.1 Improvement of the Thermalhydraulic analysis

The first methodology applied in the 1980s was based on the following steps :

- the global parameters of the primary circuit computed by the system code FRARELAP by FRAMATOME,
- the local definition of the transient in the downcomer based on an experimental correlation derived from EPRI tests (CREARE mock-up),
- the heat transfer coefficients derived from the experimental results of the VESTALE loop performed around 1980 leading to ‘envelope’ coefficients for different locations of the studied defects in the downcomer.

It was noticeable that this approach was far too conservative. So in 1997, within the frame of the French RPV safety margin reassessment, a new methodology was adopted using up-to-date CFD codes. The thermalhydraulic analysis was performed in two steps :

- the evolution of the global parameters (pressure, injection flow rate) of the primary circuit is given by the system computation code CATHARE,
- the definition of the temperatures in the downcomer, after the interruption of the natural convection in the loops, is established through an accurate analysis with a specific 3D CFD code STAR CD currently used by FRAMATOME for thermalhydraulic computations or N3S and Code_Saturne used by EDF Research and Development Division for such studies.

- ***Global definition of the 3'' SBLOCA transient (CATHARE code)***

The CATHARE model consists of :

- an axial model of the steam generators
- a full model of the ECC and auxiliary feedwater systems
- a description of the downcomer.

The following conservative hypotheses are taken into account :

- 3'' break located in a hot branch in the lower part of the pipe
- maximal flow rate and minimal temperature of the ECC water (9°C).

The system computation gives access to :

- the evolution of primary pressure and consequently of the ECC injection flow (including the accumulator injection rate),
- the evolution of the fluid temperature from the start of the break until the interruption of the natural convection in the loops.

- ***Local analysis in the downcomer (STAR CD, N3S codes)***

The computation codes have been then applied by FRAMATOME and EDF R&D to various thermalhydraulic studies. These codes are qualified for a wide range of incompressible laminar and turbulent flows with or without heat transfer and solve the Reynolds averaged Navier–Stokes equations for steady and unsteady incompressible flow with k-epsilon turbulence model.

In order to check the validity of the code for cold water injection into the downcomer, a specific qualification has been performed by FRAMATOME and EDF R&D on experimental data from representative experiments of the physical phenomena involved during the SBLOCA (thermal layers, high-density injection, thermal transfer between fluid and structures) ; as a result, this code has been assessed as being adapted to simulate PTS transients.

The studied downcomer geometry is the CP0 RPV one (First French series of 3-loops 900 MWe NPP). Considering the symmetrical configuration of the main components within the downcomer as well as that of the ECC injection, the 3D computation model consists of 1/3 total volume, described as follows :

- one main primary pump,
- one cold leg including safety injection nozzles,
- 1/3 of the RPV downcomer including an inlet nozzle and 1/3 of the thermal shield for FRAMATOME and the whole RPV for EDF R&D,
- the lower plenum with internal structures.

The energy input from the core barrel and the thermal shield is taken into account.

3.1.2 Results of the thermalhydraulic analysis

- ***CATHARE results***

The sequence of events during the 3'' break transient is summarized in the following table.

Time (s)	Event
0	3-inch break located in hot leg
24.8	Low pressure Emergency trip
24.8	Primary pump shutdown
25.1	Turbine isolation
38.2	Signal of safety injection
40.2	Safety injection start. Normal SG feedwater shutdown. Start of auxiliary feedwater
Around 400	Interruption of natural loop convection

In the primary circuit, the SBLOCA leads to an important pressure decrease down to 75 MPa approximately. At that level of pressure, there is a thermal balance between primary circuit and steam generator pressure level. Then the depressurization continues because of the great loss of energy through the break and the loss of energy through the SG tubes. In the long term, the pressure level is low enough to allow the low pressure safety injection pumps to start and the primary pressure is stabilized at the discharge pump level. The natural circulation in the primary loops is interrupted at about 400s after the start of the transient. It can be noticed at that time that :

- the lowest temperatures are obviously located in front of the inlet nozzles,
- the temperatures increase downwards and outwards from the inlet nozzles.

Throughout the whole transient, the downcomer is water filled, at least to the upper edge of cold legs. This fact allows the use of the 3D thermalhydraulic codes STAR CD and N3S, qualified for one-phase flow.

- ***Local computation (STAR CD code and N3S)***

The local 3D computation is performed up to the time when the minimum in the safety margin has been reached, i.e. up to 3500 seconds after the occurrence of the break. Three locations are studied specifically :

- the area of the cold leg between safety injection nozzles and RPV inlet nozzle,
- the RPV inlet nozzle,
- the downcomer.

The final results in terms of temperatures on the RPV shells are strongly dependent on the accurate description of the physical phenomena involved in these specific zones. The numerical simulation shows :

- in the cold leg, the progressive establishment of several thermal layers, with a cold lower flow towards the downcomer and the pump and an upper reverse flow from the downcomer and the pump towards the injection nozzle ;
- at the inlet nozzle, the fluctuation of the cold water injection which is dependent on the safety injection flow rate ;
- within the downcomer, the complex thermal field : the injection fluctuates between the core barrel and the vessel shell (behind the thermal shield) as well as under the inlet nozzle like a cold plume.

Globally, the increase in the mean downcomer temperature as compared to that derived from CREARE correlations is mainly due to the fluctuations of the cold injection and to the improvement in the heat transfer interaction with structures. As a consequence, the effect of cold water injection on the RPV inner wall is softened and that represents a major source of gain in terms of safety margin as will be shown below.
Some details and results of the thermalhydraulic analysis are given on figures 2, 3, 4, 5, 6 and 7.

3.2 Consequences on margins regarding RPV brittle failure

- ***Studied zones and geometry of selected defects :***

The geometrical characteristics of the defects taken into account are deduced from of inspection in service performances. These subclad defects, located either in base metal or in weld, are summarized in the following table :

Defect	Base metal	Weld
Orientation	Longitudinal	Circumferential
Dimensions (mm)	6 x 60	6 x 60

These defects are located in the specific area whose azimut is that of the maximum fluence value (hot spot) and cold area below the inlet nozzles (points 3, 8 and 10) or slightly moved aside this azimuth (-25°) for point 4, as indicated on figure 8.

- ***Fracture toughness K_{IC} and margin factors evaluation***

The assessment has been conducted using maximum expected EOL RT_{NDT} for the CP0 900 MWe RPV :

Zones	Points	RT_{NDT}(°C)
Base metal (C1 and C2 shells)	3 – 4 - 10	64
Weld metal	8	87

The basic principle of the approach is to define a margin factor for each defect associated with the loading transient (here 3'' break) and to compare this factor to the required margin according to the codified set of criteria (or the SBLOCA, in the third category of situations, the required criteria in cleavage is 1.6).

The RPV assessment results, given for the defects located in the points previously selected (figure 8), are presented in the following table in terms of margin factor Fm defined as the following ratio : $\boxed{Fm = K_{IC} / K_{cp}}$

Minimal Margin factor $\boxed{Fm = K_{IC} / K_{cp}}$	**Base metal** $\boxed{Fm = K_{IC} / K_{cp}}$				**Weld** $\boxed{Fm = K_{IC} / K_{cp}}$	
	Previous results	**New results (CATHARE/STAR CD)**			**Previous results**	**New results**
RT_{NDT} (°C)	**Point 3**	**Point 3**	**Point 4**	**Point 10**	**Point 8**	**Point 8**
64	**1.57**	**3.16**	**4.05**	**3.14**	/	/
87	/	/	/	/	**1.38**	**2.20**

The main trends can be summarized as follows :

- The minimal margin factors, always obtained for cleavage instability, are noticeably improved by around 50% for base metal and 40% for weld metal ; this is due to the up-to-date approach linked with the softening of the calculated cooling gradient in the RPV shell wall during the transient as compared to the previous conservative approach.
- The significant effect of the azimuthal location of the defect upon the margin factor (comparison between points 3 and 4) can be associated with the fact that the cooling under the inlet nozzle affects a limited area.
- The influence of the axial location is less noticeable (comparison between points 3 and 10) showing a quite homogeneous temperature on the inner vessel wall away from the complex injection just below the inlet nozzle.
- To sum up, the results show that the area associated with pressurized thermal shock and maximum fluence where the defects are supposed to be initiated is a very limited one.

Owing to this approach, the required criteria (1.6 for cleavage) is largely satisfied for the weld metal as well as for the base metal, taking into account conservative EOL (40-year) RT_{NDT} values. These results show a noticeable increase in term of minimal safety margins by around 50% for base metal and 40% for weld metal. As a consequence, the required criteria associated with the 3-inch SBLOCA is largely satisfied. Owing to this new methodology, the RPV integrity is firmly assessed for the 40-year design life for the CP0 RPV series. A similar analysis is in progress for the 2'' SBLOCA for the CP0 RPV configuration, as a confirmation of the 3'' SBLOCA approach and in the near future, for the CPY (3-loop) RPV configuration with a discontinuous thermal shield.

4. RECENT IMPROVEMENTS ABOUT THE FRENCH METHODOLOGY : CONSERVATISM OF K_{IC} FRACTURE TOUGHNESS CURVE

Analyses of brittle fracture resistance of PWR RPV make it necessary to determine the material fracture toughness properties at the end of its life span ie, after irradiation embrittlement.

To evaluate the effect of irradiation upon steel A508 Cl.3, the reference fracture toughness in the unirradiated state must be known. A reference curve put forward by the RCC-M code[1] is used. The curve, which is the same as ASME curve, presents the evolution of the minimum fracture toughness of vessel steel in relation to temperature. The lower bound is defined from the tests results of the most brittle materials.

The presence of segregation zones in steel may question the fracture toughness values used to evaluate the integrity of PWR power plant vessel shell rings produced from solid ingots [14]. Segregation zones are enriched in alloying and residual elements. They are displayed during the solidification of the ingot used to manufacture vessel shell rings. The axial segregations are eliminated by piercing the ingot but some of them (A or inverted V) may still be seen near the inner surface of the shell ring.

The influence of segregation zones on mechanical properties of pressure vessel steel has been determined [14]. We know that the presence of segregation zones at the notch tip of unirradiated Charpy-V specimens induces a shift of about 73°C for the transition temperature (T_{K56}). Fractographic observations of fracture surfaces of low toughness specimens revealed that the presence of large segregated zones may embrittle steel by triggering intergranular fracture.

In order to evaluate the conservatism of the RCC-M standard curve for non-irradiated vessel steel, we gathered on a same curve all the toughness values of steel A508 Cl.3 as determined from tests on specimens taken from the scrap portions of vessel shell rings in the French PWR power plants (figure 9). The reference curve of the RCC-M code indexed on the RT_{NDT} temperature is on the whole a bounding curve (except for very low temperatures), nevertheless without margin.

Similarly, all the toughness values determined from tests conducted on specimens taken from the French vessel irradiated steel have been brought together on a same curve (figure 10). That data base only includes 20 values, of which four where determined from tests made on specimens showing a segregation zone at the fatigue crack tip. All the specimens used to determine the toughness values were taken from the same vessel shell ring. On figure 10, the results are compared with the reference curve given in the RCC-M code using the RT_{NDT} indexed temperature after irradiation. The irradiated RT_{NDT} was predicted on

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[1] $K_{IC} = \mathrm{Min}\,[36{,}5 + 3{,}1\,\exp[0{,}036(T - RT_{NDT} + 55{,}5)]\,;\,220]$ [Mpa $\sqrt{m}$]

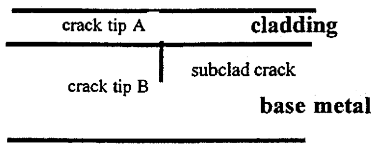
the basis of the FIS formula. We can see that the code reference curve is a global envelope curve especially concerning the values determined on specimens taken from the segregation zones.

5. RECENT IMPROVEMENTS ABOUT THE FRENCH METHODOLOGY : ENGINEERING METHODS TO ASSESS SHALLOW EMBEDDED FLAWS

A short summary related to the improvement to the RPV PTS structural integrity assessment methodology has been previously given in section 2. More details will be presented in this section.

To reduce the conservatism and improve the level of accuracy of this methodology, some significant developments have been conducted leading to new proposals of engineering methods related to the assessment of embedded flaws located under the cladding (at the interface cladding – base metal) or in the first layer of the cladding [6-8]. This work has been conducted by Electricité de France (EDF) and FRAMATOME within the framework of the studies in progress related to the French 900 MWe RPV structural integrity re-assessment.

5.1 Assessment of shallow subclad flaws



- A new plasticity correction in the elastic approach (β correction)

Based on numerous two-dimensional finite element computations involving mechanical (cladded specimens in bending or tension) and thermal loadings (cladded vessels under a PTS event), a new plasticity correction has been developped for the elastic analysis of embedded subclad flaws [6-7]. This correction (β correction), applied to the elastic stress intensity factor K_I , is based on the calculation of a parameter β depending on the r_y^A / b ratio (plastic zone size r_y^A at the crack tip in cladding divided by the remaining ligament b in cladding). The corrected elastic stress intensity factors K_{cp} at the crack tips A (cladding) and B (base metal)(embedded flaw) are given by the following expressions :

<u>For an increasing loading</u> ($dK_I / dt > 0$, $K_I \leq K_I$ max)

. At the crack tip A in cladding

$$K_{cp}^A = \beta^A . K_I^A$$

$$\text{with} \quad \beta^A = 1 + 0.3 \, [\tanh (36 . r_y^A / b)]$$

. At the crack tip B in base metal

$$K_{cp}^B = \beta^B . K_I^B$$

$$\text{with} \quad \beta^B = 1 + 0.5 \, [\tanh (36 . r_y^A / b)]$$

<u>For a decreasing loading</u> ($dK_I / dt < 0$, $K_I \leq K_I$ max)

. At the crack tip A in cladding

$$K_{cp}^A = K_I^A + (\beta^A \max - 1). K_I^A \max$$

$$\text{with} \quad \beta^A = 1 + 0.3 \, [\tanh (36 . r_y^A / b)]$$

. At the crack tip B in base metal

$$K_{cp}^{B} = K_I^{B} + (\beta^{B} \max - 1).\ K_I^{B}\max$$
$$\text{with} \quad \beta^{B} = 1 + 0.5\ [\tanh (36\ .\ r_y^{A} / b)]$$

For each crack tip (A or B), this proposal can be summarized by :

$$\mathbf{K_{cp}} = \begin{cases} \boldsymbol{\beta}\ \mathbf{K_I} & \textbf{if } \mathbf{K_I \leq K_I max} \textbf{ and } \mathbf{dK_I / dt > 0} \\ \mathbf{K_I + (\beta max - 1).\ K_I max} & \textbf{if } \mathbf{K_I \leq K_I max} \textbf{ and } \mathbf{dK_I / dt < 0} \end{cases}$$

$K_I max$ and βmax are the maximum values of the elastic stress intensity factor K_I and the correction coefficient β during the loading at each crack tip (A or B). The maximum values of the β parameter are 1.3 for the crack tip A (near cladding) and 1.5 for the crack tip B (base metal).

- Validation of β plasticity correction by comparison with 2D elastic-plastic computations

To validate this new proposal of plasticity correction, complementary finite element computations have been performed including two-dimensional elastic (K_I and K_{cp}) and elastic-plastic computations (K_J), taking into account several geometries and loadings [6]. The elastic stress intensity factors (K_I and K_{cp}) have been compared to the reference elastic-plastic stress intensity factors (K_J) in each case.

Whatever the type of loading, these comparisons between the stress intensity factors deduced from the elastic analyses ($K_{cp} = \beta\ .\ K_I$) and the elastic-plastic analyses (K_J) show that the elastic approach with the new β plasticity correction is always conservative compared to the elastic-plastic approach as soon as the cladding yielding occurs ($K_{cp} > K_J$) and more accurate that the previous plasticity corrections based on the β parameter ($K_{cp} = \alpha\ .\ K_I\ .\ \sqrt{[(2a + r_a + r_b) / 2a]}$, see [2]). These comparisons show also that the previous β correction, always very conservative during the increasing part of the loading ($K_{cp}\ (\alpha) > K_{cp}\ (\beta) > K_J$), was not necessary conservative during the decreasing part of the loading, as in the case of a PTS transient ($K_{cp}\ (\alpha) < K_J < K_{cp}\ (\beta)$).

- Validation by comparison with reference 3D elastic-plastic computations

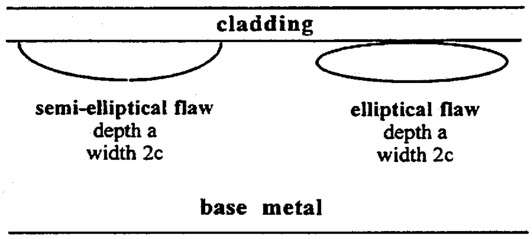
Most of the defects taken into account in the French PWR structural integrity assessment are modelled by ellipses (figure 1)[2]. For this kind of subclad defects, the elastic stress intensity factors K_I^{A} (near the interface with cladding) and K_I^{B} (deepest point in base metal) can be easily calculated according the following expressions based on the elliptical correction factors f_A and f_B taking into account the elliptical shape of the defect [3] :

$$K_I^{A} = f_A\ .\ F_A\ .\ K'^{A}_I$$
$$K_I^{B} = f_B\ .\ F_B\ .\ K'^{B}_I$$

K'^{A}_I and K'^{B}_I are the stress intensity factors for the strip in an infinite plate loaded by the stress distribution existing in the trace of the crack in the uncracked vessel (according to the superposition method). F_A and F_B account for the effect of the proximity of the free surface for a strip in plane strain under a uniform tensile loading, and are given by the stress intensity factors handbooks. Afterwards, the β plasticity correction is applied to obtain the final stress intensity factors K_{cp}^A and K_{cp}^B according to the methodology previously described.

The validity and the accuracy of this simplified elastic approach has been evaluated on a typical geometry of a French 900 MWe RPV by comparison with a reference approach based on three-dimensional elastic-plastic finite element computations. Three subclad flaws have been taken into account, a first one with depth a = 4 mm and width 2c = 60 mm, a second one with depth a = 6 mm and width 2c = 60 mm, and a third one with depth a = 12 mm and width 2c = 72 mm. The loading is a typical PTS transient corresponding to a 3'' break on the primary circuit and the materials properties are the same as data used in the French 900 MWe PWR structural integrity assessment in progress [12-13].

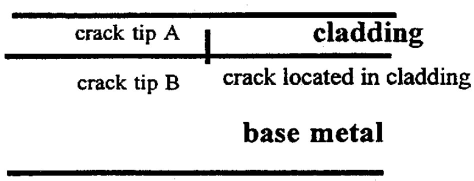
The assessment of the three different flaws is performed using both the simplified elastic approach with the β correction (considering an elliptical shape of each flaw) and the three-dimensional elastic-plastic approach (considering a semi-elliptical shape of the flaw with same depth and width as the corresponding elliptical crack).



Most of the details and results of analyses can be found in papers [12-13]. Nevertheless, the main results are summarized on the figures 11 (4 mm subclad flaw), 12 (6 mm subclad flaw) and 13 (12 mm subclad flaw) with the comparison - at the deepest point of the crack - between the stress intensity factors deduced from the simplified elastic approach based on the β plasticity correction (K_{cp}) and the 3D elastic-plastic approach (K_J). The RPV PTS assessment methodology based on a simplified elastic approach with a unique β plasticity correction appears conservative during all the transient in comparison with the 3D elastic-plastic approach ($K_{cp} > K_J$) for the different cracks. This conservatism seems reasonable.

Considering all these results, we can consider that the methodology developed for the assessment of shallow subclad flaws in RPV - based on an elastic approach with a unique plasticity correction (β correction) to take into account cladding yielding during a PTS event - is well validated <u>with a sufficient level of accuracy</u> compared to a reference approach based on 2D or 3D elastic-plastic calculations.

5.2 Assessment of embedded flaws located in the first layer of the cladding



Embedded flaws located in the cladding (particularly in the first layer of the stainless steel cladding) have to be taken into account in the French RPV PTS integrity assessment. Although the occurrence probability of such flaws is very low, the assessment of this kind of flaws is not necessary easy, particularly for the crack tip B located at the interface between cladding and base metal. It is due to a fracture mechanics methodology not necessary well adapted.

- Necessity to improve the methodology to reduce its excessive conservatism

Due to the very small remaining ligament in cladding (≈ 3 mm) and strong cladding yielding induced by the thermal loading, the elastic approach based on the β correction (first section of the paper) – initially developped for the assessment of the subclad flaws - appears to be extremely conservative for this kind of flaw compared to an elastic-plastic approach [6-7]. This conservatism is worst (higher) in this case (embedded crack located in cladding) than for subclad flaws.

- A new methodology proposed for the analysis of embedded flaws located in the cladding

In the case of embedded flaws located in the cladding (with a crack tip B at the interface between cladding and base metal), very specific results are obtained in comparison with the now well known case of subclad flaws if results just presented are examined with great care. When the flaws are located in cladding with strong cladding yielding during the loading, we notice that the elastic analysis, even without some plasticity corrections, is still conservative compared to the elastic-plastic approach, $K_I > K_J$, for both crack tips A, B (cladding and base metal)[6].

Using this important result, a new methodology has been proposed for the analysis of embedded flaws located in the first layer of cladding in RPV, based on the only use of the elastic stress intensity factor without any plasticity corrections [6-7]. We consider that a single elastic approach based on the computation of the elastic stress intensity factor K_I without any plasticity corrections is still conservative and gives higher values of K_I compared to the elastic-plastic approach based on the computation of K_J ($K_I > K_J$). It is not necessary to apply in this case some additional plasticity corrections, as in the case of subclad flaws where opposite results are obtained (elastic approach without plasticity corrections non conservative compared to an elastic-plastic approach, $K_I < K_J$).

In the new methodology proposed for the assessment of embedded flaws located in cladding of RPV, the RPV PTS assessment of such flaws is based on the comparison of the pure elastic stress intensity factor K_I with the material fracture toughness (K_{Jc} for the crack tip A in cladding, K_{Ic} for the crack tip B located at the interface between cladding and base metal). This approach presents two advantages : firstly, the elastic analysis is conservative and more easier to use than an elastic-plastic approach, secondly the use of an elastic approach avoids specific problems to elastic-plastic approaches with the computation of the elastic-plastic stress intensity factor K_J during unloading (results tend to become not path independent).

- Validation of the methodology by 2D numerical computations

In order to confirm and validate the methodology proposed for the assessment of flaws located in cladding, several two-dimensional elastic and elastic-plastic computations have been performed using the EDF finite element *Code_Aster*. The geometry taken into account is a typical French 900 MWe reactor pressure vessel containing a 4.2 mm flaw located in the first layer of cladding (4 mm in cladding and 0.2 mm in base metal in order to calculate the elastic and elastic plastic stress intensity factors K_I and K_J). The thickness of cladding and base metal are respectively 7 mm and 200 mm. Two thermal transients have been considered, a pure thermal transient without pressure (decreasing of the temperature from 280 °C to 20 °C in 20 seconds) and a severe representative PTS transient (corresponding to a small break on the primary circuit).
Several types of mechanical computations have been conducted, including notably :

. an elastic approach without any additional plasticity corrections
. an elastic-plastic approach without taking into account the residual stresses in the vessel after the stress relief heat treatment (SRHT) (main assumption : no residual stress at the beginning of the transient)
. an elastic-approach including the SRHT before the PTS transient (residual stresses are taken into account in this case)
. a non linear elastic approach without taking into account residual stresses

In each case, the stress intensity factor has been calculated at crack tips A and B (cladding and base metal).

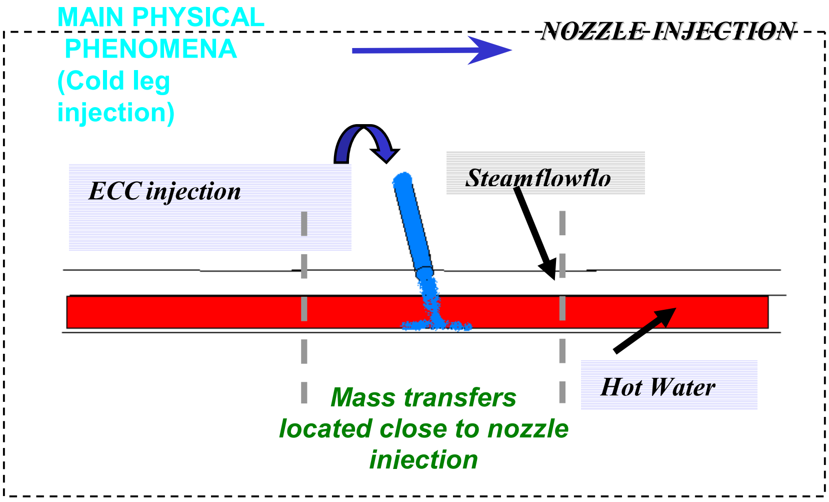
Main results are presented in the reports [6-7] and some results are summarized on the figures 14 and 15. All the results confirm the validity of the methodology proposed for the assessment of embedded flaws located in the cladding. The elastic approach, based on the computation of the elastic stress intensity factor K_I without any additional plasticity corrections, is always conservative compared to other approaches (elastic-plastic and non linear approaches) and gives higher values of the stress intensity factor ($K_I > K_J$). For embedded flaws located in cladding, it is not necessary to apply an additional plasticity correction, as in the case of subclad flaws where opposite results were obtained (elastic approach without plasticity corrections non conservative compared to the reference approach based on elastic-plastic computations, $K_I < K_J$).

6. SOME RESEARCH AND DEVELOPMENT ACTIONS IN PROGRESS

The RPV PTS re-assement of French RPV is still in progress, including the 900 and 1300 MWe plants. Within this framework, some additional Research & Development studies have been engaged on all the aspects with respect to a structural integrity assessment. An overview of the present Research & Development programme will be briefly described in this section, limited to thermalhydraulic, materials and mechanical aspects.

6.1 Actions in progress on thermalhydraulic aspects

In fact, currently, CFD softwares are mainly applied to single-phase flows in nuclear reactor safety analysis, e.g. to mixing problems and to natural convection studies in the containment but certain of these scenarios display two-phase flows situation that the CFD codes are not able to simulate (see the following figure).



ECC injection in two phase flow scenario.

For that, the Research program engaged is based on the definition of the main physical models for PTS phenomena and more particularly in the two phase flow case including turbulence effects, transition phase, evaporation and condensation [20]. A significant work will be realised into the numerical formulation and the enhancement of the coupling algorithms to achieve rapid convergence for complex two phase flow. A qualification task will be undertaken in order to validate the physical models implemented into the CFD code before its use in the frame of reactor studies.

6.2 Actions in progress on materials aspects

6.2.1 Modeling of segregated zones influence on RPV toughness steel [15-19]

In order to evaluate quantitatively effect of the presence of segregated zone near a crack tip of toughness specimen, two fracture criteria have been developed for the rupture of segregated zones [15-17].

The first one is based upon the achievement of a critical intergranular fracture stress σ_c over a characteristic distance from the crack tip. That criterion is similar to the RKR model for cleavage fracture. That model assumes that fracture is always initiated in the segregated zones : that phenomena is mainly due to the embrittling effect of phosphorus segregation to grain boundaries which may trigger intergranular fracture. The value have been determined for average and highly segregated zones from two different synthetic steels.

The second criterion has been developed, based upon a local approach to brittle cleavage fracture which is the Beremin model [18]. A 1T-CT specimen containing one segregated zone perpendicular to the crack front has been modeled by 2D finite elements.

The toughness results obtained with the critical intergranular stress criterion are shown on figure 16 and compared to experimental results obtained from the French fracture toughness values database (see above). It is observed that model is able to account for the lowest values of for the lowest values of toughness, in particular for the highly segregated zones.

The toughness results obtained with the Beremin model are shown on figure 17. Il is observed that the model gives a reasonably good evaluation of the scatter in the test results. In particular, the model is able to account for the lowest values of fracture toughness which were experimentally determined.

6.2.2 Determination of toughness by instrumented Charpy impact test [19]

Charpy-V-notch impact testing is widely used to characterize the resistance of a material, by measuring the energy consumed by a specimen during the impact. However, to justify the integrity of components, fracture toughness versus temperature curves are required and no satisfactory link exists between the Charpy impact energy and the fracture toughness.

It is possible to calculate directly the fracture toughness from the Charpy impact test. That can be done with the use of F.E.M. calculations and the local approach to fracture. The Beremin model [18] has been widely used to simulate the brittle fracture of the pressure vessel steel. Parameters of Beremin model have been identified from a 2D calculation of Charpy test compared with a large experimental program of Charpy Tests. The last step of the method is the simulation of the fracture toughness. The calculation is made with a 2D-mesh in plane strain condition. The fracture is predicted with the Beremin model, using identified values of parameters. That method gave encouraging results and toughness is predicted with a good confidence [19].

A software, named CENTENAIRE®, has been developed to make automatically the different steps of the method [19]. The same method will be applied in order to evaluate the fracture toughness in the transition at higher temperatures and by taking into account of the influence of irradiation.

6.3 Actions in progress on mechanical aspects

This overview will be limited to three aspects : the warm pre-stress (taking into account the load history effect in a RPV structural integrity assessment), the crack arrest during a PTS event, and the constraint effect. Other important aspects, such as the probabilistic approach and the master curve approach for example, will be not evoked in this paper.

6.3.1 Warm pre-stress

The French RPV PTS structural integrity assessment doesn’t take into account the potential beneficial effect of the warm pre-stress, moreover this effect is not included in the French RCCM code [1]. A significant Research programme has been recently started. Its aim is to better understand this effect in a RPV assessment, and to define and establish some recommendations for a further codification. All the elements necessary to propose a method will be gathered or obtained. This will be done through experimental works, leading to a deep understanding of metallurgical and mechanical phenomena, and through numerical works and development of models. The results will permit a much more precise prediction of a possible fracture in a RPV submitted to a PTS transient.

The first step of this programme has been recently achieved [21-24]. A large experimental programme, including numerous WPS type experiments on conventional CT specimens and development of numerical simulations, has been conducted between EDF and

MPA Stuttgart on a French RPV steel 18MND5. A lot of experimental results are now available on conventional CT specimens, and confirm the beneficial effect on the warm pre-stress effect with a significant increase of the fracture toughness of the material (figure 18). No unexpected results were observed compared to the literature results.

The second step of this programme has just started (2002 – 2004 programme), within the framework of the European project SMILE [25]. One of the key point of this project will be to confirm this effect on a 'large scale component'. A PTS transient will be simulated on a large scale vessel containing a circumferential crack. This further experiment, conducted on MPA Stuttgart test facility, must confirm the warm pre-stress effect under conditions very close to realistic PTS loading scenario. The main objectives of this project can be summarized as follows :

- a good understanding of fundamental mechanisms
- a confirmation of the effect in conditions close to realistic PTS events
- an assessment of models able to take account the WPS
- a demonstration of the capabilities of numerical models
- a preparation of a synthesis and some recommendations for a further precodification of WPS

The feasibility to confirm the WPS effect on irradiated material is also under consideration.

6.3.2 Crack arrest

The purpose of this theme is to study the crack arrest concept and its application to a RPV submitted to a severe PTS transient. Due to the temperature elevation and fracture toughness increase (temperature elevation and decreasing irradiation) through the vessel wall, a crack which will propagate after fracture initiation would be stopped after a certain jump.
An experimental and numerical has just started on this subject, including in particular some original thermal shock experiments on small cracked specimens (cracked discs submitted to a severe thermal shock loading induced by coils). The main expected objectives of such a programme are :

- a database of crack arrest toughness K_{Ia} on the French A508 Cl.3 RPV steel
- the development of a model (engineering or numerical) in order to simulate the crack growth and the crack arrest
- the application of this approach to French RPV structural integrity assessment

6.3.3 Constraint effect [26-27]

A RPV structural integrity assessment consider the behaviour of defects under normal and abnormal loading conditions to assess safety margins aand component lifetimes as materials become degraded and (or) thermal ageing. In essence, these analyses compare load and resistance terms to demonstrate that the crack driving force does not exceed the material fracture toughness. It is usual for fracture toughness data to be derived from tests on standard (deeply-notched) specimens and conservative validity criteria designed to ensure data representing high hydrostatic stresses near the crack tip (high constraint) and plane conditions. This is to provide a lower bound material property independent of specimen size. However, there is ample evidence that in many cases the loading on defects in components leads to lower hydrostatic stresses at the crack tip (lower constraint) with an associated increase in fracture resistance (e.g. shallow crack effect). Comparison between the crack driving force on a defect in a component and fracture toughness data from standrad, high-constraint specimens then has two major consequences :

- a potentially over-conservative assessment of the margins associated with the loading to which the component is subjected
- a potential economic penalty due to under-estimation of the component safe lifetime

The European project VOCALIST (Validation of Constraint-Based Assessment Methodology in Structutal Integrity) [26-27], initiated and managed by SERCO Assurance, has been initiated (2001 – 2003) to develop and validate innovative procedures for assessing the level of, and possible changes to, constraint-related saafety margins in ageing pressure boundary components. EDF is engaged in this project.

7. CONCLUSIONS

A very large and extensive programme has been initiated by Electricité de France regarding the structural integrity re-assessment of the French 900 & 1300 MWe reactor pressure vessels, in order to increase their lifetime. Within the framework of this programme, numerous research developments have been performed or are in progress, involving several aspects such as thermalhydraulic analyses, materials characterization and structural analyses.

Some important results have been already obtained, leading to a more satisfactory RPV PTS assessment and increasing margins :

- An improvement of the thermal hydraulic analyses for the most severe PTS transients such as the small break LOCA, inducing an increase of the safety margins regarding the brittle failure of the vessel
- A study of the influence of the segregated zones (inside the vessel near the cladding) on the base metal fracture toughness properties, showing the conservatism of the French RCCM reference fracture toughness curve
- The development of new engineering methods related to the assessment of shallow embedded flaws located under the cladding or in the cladding ; this methodology is now well validated with a sufficient level of accuracy compared to a reference approach based on 2D or 3D elastic-plastic computations. These ‘engineering approaches’ have been accepted by the French Safety Authority and are now currently used

The RPV PTS re-assessment of French RPV is still in progress. Some additional Research & Development studies have been engaged concerning the thermalhydraulic, materials and structural integrity analyses aspects. Concerning the structural integrity analyses, the privileged themes are mainly the warm pre-stress effect, the crack arrest and the constraint effect. A part of this work is conducted within the framework of European projects such as SMILE (WPS) and VOCALIST (Constraint effect).

REFERENCES

[1] RCCM code - January 1988 Edition 'Design and construction rules for mechanical components of PWR nuclear islands - RCCM code - AFCEN France'

[2] A. Pellissier-Tanon, J. Grandemange, B. Houssin, C. Buchalet 'French experience on the verification of pressurized water reactor vessels integrity', EPRI Research Project 2975-2, 1989

[3] A. Pellissier-Tanon, S. Bhandari, J. Vagner 'Practical methods for surface and sub-surface cracks in reactor pressure vessels', ASME Pressure Vessels and Piping Conference, San Diego, 1991

[4] H. Churier-Bossennec, P. Fayolle, C. Faidy 'Reactor pressure vessel behaviour with a small crack in the cladding', ASME Pressure Vessels and Piping Conference, Nashville, 1990

[5] D. Moinereau, C. Faidy, M.P. Valeta, S. Bhandari, D. Guichard 'Evaluation of fracture mechanics analyses used in RPV integrity assessment regarding brittle fracture', Methodology for Pressurised Thermal Shock Evaluation, IAEA Meeting, 5-8 May 1997, Esztergom, Hungary

[6] D. Moinereau, G. Bezdikian, C. Faidy 'Methodology for the pressurized thermal shock evaluation : recent improvements in French RPV PTS assessment', Specialists Meeting on Methodology and Supporting Research for the Pressurized Thermal shock Evaluation, 18-20 July 2000, Rockville, Maryland, USA

[7] D. Moinereau, C. Messelier-Gouze, G. Bezdikian, F. Ternon-Morin, Y. Meziere, C. Faidy, A. Pellissier-Tanon, J. Vagner, D. Guichard 'Some recent developments in French reactor pressure vessel structural integrity assessment', ASME Pressure Vessels and Piping Conference, San Diego, 1998

[8] F. Ternon-Morin, E. Meister, G. Bezdikian, D. Moinereau, A. Heuzé, J. Vagner Integrity assessment of French RPV. A state of the art safety margin evaluation', ASME Pressure Vessels and Piping Conference, San Diego, 1998

[9] A. Martin, S. Bugat, F. Ternon-Morin, S. Bellet 'Assessment of the French RPV integrity in PTS conditions. Thermalhydraulic and thermalmechanical studies of the SBLOCA in a 3 loop PWR plant', ASME Pressure Vessels and Piping Conference, 04-08 August 2002, Vancouver, BC, Canada

[10] D. Moinereau, G. Bezdikian, C. Faidy 'Three-dimensional assessment of flaws in a French PWR vessel submitted to a severe overcooling transient : the effect of cladding', ASME Pressure Vessels and Piping Conference, Seattle, 2000

[11] F. Ternon-Morin, A. Martin, P. Haussaire, S. Bellet, G. Bezdikian 'Assessment of French RPV integrity : a thermalhydraulic and fracture mechanics analysis of the small break loss of coolant accident', 8th International Conference on Nuclear Engineering, 2-6 April 2000, Baltimore, USA

[12] D. Moinereau, G. Bezdikian, C. Faidy 'Three-dimensional analyses of shallow subclad flaws in a PWR vessel submitted to a severe overcooling transient', ASME Pressure Vessels and Piping Conference, Boston, 1999

[13] D. Moinereau, G. Bezdikian, C. Faidy 'Three-dimensional assessment of flaws in a French PWR vessel submitted to a severe overcooling transient: the effect of cladding', ASME Pressure Vessels and Piping Conference, Seattle, 2000

[14] M. Bethmont, J.M. Frund, B. Houssin, P. Soulat 'The Toughness of Irradiated PWR Vessel Shell Rings and the Effect of Segregation Zones'. Effects of Radiation on Materials : 17th International Symposium, ASTM STP 1270, 1996.

[15] Naudin 'Modélisation de la ténacité de l'acier de cuve REP en presence de zones de ségrégation, Thesis of Ecole des Mines de Paris, 1999.
[16] C. Naudin, A. Pineau, J.M. Frund 'Toughness modeling of PWR steel containing segregated zones', proceedings, the International Conference NACE Conference, Lake Tahoe, 2001.
[17] C. Naudin, J.M. Frund, A. Pineau 'Intergranular fracture stress and phosphorus grain boundary segregation of a Mn, Ni-Mo steel', . Scripta Materialia., Vol. 40, n°9, pp. 1013-1019, 1999.
[18] Beremin F. M. 'A local criterion for cleavage fracture of a nuclear pressure vessel steel'. Metallurgical Transaction A, Vol. 14A, pp. 1277-1287, 1983.
[19] N. Verdière, A. Parrot, P. Forget, J.M. Frund 'Fracture toughness determination of a nuclear pressure vessel steel by instrumented Charpy impact test', ICONE 9, 2001.
[20] A. Martin, G. Bezdikian, F. Ternon-Morin, S. Bellet 'CFD-Tool for assessment of the Reactor Pressure Vessel Integrity in Pressure Thermal Shock conditions for Lifetime evaluation. Qualification phasis and Thermal-Hydraulic study of a Safety Injection in a three loop PWR plant', International Symposium on Nuclear Power Plant Life Management, 4-8 November 2002, Budapest, Hungary.
[21] M.P. Valeta, C. Sainte Catherine, G. Barbier, W. Lefevre, U. Eisele, S. Bhandari 'Structural behavior during a PTS transient taking into account the WPS effect', International Journal of Pressure Vessels and Piping 78 (2001) 137-146, 2001.
[22] G. Barbier, U. Eisele, W. Lefevre 'EDF/MPA collaborative program on WPS effect : Experimental results', ASME Pressure Vessels and Piping Conference, 23-27 July 2000, Seattle, USA.
[23] W. Lefevre, G. Barbier, R. Masson, G. Rousselier 'A modified Beremin model to simulate the warm pre-stress effect', Nuclear Engineering and Design 216 (2002) 27-42, 2002.
[24] R. Masson, L. Nicolas, D. Moinereau 'RPV structural integrity assessment during a PTS event : Application of an extended Beremin model consistent with WPS test results', ASME Pressure Vessels and Piping Conference, 04-08 August 2002, Vancouver, Canada.
[25] SMILE project 'Structural Margin Improvements in aged-embrittled RPV with Load history Effects', Contract n° FIKS-CT-2001-00131.
[26] VOCALIST project 'Validation of Constraint Based Methodology in Structural Integrity', Contract n° FIKS-CT-2000-00090.
[27] D.P.G. Lidbury, B.R. Bass, S. Bhandari, D. Connors, U. Eisele, E. Keim, H. Keinanen, S. Marie, G. Nagel, N. Taylor, Y. Wadier 'VOCALIST : A Programme for the Validation of Constraint Based Methodology in Structural Integrity', Transactions, SMIRT 16, Washington DC, August 2001.

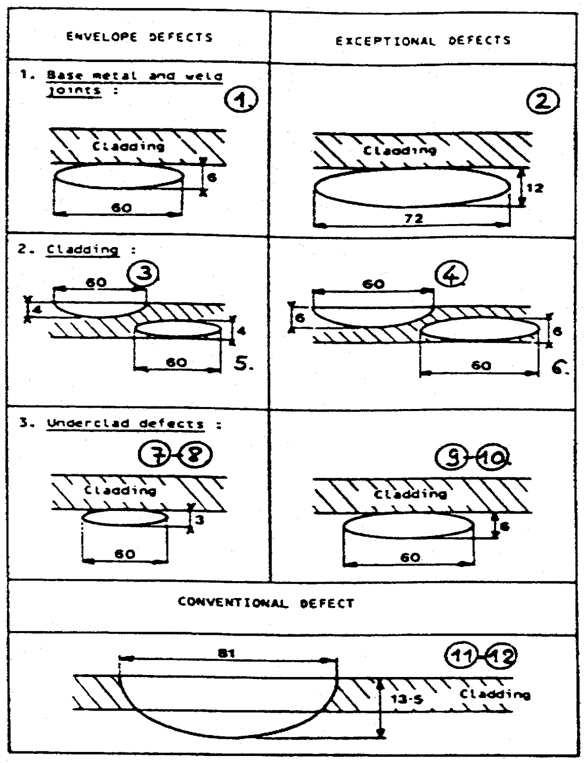


FIG.1. Reference defects for vessel beltline (dimensions in mm).

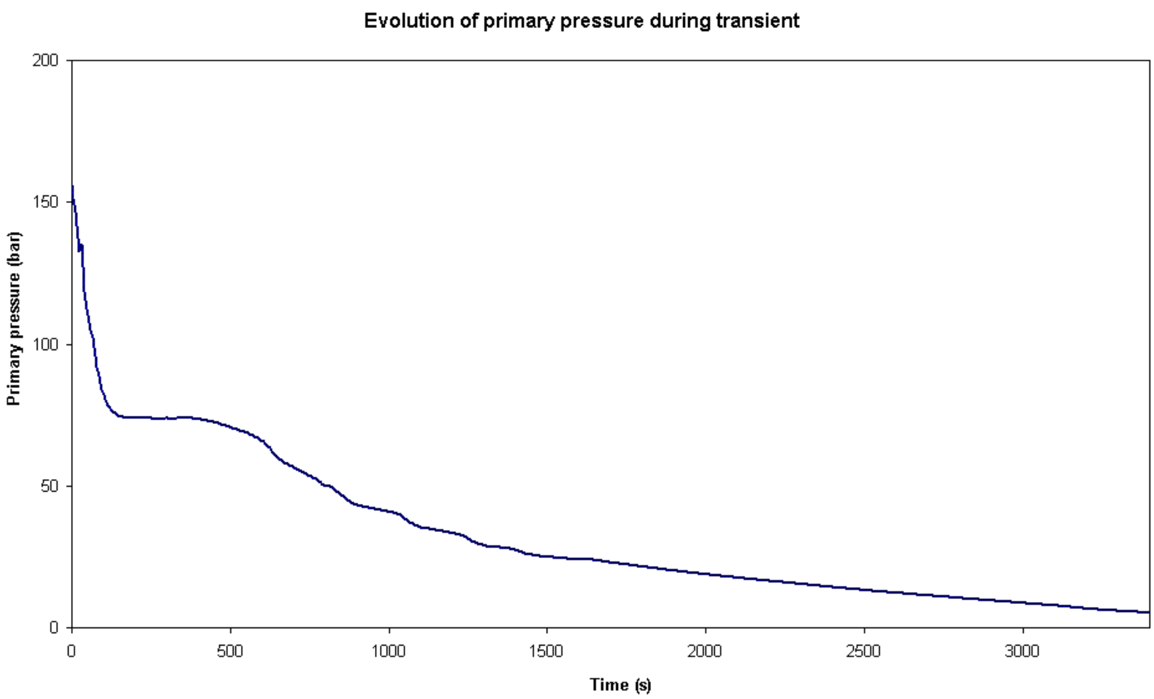


FIG. 2. CATHARE results – Evolution of primary pressure versus time.

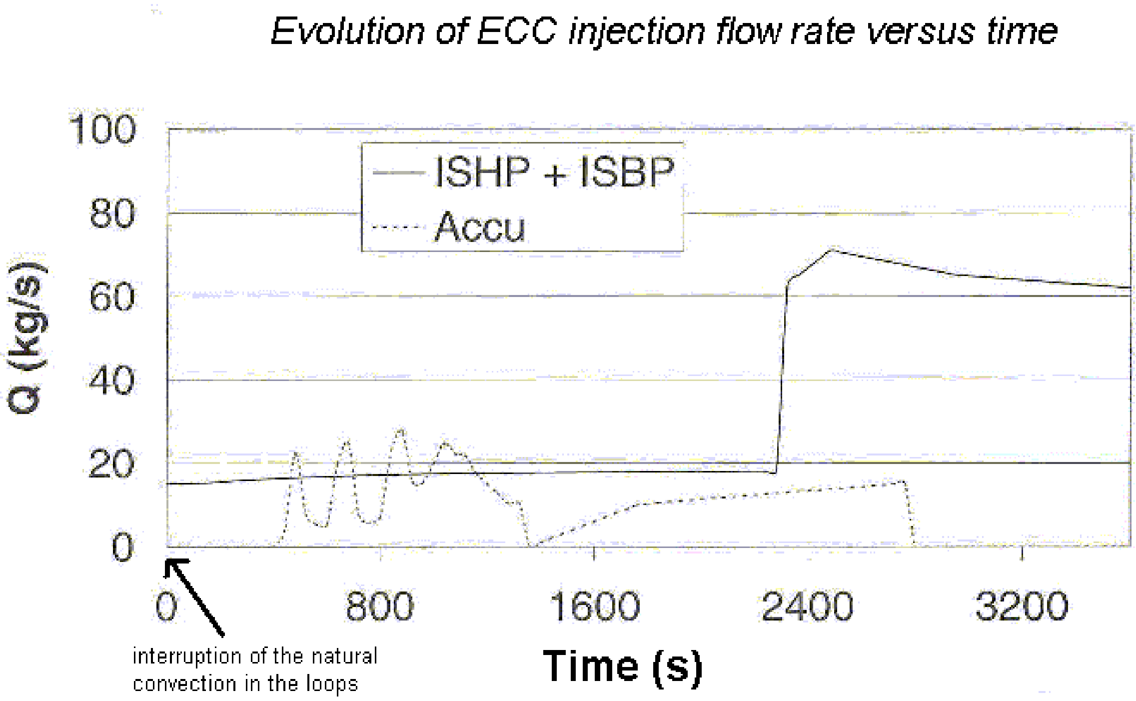


FIG. 3. CATHARE results – Evolution of ECC injection flow rate versus time.

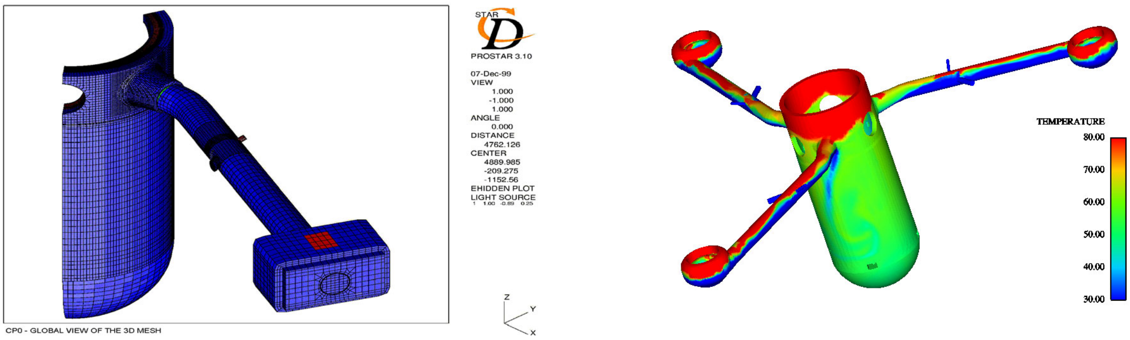


FIG. 4. Global view of the 3D geometry used for STAR CD computation.

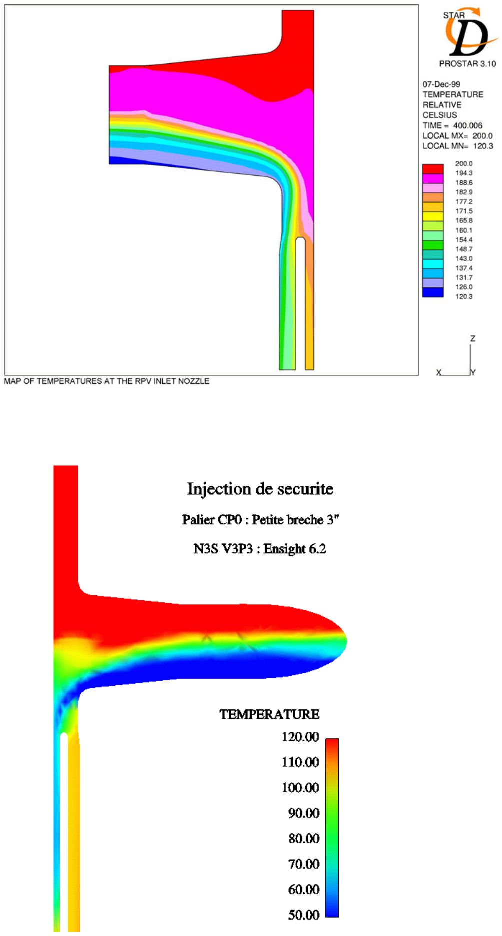


FIG. 5. STAR CD results – Map of temperature at the RPV inlet nozzle.

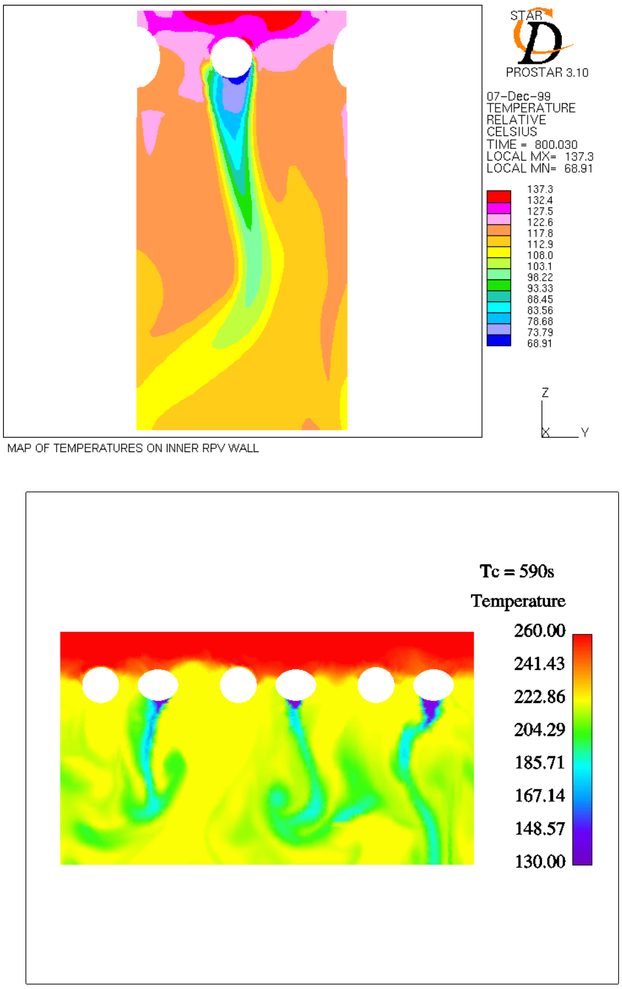


FIG. 6. STAR CD results – Map of temperatures on inner RPV wall.

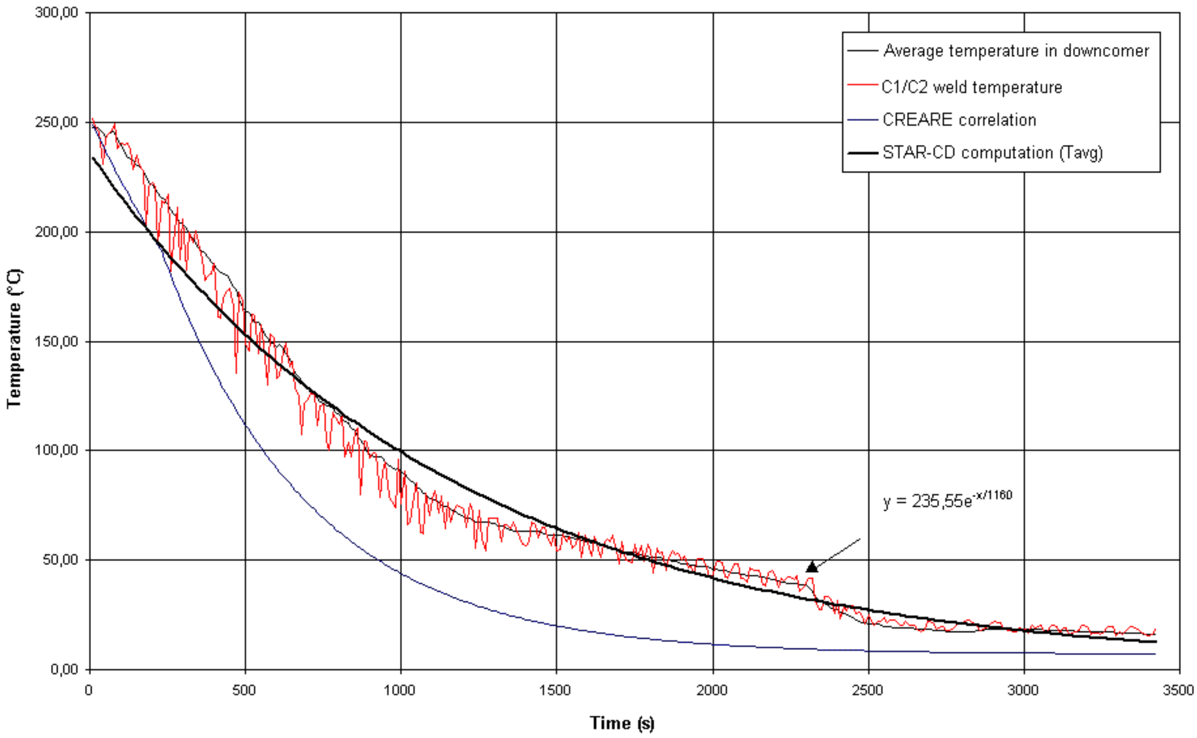
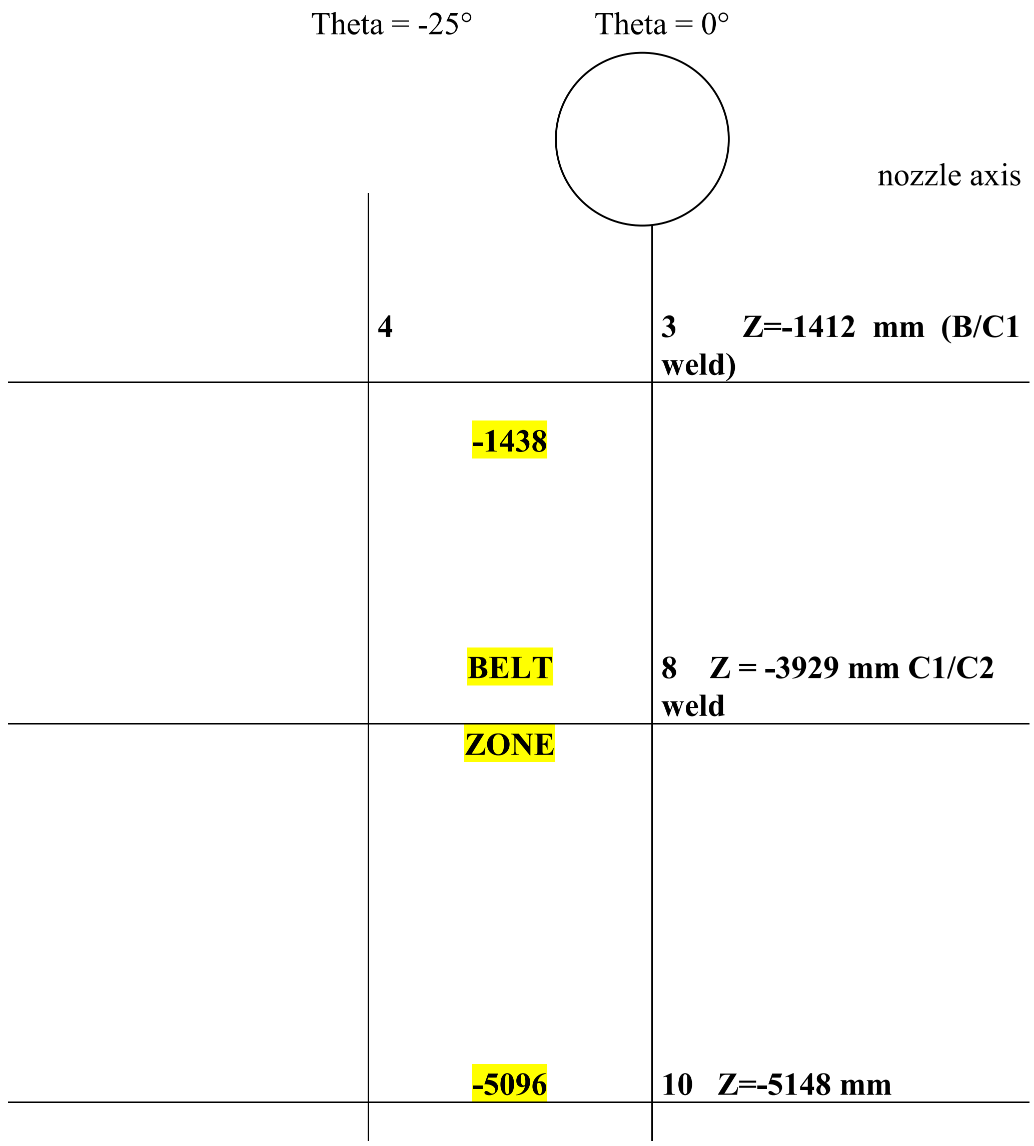


FIG. 7. Cooling gradient during 3'' SBLOCA – Comparison between CREARE correlation and STAR CD computation.



Point	Altitude (mm)	Azimuth (°)	Area
3	**-1412**	**0**	**Altitude B/C1 weld CP0 under the inlet nozzle Representative of base metal area**
4	**-1412**	**- 25**	Same as point 3 **but moved azimuthally (not under the inlet nozzle) Representative of base metal area**
8	**-3929**	**0**	C1/C2 weld Representative of weld zone
10	**-5148**	**0**	**Representative of base metal (out of core belt)**

FIG. 8 Location of the selected defects.

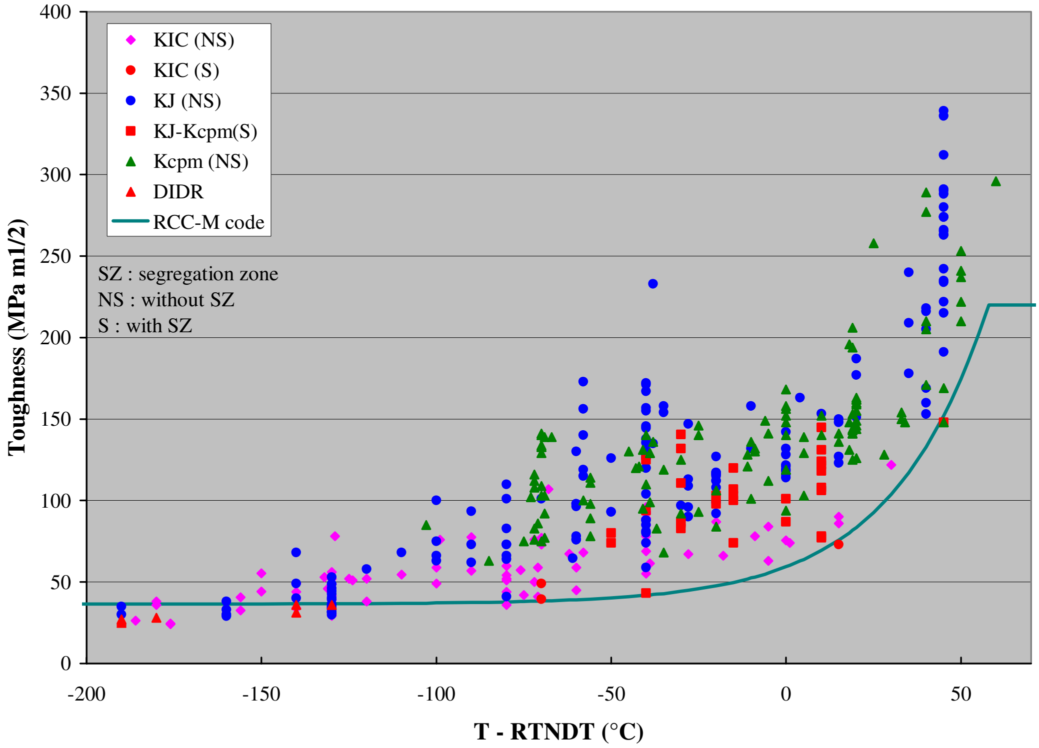


FIG. 9. Fracture toughness curve for French A508 Cl. 3 RPV steel.

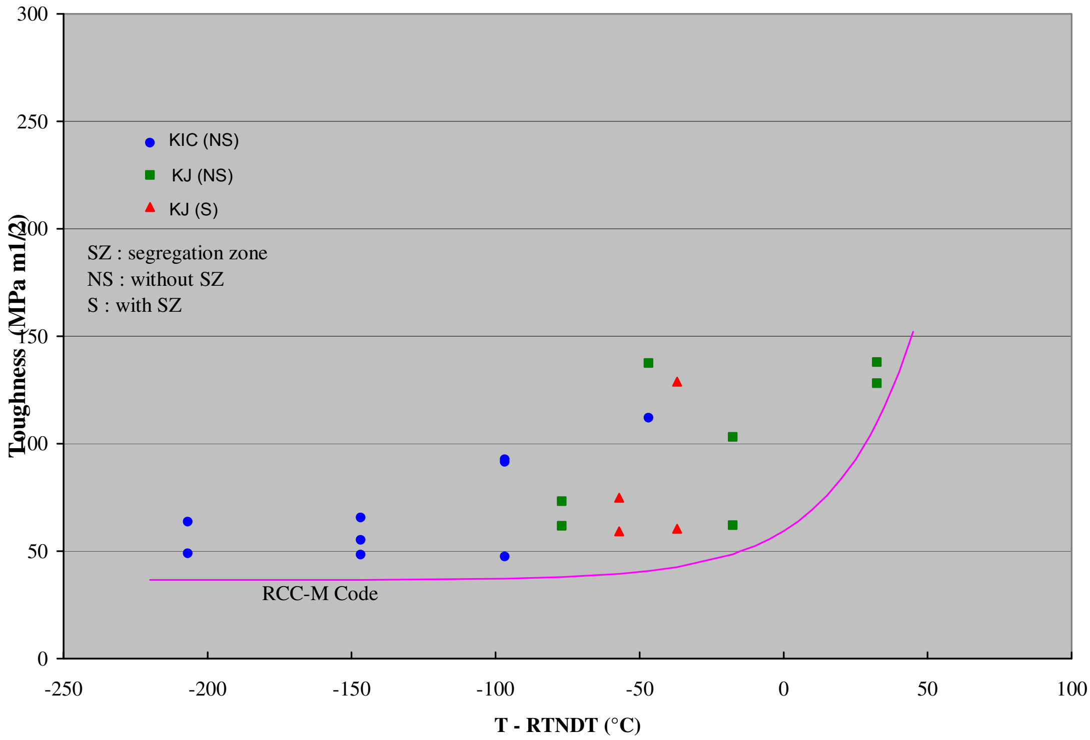


FIG. 10. Fracture toughness curve on irradiated French A508 Cl. 3 steel.

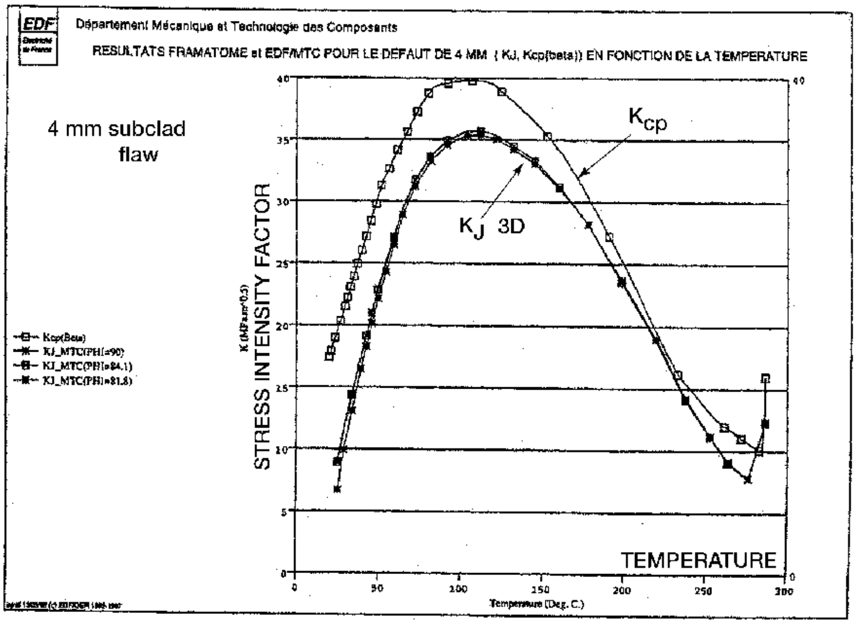


FIG. 11. Analysis of a 4 mm subclad flaw in RPV. Comparison between the 3D elastic-approach (K_J) and the simplified elastic approach (K_{cp} with correction).

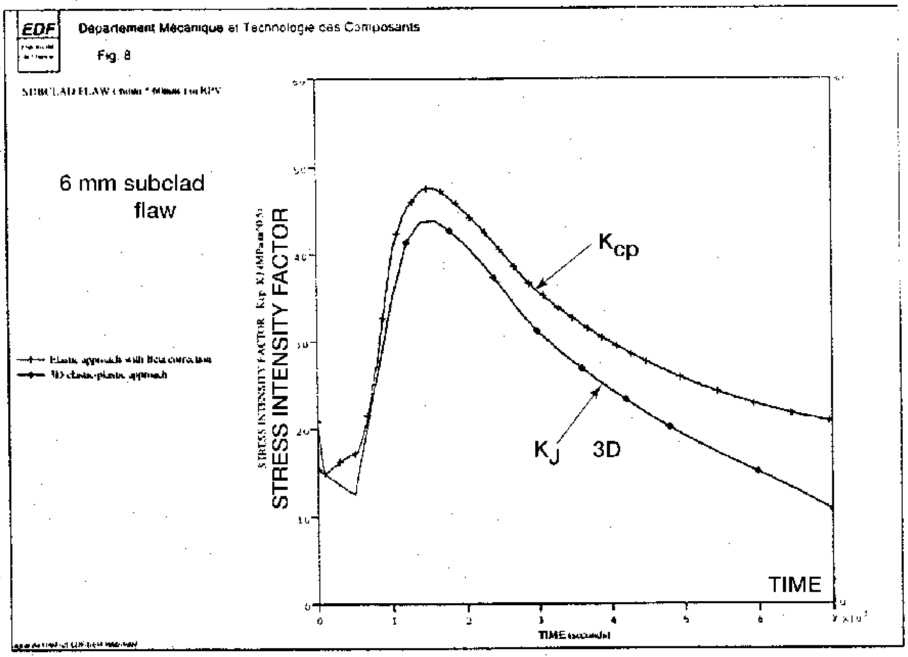


FIG. 12. Analysis of a 6 mm subclad flaw in RPV. Comparison between the 3D elastic-approach (K_J) and the simplified elastic approach (K_{cp} with correction).

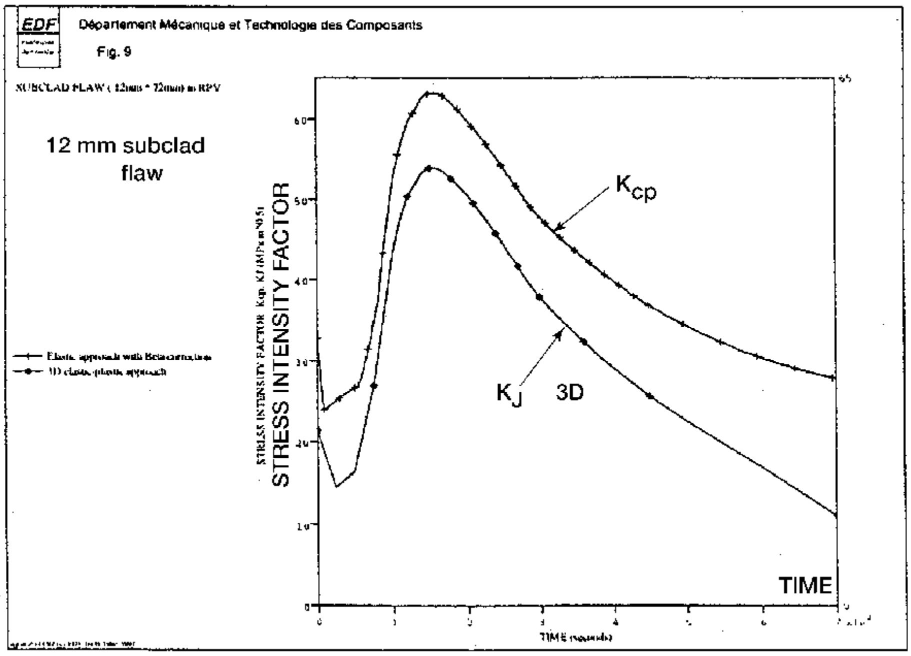


FIG. 13. Analysis of a 12 mm subclad flaw in RPV. Comparison between the 3D elastic-approach (K_J) and the simplified elastic approach (K_{cp} with correction).

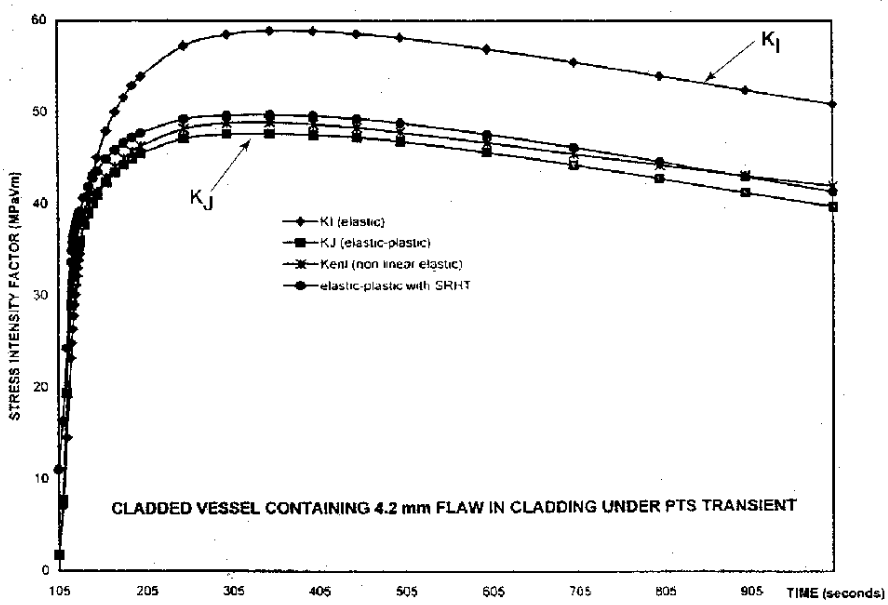


FIG. 14. Analysis of a 4.2 mm flaw located in cladding of a RPV. Comparison between the elastic approach (K_I without plasticity correction) and the elastic-plastic approach (K_J) at the crack tip in base metal.

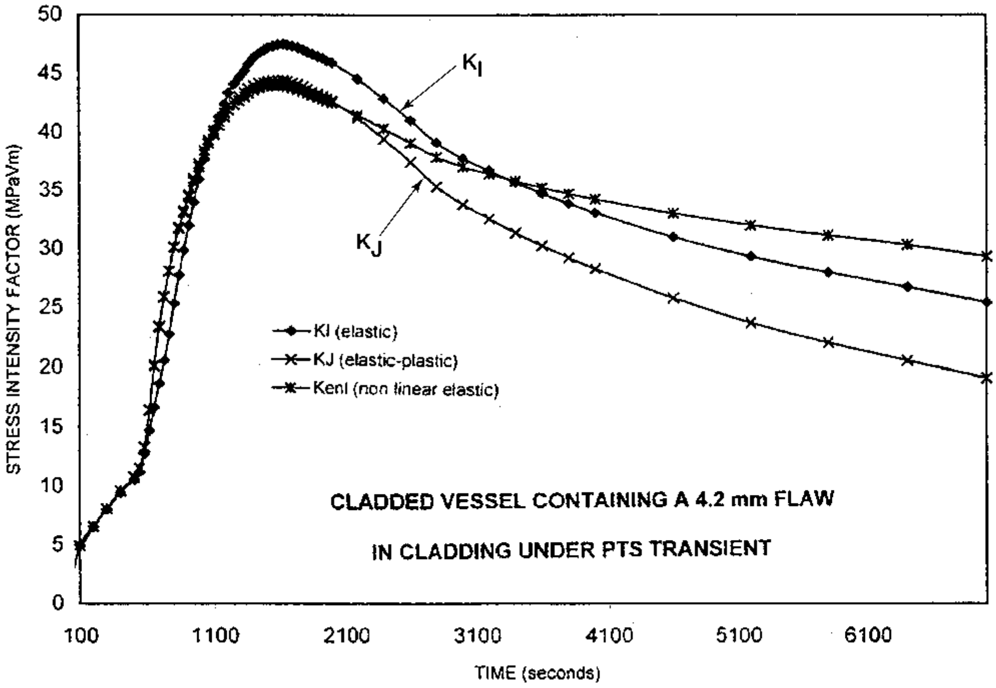


FIG. 15. Analysis of a 4.2 mm flaw located in cladding of a RPV. Comparison between the elastic approach (K_I without plasticity correction) and the non linear elastic approach (K_{ENL}) at the crack tip in base metal.

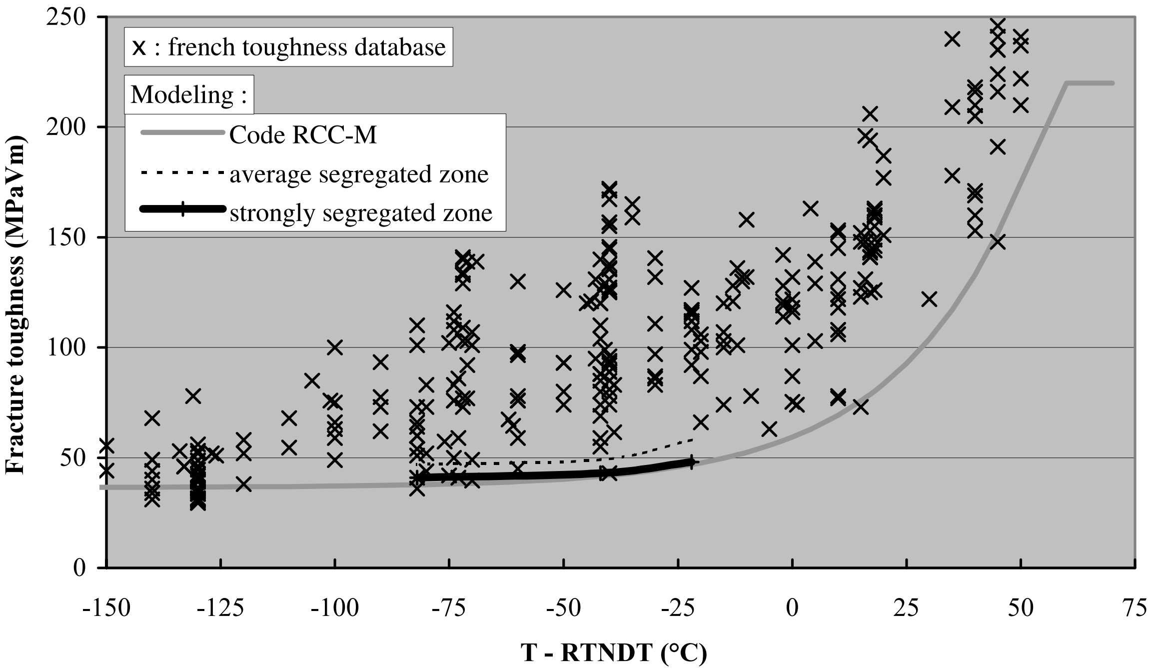


FIG. 16. Modeling of segregated zones influence on RPV toughness steel.

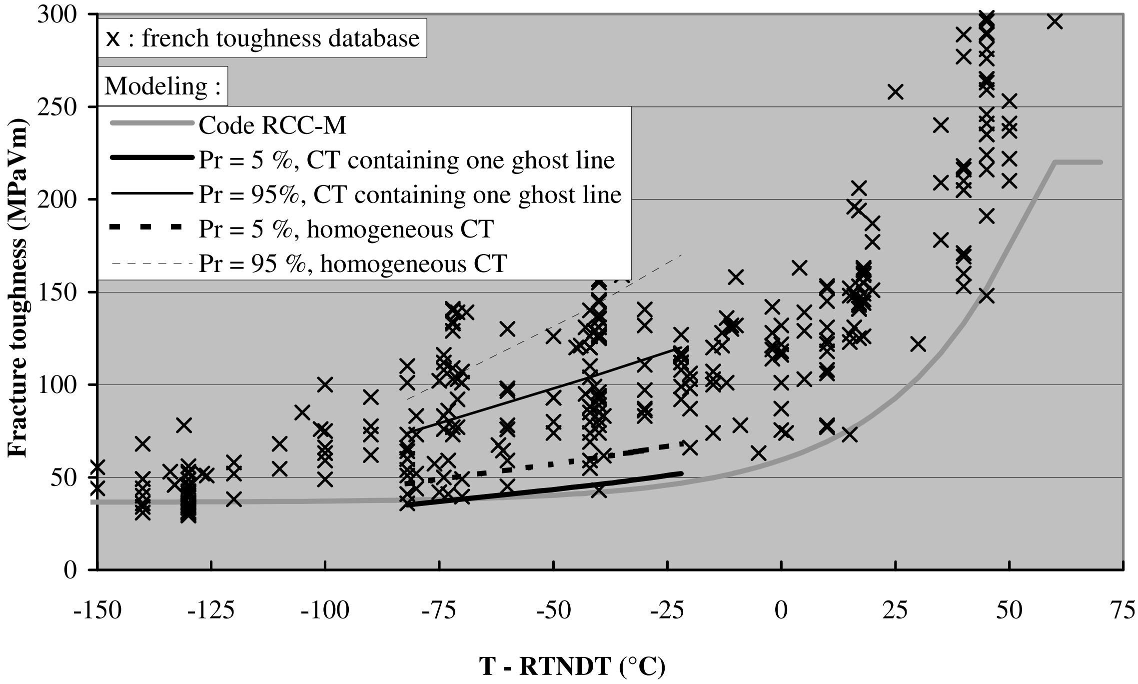


FIG. 17. Modeling of segregated zones influence on RPV toughness steel.

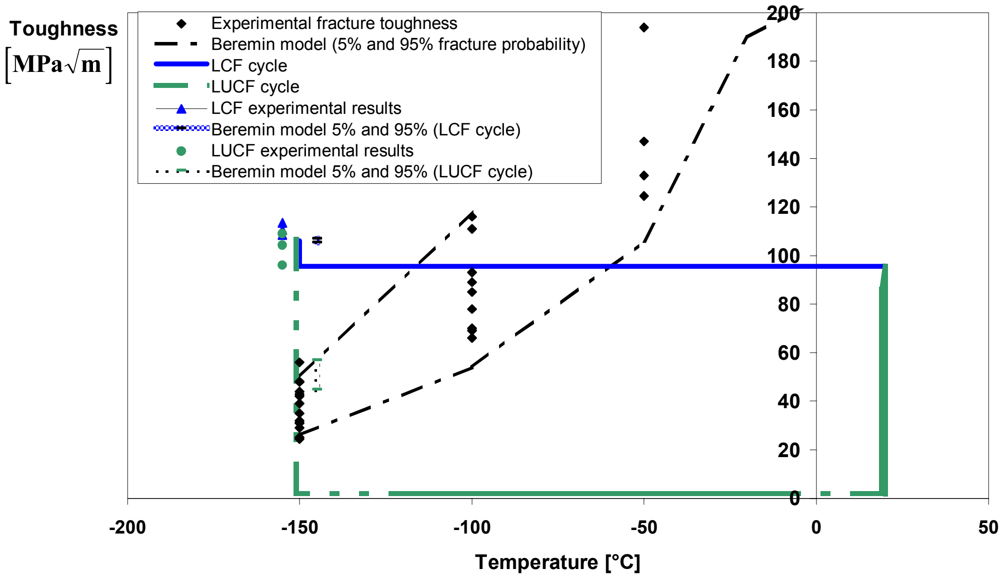


FIG. 18. WPS experiments on CT specimens on French A508 Cl3 RPV steel.

EFFECTIVE TECHNOLOGIES OF PLIM ELEMENTS APPLICATION AT RUSSIAN FEDERATION NPPs

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Abstract

Main elements of PLIM approach are discussed according to general statements of developed “PLIM Conception for Russian NPPs” adopted by RF Minatom Department of Nuclear Power and Russian Utilities. PLIM issues are prioritised with respect to different levels of decision-making: (1) Nuclear Industry/Utility, (2) NPP/NPP Unit, (3) System/Component. Interaction of generic and specific elements of PLIM technology is shown on some examples of application at NPPs. Peculiarities of PLIM application to NPP Units of different generation is discussed using practical experience for Russian RBMK NPPs based on generic PLIM elements development (regulatory base, technological base, informational base, etc.).

1. INTRODUCTION

Plant Life Management (PLIM) have been recognised as an integrated optimisation process to achieve economically effective Nuclear Power Plant (NPP) Life Cycle from the stage of pre-operation until final stage of post-operation (decommissioning). These could be possible only on the basis of profitable NPP Operational Life, which requires maintaining appropriate safety level of NPP operation. Only this part of NPP Life Cycle makes it possible to return investments and gain financing for implement activities at post-operation stage.

Here below views of technical support organisations on PLIM aspects for different levels of decision-making process are provided using some examples of NPPs with RBMK reactors.

ENES is RDIPE filial company. So services to RBMK NPPs are always provided in co-operation with RDIPE, which is Chief Design Enterprise for RBMK reactor facilities (Fig. 1) and utility. There are 11 NPP Units with RBMK-1000 currently operating in Russia and Unit 5 of Kursk NPP is in process of assembling.

Input of RMBK NPPs to electricity generation in Russia is presented at Fig. 2 according to data from Russian Utility - Concern “ROSENERGOATOM” [1].

It should be mentioned that in general economic situation in Russia currently is favourable for nuclear industry development. Perspectives of foreseen development are provided in the Utility Report [1].

2. POSITION ON PLIM PROCESS

Due to accumulated many year experience in Nuclear Industry in the role of design institute RDIPE have a broad understanding of PLIM process and treats it not only as an operational life management of reactor facility and NPP Unit, but as management presented at all stages of life cycle, including pre-operation and post-operation as well. This is first important issue taken into account in Russian PLIM concept developed by a special Co-ordinating Board [2]. This concept has been adopted in 2000 by Department of Nuclear Energy of RF Minatom and Utilities. Proclaimed approach concept is proposed by RDIPE for currently operating RBMK NPPs and Kursk NPP Unit 5.

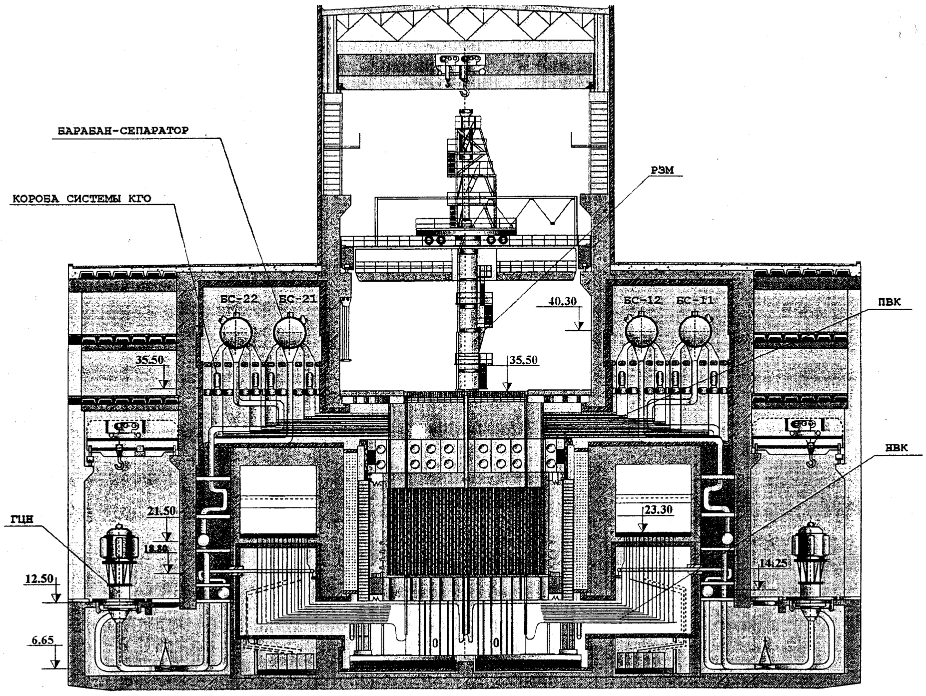

FIG. 1. Cutaway of an RBMK nuclear island.

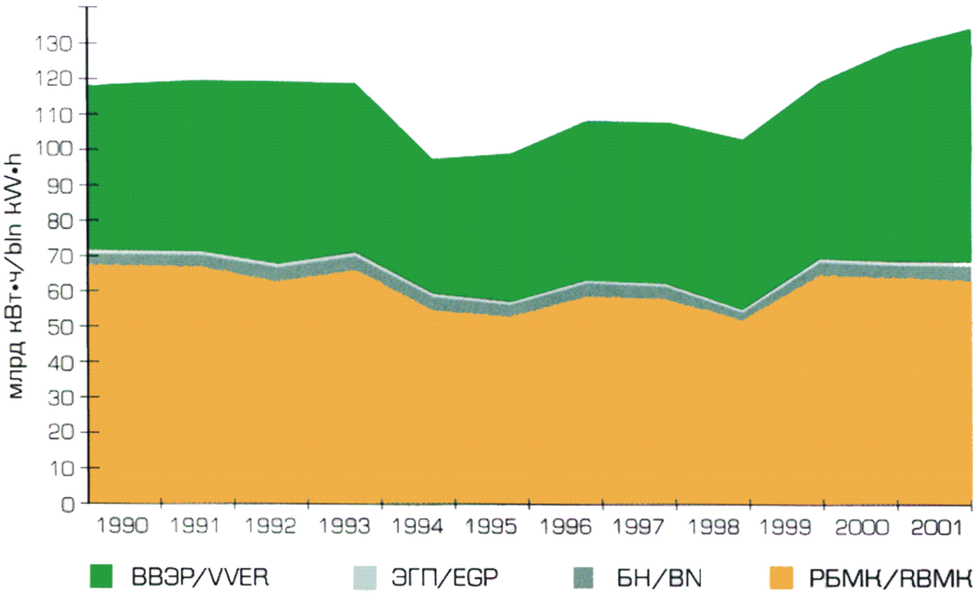

FIG. 2. Electric power production of Russian NPPs.

The next PLIM process peculiarity is that economic considerations should be the driving force for PLIM at all levels of decision making. This is very important feature, because previously PLIM activities were considered to be mostly technical area. There were number of specific technical areas, like design, manufacturing, assembling, pre-operation monitoring, putting reactor facilities to power, operation, ISI, etc., which are now being optimised via PLIM process by application of economical criteria.

PLIM issues are prioritised according to peculiarities at different levels of decision-making process:

- Nuclear Industry/Utility,
- NPP/NPP Unit,
- NPP System/Component,

which will be illustrated here below.

3. TECHNOLOGICAL BASES OF PLIM

3.1. Nuclear Industry/Utility Level

For the Nuclear Industry/Utility level it seems reasonable to focus on:

- aspects of strategic economic planning to understand trends for future,
- aspects of new nuclear technologies development to replace existing reactor facilities in future,
- aspects of development generic bases for PLIM: in addition to program of normative and methodological bases development presented (by Prof. Bugaenko at this Symposium [3]) technological and information bases should be mentioned also.

Technical support organisations of Russian NPPs are able to provide effective technological basis for pre-operation, operation and decommissioning stages (including repair, in-service inspection, monitoring, water chemistry, etc.). These typical technological processes once developed and qualified then are applicable at different NPP sites.

Development of unified infrastructure for information basis of PLIM also could be effectively managed at the Ministry and/or Utility levels with further application of unified information technologies at specific NPP Units for data collection and processing.

3.2. NPP/NPP Unit Level

For the level of NPP or NPP Unit it seems reasonable to focus on:

- safety improvement based on:
- refurbishment/upgrading,
- main circulation circuit (MCC) integrity improvement,
- economically effective operation based on optimized technologies (including PLIM Data Bases),
- transparency for public to support acceptance of nuclear power in NPP region.

First priority issue is safety level improvement, which comes through periodic safety assessment as a basis for operation license approval by GOSATOMNADZOR, by refurbishment/upgrading and MCC integrity improvement.

Obsolescence effects are being reduced by system improvement or replacement. For example, modern complex Control and Protection Systems was installed at Unit 1 of Kursk NPP in 2002.

There is a schedule for production of Safety Analysis Reports (SAR) for 1st generation of RBMK NPPs and two of them have been submitted already to Regulatory Body by the Utility. Kursk NPP 1 SAR has passed GOSATOMNADZOR expertise and now is being submitted to international expert review. Team for SAR development includes RDIPE, Kurchatov Institute, AEP, other leading enterprises and also Utility&NPP specialists.

In the framework of SAR materials development environmental effect influence on component life assessment is taken into account [4].

One of options, which should be mentioned separately, is integrity improvement of large-scale Main Circulation Circuit components. In the framework of this process technical analysis based on LBB or Break Preclusion criteria [5] is developed and next is implementation of state-of-the-art Automatic Leak Detection Systems, improved ISI procedures and means, etc.

3.3. NPP System/Component Level

For the level of systems/components we should mentioned risk-based inspection (RBI) and reability centred maintenance (RCM) as effective tools for implementation of maintenance strategies using available technological basis (repair, ISI, condition monitoring, water chemistry, mitigation, replacement, etc.).

For critical RBMK-1000 reactor components (including reactor graphite stack, metal structures, fuel channels, large scale MCC components, civil structures, etc.) effective technologies to support their Life Management are being permanently improved or developed. Some examples of these are given here below.

It needs to be mentioned here that RBMK reactors faced the problem of replacement of pressure tubes after 16-20 years of operation. For this purpose effective technologies for re-tubing have been developed for Leningrad NPP Unit 1 and then have been repeatedly improved [6].

Next example deals with development of technology for repair of RBMK-1000 graphite stack from group of “non-replaceable” to group of “partially replaceable/repairable”. These results in less technical limitations in managing graphite stack life and increase number of available economical scenarios of specific NPP Unit Life Management.

Other example from RBMK practice is Life Management process of austenitic Dia300 piping (Fig. 3) at RBMK NPPs, which experienced IGSCC damages at many RMBK NPP Units. Technologies (welding, WC management, ISI, Leak Detection, etc.) are now being applied on-site as a part of systematic Life Management process [7]. These generic technologies have been already developed, qualified and approved by GOSATOMNADZOR during 1998-2001 years and some mitigation technologies are being currently adapted to RBMK conditions (overlay welding, stress improvement, etc.). So cost-effective application of these technologies at specific NPP Units should be supported by appropriate economical analysis.

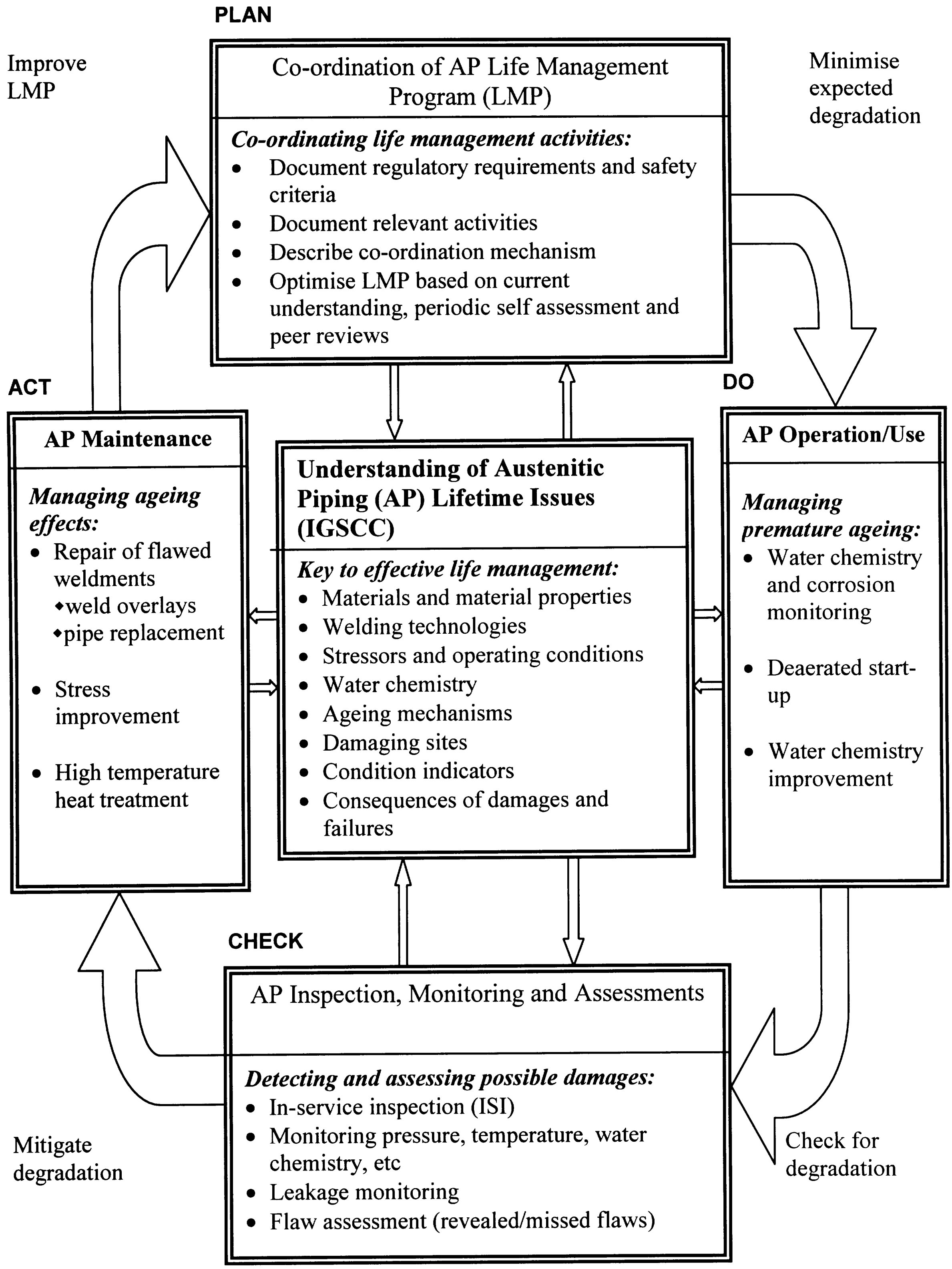

FIG. 3. RBMK austenitic piping (AP) life management (LM) programme (utilising the systematic ageing management process proposed by IAEA [6].

4. CONCLUSION

It needs to be emphasised that accelerated improvement and/or development of generic support bases for PLIM is of primary importance for Nuclear Industry in Russia. This includes the following type bases for PLIM:

- regulatory;
- methodological;
- technological;
- information.

Availability of unified PLIM bases provides favourable opportunities for economically effective NPP Life Management.

REFERENCES

[1] 2001 Russian Nuclear Power Plants (Rosenrgoatom), Moscow, 2002.

[2] S.E.Bugaenko / PLIM Conception for Russian NPPs // PLIM+PLEX'99. Plant Management and Plant Life Extension in Nuclear Facilities.-Madrid, Spain.-2-5 Nov. 1999.-Conf. Proc.-1999.-Nucl. Eng. Int., Wilmington Business Publ.-P. 39-50.

[3] S.E.Bugaenko / PLIM Support R&D Programme of Russian Minatom / Int. Symp. on Nuclear Power Plant Life Management – Budapest, Hungary – 4-8 Nov. 2002 – Book of Extended Synopses – 2002, P.22.

[4] S.V.Evropin, E.N.Sinitsyn, V.M.Filatov, A.V.Tashkinov / Strength Calculation Aspects of Vessels and Pipes of Power Reactor Facilities During Operation // The 7th Int. Conf. on Material Issues in Design, Manufacturing and Operation of NPP Equipment, St. Petersburg, 17-21 June 2002.

[5] RF Minatom. Guideline of Application of Leak Before Break Concept to NPP Piping (R-TPR-01-99) RD-95 10547-99 (approved by GAN for application at NPPs), 1999.

[6] L.Belianin, V.Lebedev, E.Adamov, Yu.Cherkashov et al. Safety of NPPs with Channel Type Reactors. Core Upgrading. – M.: Energoatomizdat, 1997. – 256 p.

[7] Implementation and Review of a Nuclear Power Plant Ageing Management Programme / Saf. Reps. Sers. N 15, IAEA, Vienna, April 1999.

Session 4
TECHNOLOGICAL AND OPERATIONAL ASPECTS OF PLIM

Sub-Session 4.2

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CFD-TOOL FOR ASSESSMENT OF THE REACTOR PRESSURE VESSEL INTEGRITY IN PRESSURE THERMAL SHOCK CONDITIONS FOR LIFETIME EVALUATION. QUALIFICATIONS PHASIS AND THERMAL-HYDRAULIC STUDY OF A SAFETY INJECTION IN A THREE LOOP PWR PLANT.

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Abstract

For Reactor Pressure Vessel (RPV) assessment and lifetime evaluation of the nuclear plants, in Pressurised Thermal Shock (PTS) conditions, special thermal-hydraulic studies of a safety injection in the down comer are applied by French Utility to demonstrate the integrity and the safety of the vessel, in various conditions of loadings.

This paper explains the Research and Development program started at E.D.F about the cooling phenomena of a PWR vessel after a Pressurised Thermal Shock. The numerical results presented here are obtained with the thermalhydraulic Finite Element (FE) code N3S coupled with the thermal-solid code SYRTHES to take into account the conjugate heat transfer on the cooling of the vessel. In a first time, we explain the numerical program concerning the qualification task of this CFD-Tool for Safety Injection studies. For that, we have investigated several configurations related to an injection of cold water particularly in cold leg and in a down comer. Two experiment test cases have been studied and we present a comparison between experiment and numerical results in terms of temperature field. In a second time and for the reactor thermal hydraulic study, the geometry used represents a three loop PWR. In this calculation, the simulated finite element mesh takes into account as much as possible the exact geometry of the lower plenum. The configuration investigated is related to the injection of cold water in the vessel during an accidental transients and its impact on the solid part formed by cladding and base metal. Numerical results are given in terms of temperature field in the cold legs and in the down comer.

The main purpose is to provide the accurate distribution of the fluid temperature in the down comer and the heat transfer coefficients on the inner RPV surface for a future structural integrity assessment and the corresponding RPV safety margins.

INTRODUCTION

Within the frame of the plant lifetime extension project, the assessment of the French 900 MWe (3-loops) series Reactor Pressure Vessel (RPV) integrity has been performed according to a specific approach derived from the codified fast fracture analysis [RCCM code, ZG appendix], based on a selection of shallow sub clad flaws and a set of loading transients. This analysis has shown that the most severe loading conditions are given by the small break loss of coolant accidents (2" and 3" SBLOCA) due to the pressurised injection of cold water into the downcomer of the RPV. For these PTS transients (third category), thermal hydraulics analysis has to be carried out in order to better adjust the distribution of fluid temperature in the downcomer and the heat transfer coefficients on the inner RPV surface.

This paper reports the thermalhydraulic study performed with the Finite Element (FE) code N3S coupled with the Thermal-Solid code SYRTHES to take into account the conjugate heat transfer on the cooling of the vessel during LOCA. In this scenario, the generated thermal loading induces a strong loading on the vessel (cold thermal shock). A refined thermalhydraulic analysis constitutes a new approach, which can lead to a better knowledge of the thermalhydraulic conditions on the cooling of the vessel. The geometry used and described consists of a 900 MWe PWR plant of CP0 (first French PWR RPV with a circular RPV thermal shield) plant. This plant has been chosen because it is the one which has the vessels whose Nil Ductility Transition Reference Temperatures (RT_{NDT}) are the highest and therefore whose margin seeking is the priority. This computation takes place between the CATHARE system computation (which gives the main initial conditions of the primary circuit) and structural integrity assessment regarding the risk of brittle failure. The 900 MWe CP0 Pressurised Water Reactor plant is presented and after that, the numerical simulation which concerns the transport and the mixing of cold water (Safety Injection) in a hot environment through the cold legs and the downcomer. The interest involves the fluid-solid coupling to realize a computation in a solid part of the vessel formed by cladding ant base metal. Globally, we analyze the thermal behaviour of a PWR vessel in its fluid part as well as in its solid. The thermal coupling fluid-structure leads to master the physical characteristics present in the whole vessel. Their assessment is a major asset in comparison with those coming from system codes and from previous methods based on experimental correlations for increasingly significant margins.

In a same time, EDF R&D engaged a qualification phasis in order to show the N3S CFD code capabilities to simulate the main physical characteristics present in this kind of scenarii. For that, two geometries used and described for the first one, of a EPRI-Creare 1/5 scale mixing facility, which has a geometry type of PWR that includes one cold leg, a planar downcomer with or without an internal thermal shield and an High Pressure Injection and for the second one, a Upper Plenum Test Facilities (UPTF) which has a geometry type of 1300 MWe pressure water reactor at scale 1. The paper reports this qualification task with the thermalhydraulic studies performed with the Finite Element (FE) code N3S coupled, if necessary (depends on the qualification test case), with the Thermal-Solid code SYRTHES to take into account the conjugate heat transfer. Different experiment results are available with different probes located mainly in the cold leg (stalks are used to analyse the thermal layers) but also just at the beginning of the downcomer, area directly concerned by the thermal loading interesting in the safety studies. In fact, some probes are located respectively on the external wall (representative of the vessel wall) and on the internal wall (representative of the core barrel). Globally, we analyse the thermal behaviour of a cold leg and a downcomer in its fluid part. Concerning the UPTF test case, the thermal coupling fluid-structure leads to master the physical characteristics present in the geometry. The fluid behaviour can be assessed and in a same time, the numerical tools can be compared with the experiments results.

GEOMETRY OF A 900 MW PWR PLANT (CPO SERIES)

The modeling of the 900 MW PWR takes into account as much as possible the accurate geometry of the vessel with its different components such as the circular thermal shield, the centering pins and the columns and plates instrumentation. In addition, the geometrical parameters used to describe the safety injection phenomena were modeled. Calculation domain is composed of : three primary pump volutes, three cold legs with their SIS (Safety Injection System) or ECC (Emergency Core Cooling system) and Accumulator injection nozzles respectively 6 and 12 inches, the vessel with its components mentioned above and the hot legs which are only represented by the space required by them in the down comer. Figure 1 illustrates the domain, which was simulated in the study (skin mesh).

The fluid-solid coupling is done from the injection nozzles up to the RPV centering pins without considering the thermal exchanges along the core barrel, the thermal shield and the structures of the lower plenum. Thus, a potential heating source is not taken into account. A 360° modeling was used to assess the pertinence of a complete vessel model relating to cold legs interaction. The fluid mesh includes about 840 000 velocity nodes and 550 000 tetrahedral elements. The solid mesh includes about 320 000 nodes and 220 000 tetrahedral elements.

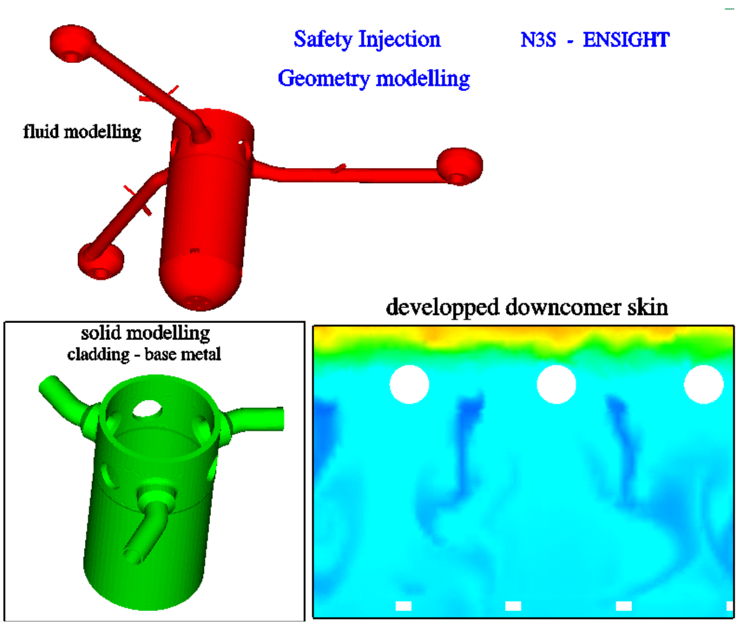


FIG. 1. Fluid and solid modeling.

THE N3S AND SYRTHES NUMERICAL TOOLS

The N3S code

The FEM code N3S has been developed by the Research Development Division of E.D.F for thermalhydraulic studies in nuclear engineering design, taking advantage of our experience acquired on 3D finite difference codes. N3S's development started in 1982 with, as main feature the use of unstructured meshes for complex geometries modeling. After intensive testing required by EDF Quality Assurance policy, it has become available for use as a general purpose tool which has been applied successfully to a wide variety of incompressible laminar or turbulent flows with or without heat transfer.

For our study, we used the N3S package, which contains four main steps: the pre-processor, the solver's interface, the N3S solver itself and the post-processor.

For the pre-processing task, the I-DEAS™ software and most exactly Object Modeling (OM) and Finite Element Modeling (FEM) have been used for the geometry definition (mapped meshing) and SIMAIL™ (free meshing) for the unstructured mesh generation.

The solver's interface PREN3S checks the mesh and prescribes the boundary conditions.

N3S solves the Reynolds averaged Navier-Stokes equations for an unsteady incompressible or compressible flow, with a constant time step or a space or time variable time step. Different turbulence models can be used but in our study, we use the standard k-ε two equations turbulence model, since a high level of turbulence is observed in this configuration. For buoyancy effects driven flows with small temperature differences, the momentum equation is normally coupled with the energy equation using the Boussinesq approximation. To take into account more important thermal effects we consider here the Navier-Stokes equations in which density depends on temperature : $\rho = \rho(T)$. The time discretization is based on a fractional step method. At each time step the code solves successively:

- an advection step, for the non-linear convection terms of the Navier-Stokes equation and the k and ε equations, by using the characteristics method,
- a diffusion step on scalar variables,
- a generalised Stokes problem for the velocity and the pressure, solved by a Chorin-Teman algorithm (preconditioned conjugate gradient [P/G/N]).

Assuming a logarithmic velocity profile, we use wall functions on the boundaries to compute the friction shear stress at each time step. Additionally, a zero flux condition is given for k which leads to the definition of a second velocity scale used to prescribe ε at the wall.

For the post-processing task, the ENSIGHT software was used for FE visualizations in fluid dynamics. ENSIGHT allows the visualization of 2D and 3D fields. All specific functionalities were developed according to EDF specifications.

The solid code SYRTHES

The solid code uses a finite element technique to solve a general heat equation where all properties can be time, space or temperature dependent.

$$\rho c_p \frac{\partial T}{\partial t} = \mathrm{div}(k_s \mathrm{grad} T) + \Theta_v$$

T being the temperature, t the time, Θ_v a volumetric source or sink, ρ and c_p respectively the density and the specific heat, k_s (symmetric matrix when the material is anisotropic and a scalar multiplied by the identity matrix when the material is isotropic) designates the conductive behaviour of the medium.
For optimization reasons, only two kinds of elements have been retained (6 nodes triangle in 2D, 10 nodes tetrahedra in 3D). Indeed, for these elements, elementary matrices can be derived analytically, and therefore, at low cost. . Like the fluid codes, SYRTHES has been checked thoroughly against experimental and analytical test cases proving that it provides very accurate solutions in transient problems.

Recently, the possibility to take into account radiation from wall to wall (non participating medium) has been included. The methodology followed is based on a radiosity technique. It is therefore possible to take into account problems where conduction, convection and radiation are present simultaneously.

SAFETY INJECTION IN A THREE LOOP PWR PLANT (SBLOCA, 3”)

The results corresponding to a thermal loading computed by N3S and SYRTHES codes are presented in terms of skin temperature (internal side of the vessel) and solid temperature (cladding and base metal). The calculation was carried out for a time during which the minimum mechanical margin factor was reached. This justifies the stopping of the thermalhydraulic computation. The calculated transient time is 2800s. The description and analysis of the results obtained on the fluid and the solid are evaluated on this thermal transient. This paper presents the results leading us to better master the physical phenomena at the key points. This approach enables us not only to take into account the fluid mixing during its transport along cold legs and downcomer but also to obtain the thermal gradients on the wall of the vessel. In a first step, the fluid physical phenomena are presented in cold legs, in the injection nozzles, near of the connection zone between cold legs and the downcomer and into the downcomer for several elevations. After that, the solid physical results obtained from the Syrthes solid code are illustrated by temperature maps located on the portion of the geometry where the coupling fluid-structure is modeled (on the skin solid mesh and in the cladding and the base metal).

Initial and Boundary Conditions

Initial conditions

At the beginning of the computation, the initial time corresponds to the Loss of Natural Circulation. The initial conditions are issued from a full system computation CATHARE (reactor, primary and secondary loops,…). At this time, we consider the pressure at 75 bar. The Table 1 sums up the initializations used in the three cold legs and in the down comer for fluid and solid parts.

Table 1 :Initial conditions for the three cold legs

	Between nozzles and vessel (fluid and solid)			upstream nozzles (fluide and solid)	downcomer up to centering pins (fluid and solid)	lower plenum up to the core inlet
	cl1	**cl2**	**cl3**	**cl1-cl2-cl3**		
Temp (°c)	233.1	241.6	258.8	282	257.6	265.7

The fluid and solid (cladding and base metal) physical properties are taken at this pressure value and depend on the temperature evolution during the transient. Figure 2 illustrates temperature conditions.

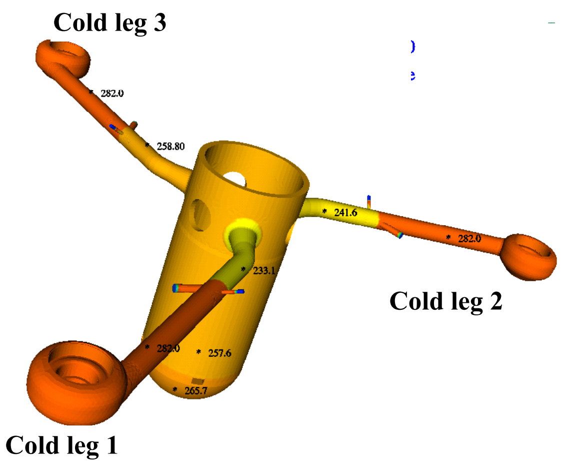
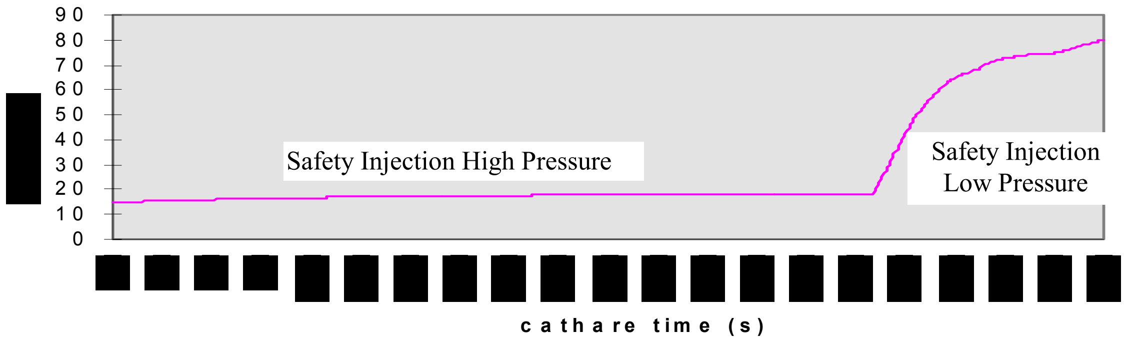


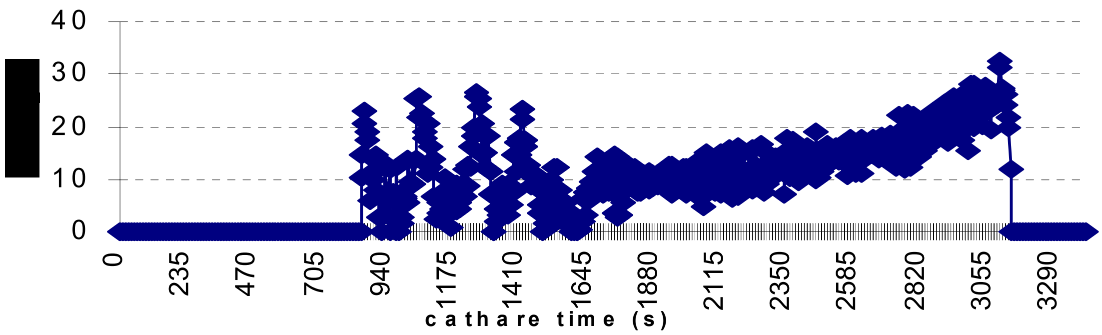
FIG. 2. Simulation area and Initial temperatures Conditions.

Boundary conditions

Figure 3 a) and b) illustrate the main boundary conditions in terms of mass flow rate in the computation. The study takes into account the simulation of the Safety Injection by IS pumps (High pressure and Low pressure), the Safety Injection by Accumulators and the primary pump joint leaks. The cold temperature is also taken into account by these injections (9°C for IS pump and 20°C for IS accumulators). The temperature injected by the primary pump joints is taken as that flowing out from the water box of the steam generator and is hotter. During the transient, its value remains about 190°C.



a)



b)

FIG. 3a). 6'' injection (RIS), b) 12'' injection (accumulators.

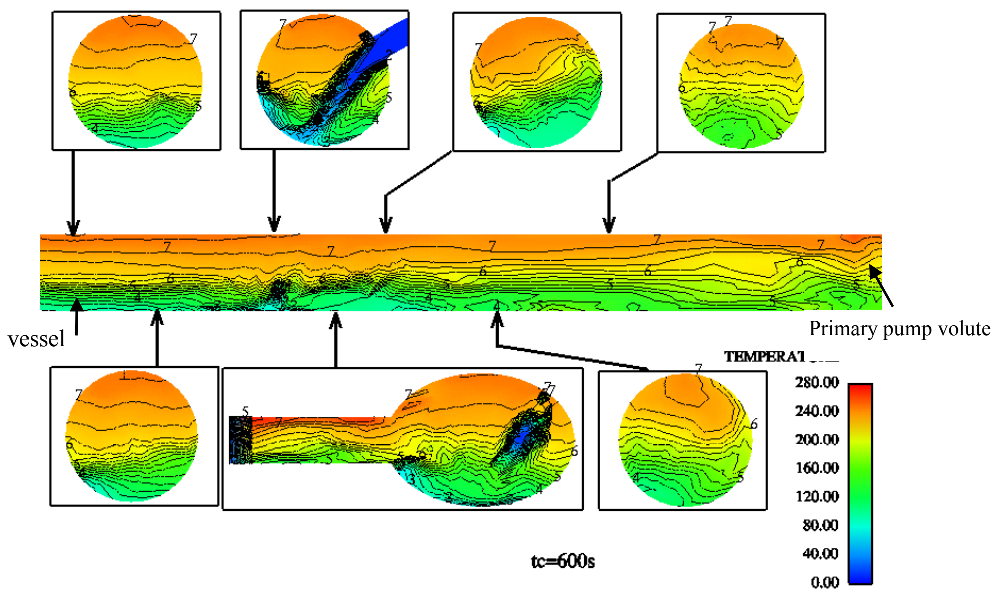


FIG. 4. Vertical section in cold leg and median section in the 2 nozzles.

Fluid results

Just after the loss of natural circulation, the water injected in the three cold legs causes the appearance of thermal layers induced by density effects in these cold legs. These effects are generated by the temperature difference between the water injected in the nozzles (respectively 9°C for the 6 inches and 20°C for the 12 inches injection) and that of the primary circuit (averaged at 250°C at the beginning of the transient). During this thermal transient, the physical phenomena present in the cold legs have a real influence on the fluid behaviour in the downcomer. Figure 4 shows the thermal layers in one of three cold legs. This figure illustrates the complexity of the physical phenomena near the nozzles.

We can see different sections of the cold leg upstream and downstream of the two nozzles and the fluid flow behaviour on a median plan of these two injection nozzles. On Figure 5, a light swirling effect appears as we go towards the primary pump volute.

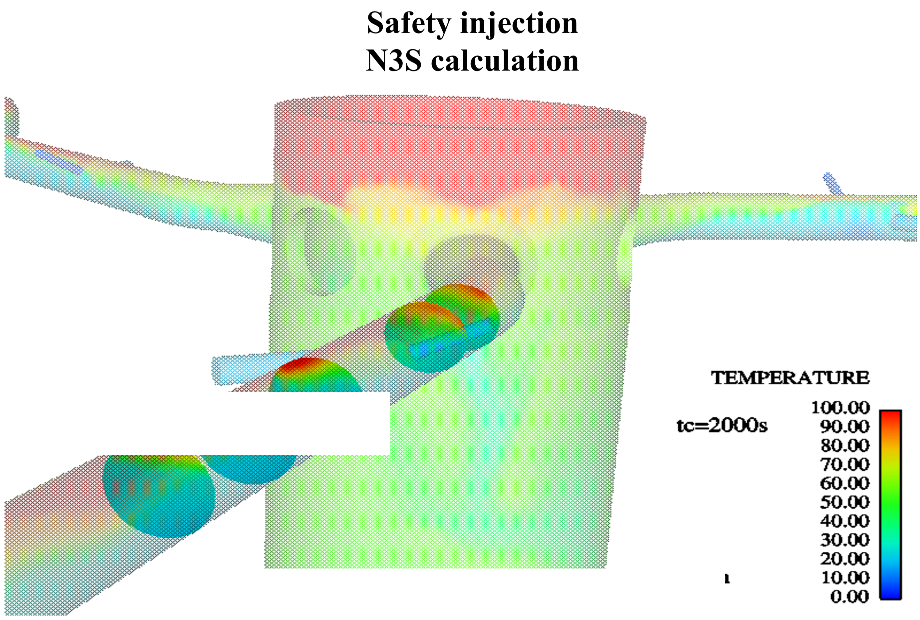


FIG. 5. General view of one cold leg.

Figures 6 and 7 illustrate the profile and the path of the flow when it arrives to the connection zone (RPV/cold legs fillet) and in the downcomer. The main result in this area concerns the fluid flow separation. This behaviour strongly depends on the fluid characteristics (velocity, thermal layers...), the recirculation phenomena in the downcomer and the fillet radius. During the transient, the fluid fluctuates on both side of the thermal shield. The cold fluid flows sometimes along the RPV and sometimes along the core barrel. So, the thermal shock for the vessel will be less hard. These results can be observed again on the figure 3 (on the developed downcomer) where the plume fluctuation appears.

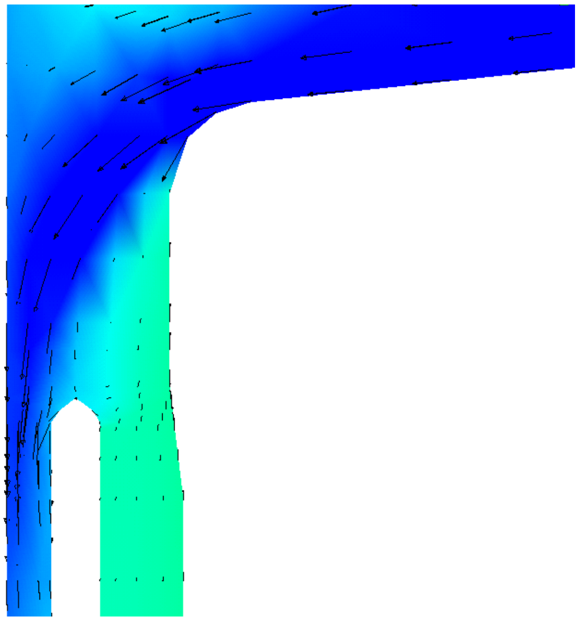


FIG. 6. Fluid behaviour in the fluid flow separation region.

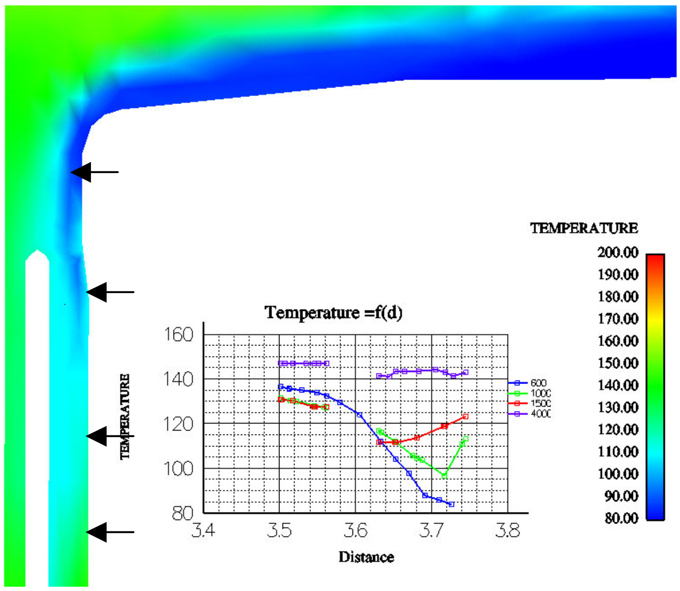


FIG. 7. Fluid behaviour in the RPV/cold legs fillet.

Solid results

Figure 8 illustrates the solid temperature obtained from the coupled computation realised with the Syrthes software. On this figure, the temperature is given on the vessel skin and inside the vessel walls. Temperature is calculated on the seven first millimeters (cladding) and through the base metal wall. A strong radial thermal gradient is present in the solid and so, in these conditions, we will have, a dumping thermal shock in the metal. At this time, the temperature heterogeneity on the total internal area is about 15°C. At the end of the transient, the internal temperature of the vessel has become homogeneous.

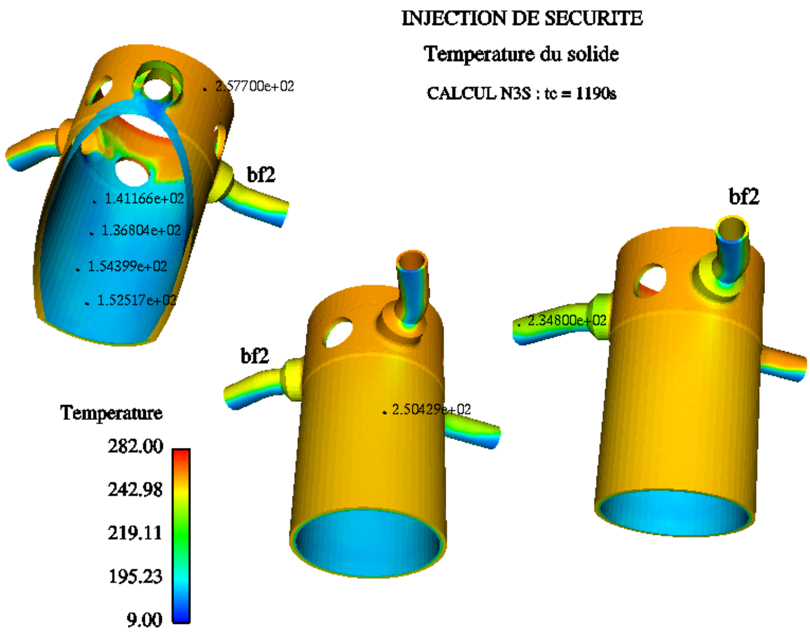


FIG. 8. Temperature in cladding and base metal.

THE NUMERICAL TOOLS QUALIFICATION

Analysis of the main physical phenomena and purpose of the qualification task

Knowledge of fluid mixing is of interest to analysis of plant overcooling transients where coolant water is injected at high pressure and hence pressurised thermal shock at the reactor vessel is theoretically possible. This paper presents the results leading us to better master the physical phenomena at the key points. This approach enables us to take into account the fluid mixing during its transport along cold legs and downcomer and confirm the CFD code capabilities to simulate correctly the thermal layers and the fluid separation in the downcomer.

As final result, N3S has to be assessed as being adapted to simulate PTS transients and for that, he has to be able to reproduce the physical phenomena displayed in the first reactor study. Two experiment facilities have been chosen and in each case, we injected cold water into stagnant primary water with a zero mass flow rate through the loop. These experiment test cases have regard for reactor conditions (turbulence level, density effects…). For a best examination of data, thermocouples or probes have been placed in key locations and more particularly along and on all height of the cold leg and at the beginning of the downcomer, under the cold leg, close to the vessel wall and the core barrel.

In a first step, a qualitative analysis shows for each test case, the fluid physical phenomena in cold legs and into the downcomer for the N3S code. After that, the experiment and numerical results are presented in term of temperature values given by the different thermocouples. For this quantitative analysis, we also took advantage of the Star Cd[1] numerical results availability to compare two numerical results. The whole data are used for the experiment and numerical comparison.

[1] Star CD is a CFD code used by FRAMATOME in the frame of the plant lifetime extension project. Thus, Star CD is qualified for this kind of physical configuration and has its own qualification file.

The EPRI/CREARE geometry

Figure 9 shows the geometry of the EPRI/CREARE (NUREG/CR-3822, EPRI) facility at 1/5 scale.

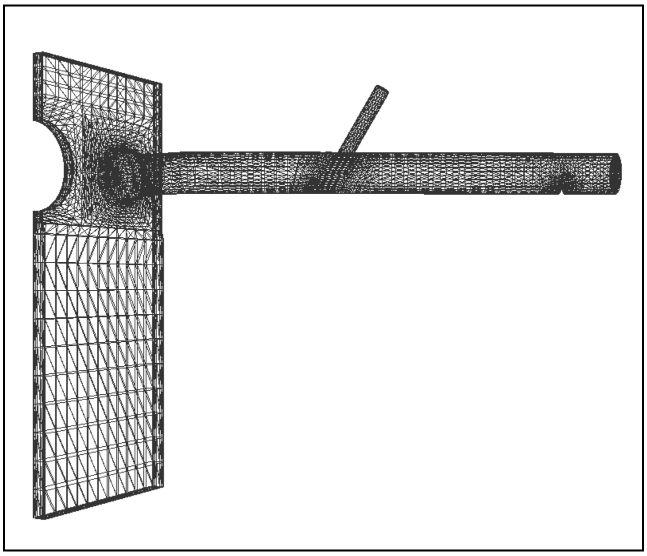


FIG. 9. Creare geometry.

The EPRI/CREARE facility consists in a transparent acrylic facility designed for operation near atmospheric pressure. The downcomer is represented by a planar section having width and height comparable to a 90° sector of a reactor downcomer. The downcomer includes some internal geometries such as the thermal shield for one configuration and a part of the hot leg which represents in fact the space required by the hot leg in a RPV geometry. The vessel wall is flat and a thermal shield is represented below the cold leg nozzle and spans the full width of the 90° downcomer section. In the experiment and computation comparison, we used the CREARE geometry with and without thermal shield. The High Pressure Injection is modelled on the cold leg with a 60° angle nozzle. Experiment data are available end for our comparison, they are issued mainly from thermocouples located in the cold leg and in the downcomer close to the vessel wall and the core barrel.

Upper Plenum Test Facility (UPTF)

The UPTF (P.A. Weiss and al., 1986) is a full-scale simulation of the primary system of a four-loop 1300 MWe pressurised water reactor. The test vessel upper plenum internals, downcomer and primary coolant piping were replicas of the reference plant. In our simulation, we didn't take into account the core and the geometry is modelled from the primary pump up to the core entering. The dimensions of the UPTF test vessel are identical to the reactor vessel of the reference PWR except for wall thickness. The inner structures are modelled by an equivalent volume necessary to take into account the thermal coupling between the fluid and the solid. A nozzle located on the cold leg represents the Emergency Core Cooling System. Figure 10 shows the geometry of the UPTF test case at one scale.

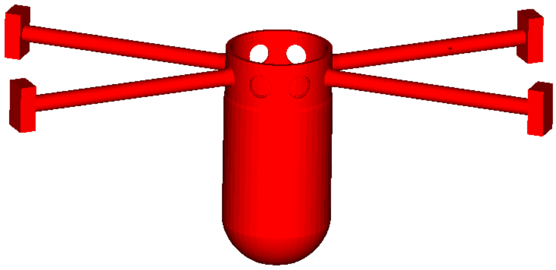


FIG. 10. UPTF geometry.

Experiment data are given and are issued from thermocouple located in the cold leg and in the downcomer close to the wall vessel and the core barrel.

For UPTF test case, additional information is given in term of solid physical result obtained from the coupling code N3S and Syrthes solid code. The results are illustrated by temperature map located on the portion of the geometry where the coupling fluid-structure is modelled.

Initial and Boundary Conditions for the two test cases.

Creare and UPTF Initial conditions

At the beginning of the computation, we consider the pressure at 1 bar for Creare and 18 bar for UPTF. The table below sums up the initialisation used into the two geometries in term of temperature. In both cases, the considered value of the mass flow rate is zero and the fluid physical properties are taken at the initial pressure.

	Creare Mix 2	Creare Mix3	UPTF Test 1
Temperature	65,5°C	66,6°C	190°C

Creare and UPTF Boundary conditions

The studies take into account the simulation of the Safety Injection by HPI (High Pressure Injection). The cold temperature is taken into account by these injections. The table below sums up the boundary conditions used in the three simulations in term of temperature and velocity injection.

	Creare Mix 2	Creare Mix 3	UPTF Test 1
Temperature	17.8°C	16.7°C	27°C
velocity	0.125m/s	0.058m/s	1.87m/s

Fluid results in the cold leg and into the downcomer

Whatever the geometry, the main physical phenomena are the same in the cold leg and in the area of the downcomer the most interesting thermal from a loading point of view. Just after the beginning of the cold water injection, this cold water induces the development of thermal layers due to density effects. These effects are generated by the difference between the water injection in the nozzle (respectively about 17°C for the two CREARE test cases and 27°C for UPTF) and the initial temperature (66°C in CREARE and 190°C in UPTF). During this thermal transient, the physical phenomena present in the cold legs have a real influence on the fluid behaviour in the downcomer. Figure 11a et 11b show the development of the thermal layers in the cold legs for the two test cases.

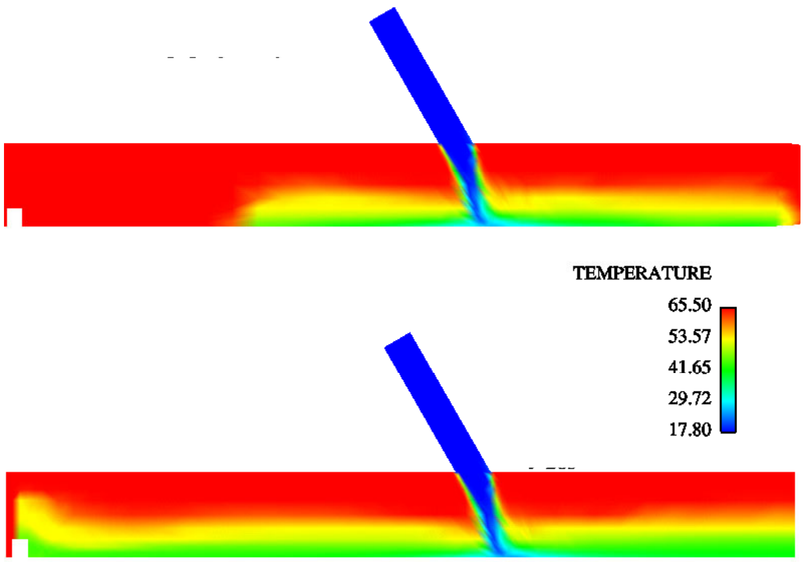


FIG. 11a. Thermal layers for CREARE test case.

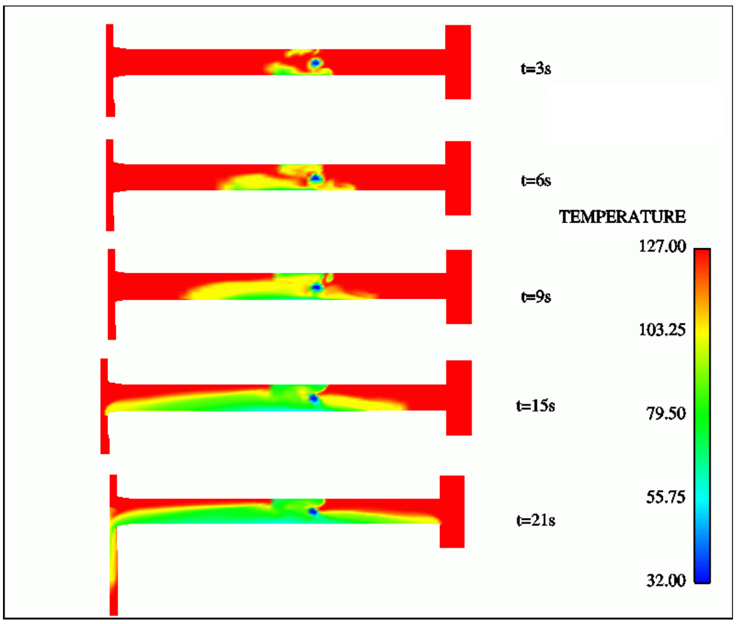


FIG. 11b. Thermal layers for UPTF test case.

In fact, a cold stream flows along the bottom of the cold leg and a hot stream flows along the top of the cold leg countercurrent to the cold stream. Figure 12 shows the cold stream from the cold leg penetrated into the downcomer as a plume, which fluctuates during the transient (UPTF computation).

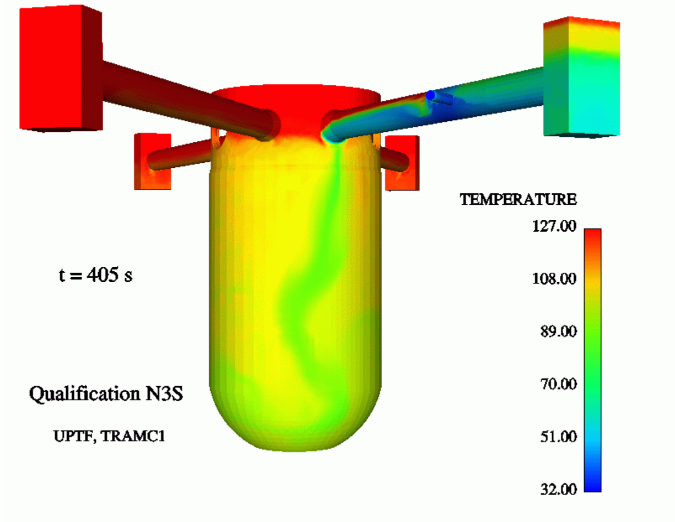


FIG. 12. General view of flow behaviour.

Temperature measurements in the downcomer indicate that, due to the mixing in the cold leg and at the cold leg/downcomer interface, the temperature of the fluid is of course significantly higher than the temperature injection. For the UPTF facility, figure 13 illustrates the profile and the path followed by the flow when it arrives to the connection zone (RPV/cold leg fillet) and in the downcomer.

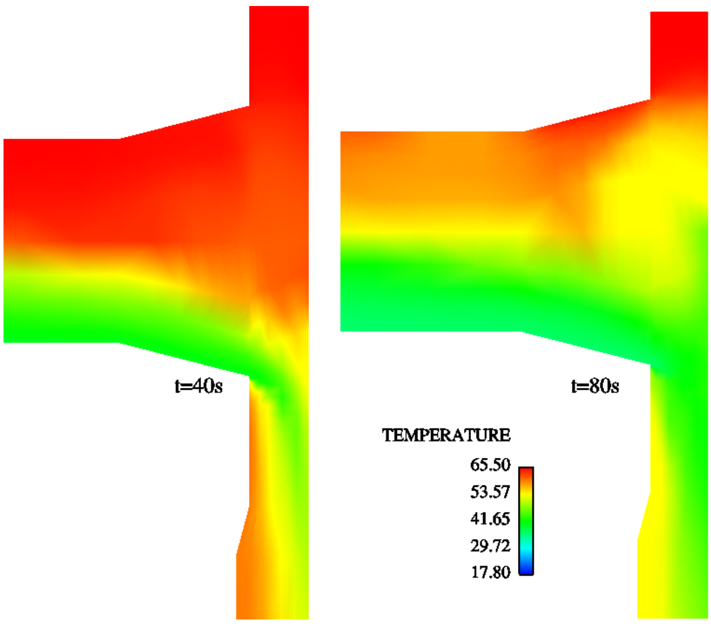


FIG. 13. Fluid flow in the connection zone (inlet vessel nozzle).

The main result in this area concerns the fluid flow separation (seen also in the CREARE test case). This behaviour strongly depends on the fluid characteristics (velocity, thermal layers...), the recirculation phenomena in the down comer and the fillet radius.

During the transient, we find again the same physical phenomenon displayed in the reactor study. The fluid fluctuates on both sides of the down comer walls, sometimes along the RPV and sometimes along the core barrel. So, about thermal loading of the RPV point of view the thermal shock for the vessel can be considered as less hard. For the qualification task and quantification point of view, different measurements have been realised in the cold leg and into the downcomer. Figure 14a et 14b illustrate the location of the probes and stalks on the two experiment facilities.

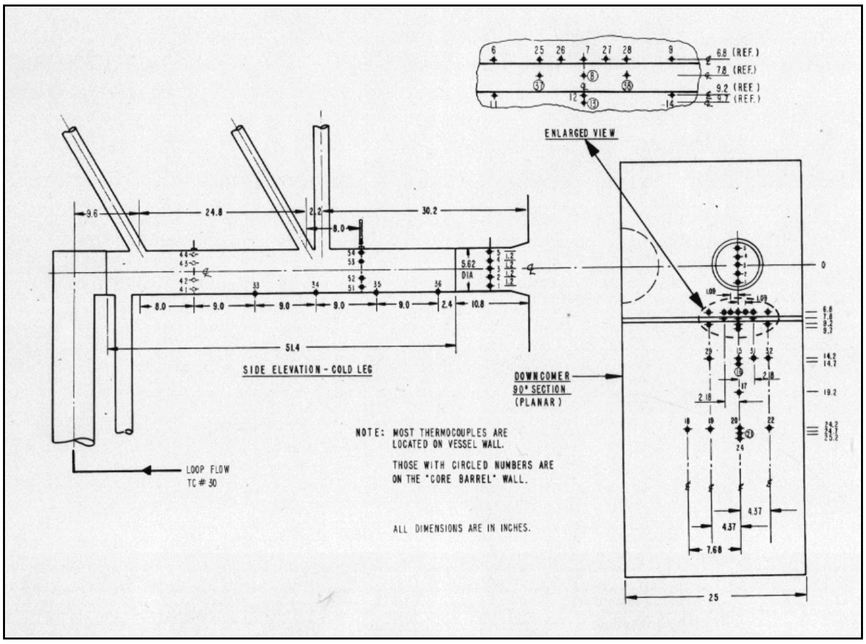


FIG. 14a. CREARE Instrumentation.

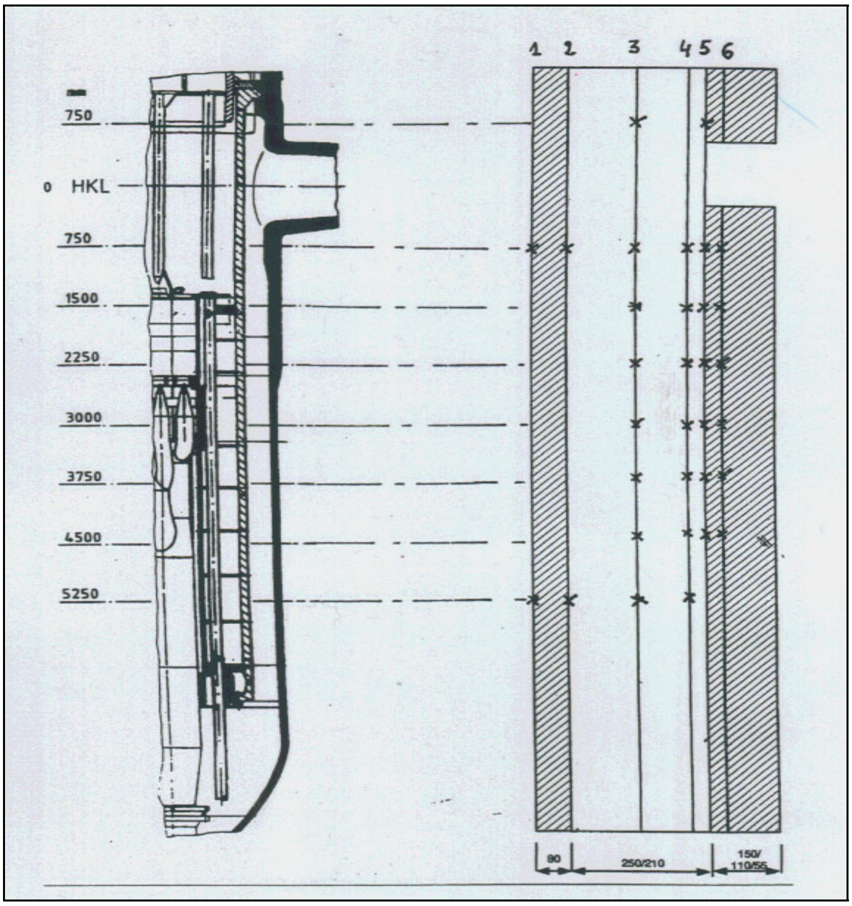


FIG. 14b. UPTF Test 1 instrumentation.

Figure 15 shows the good agreement between experiment and numerical results in term of temperature measurements (UPTF computation). The thermal layers are correctly predicted with a good transition between the hot and the cold stream (probe 33). Sometimes it was difficult to quantify the experiment values. It is why, someone are illustrated by upper and lower values.

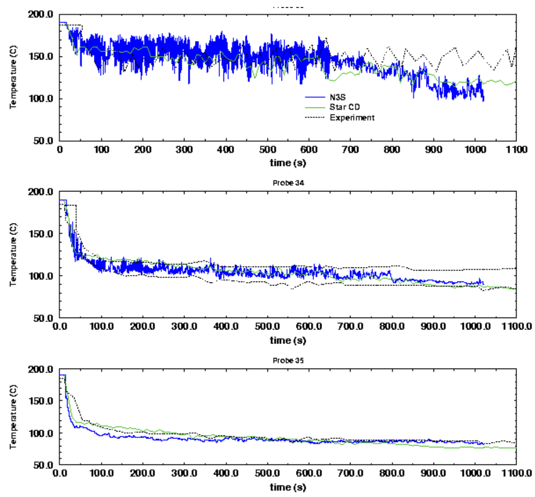


FIG. 15. Temperature stalk in cold leg.

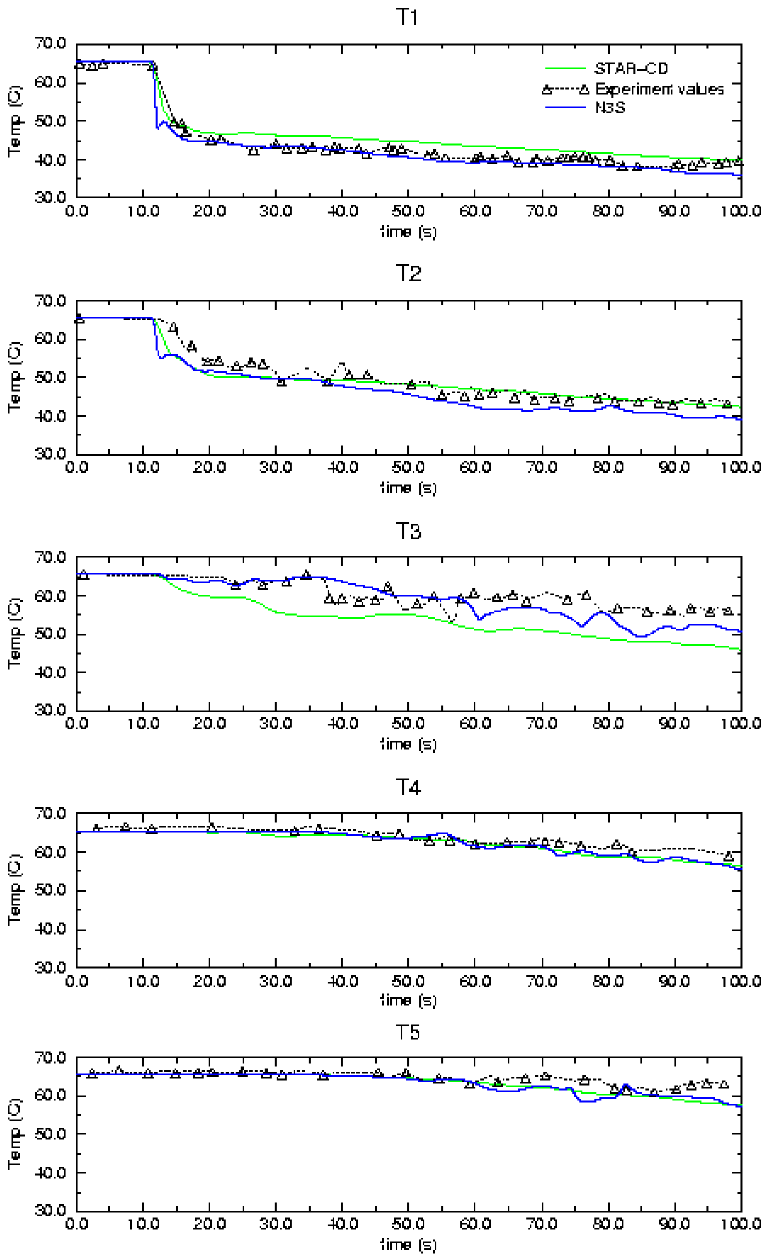


FIG. 16. Stalk with probes 1-2-3-4-5.

Figure 16 illustrates another stalk located just before the connection with the down comer (CREARE simulation). The results show a good agreement between numerical results and experiment values. The thermal layers are well predicted as well as in cold part than the hot part of the leg.

Figure 17 illustrates a last result in cold leg (Creare test without thermal shield) with the probes 33 and 34 which are located on invert. This result shows the good behaviour of the numerical tools. A first result into the downcomer leads to study more particularly the area just at the vertical below the connection zone. In fact, whatever the test case taken into account (Creare with or without thermal shield), two probes are systematically located here and there of the downcomer (on the internal and external wall). The probe 8 (core barrel) value is colder than the probe 7 and so, this result indicates the good prediction of the flow separation at this location.

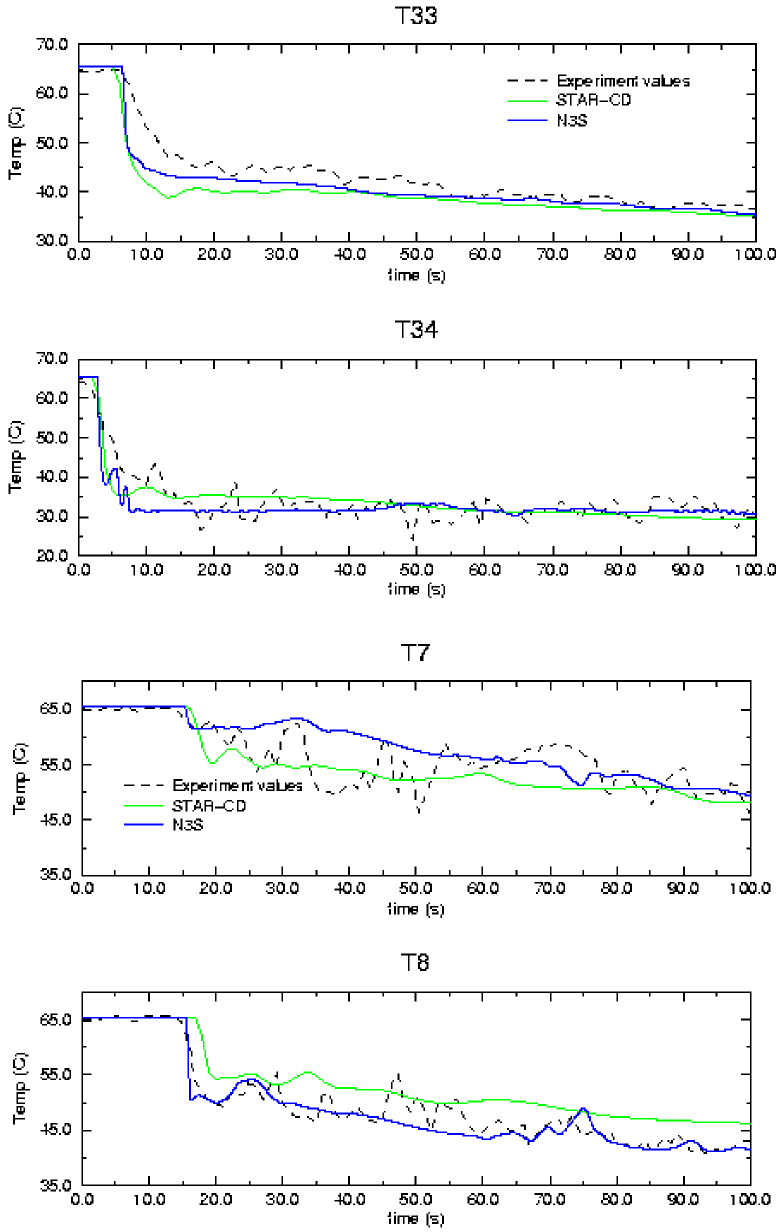


FIG. 17. Probes (Creare) 33-34-7-8.

Figure 18 illustrates more particularly the code capabilities to simulate the flow separation at this level. For that, the Creare test case with thermal shield used.

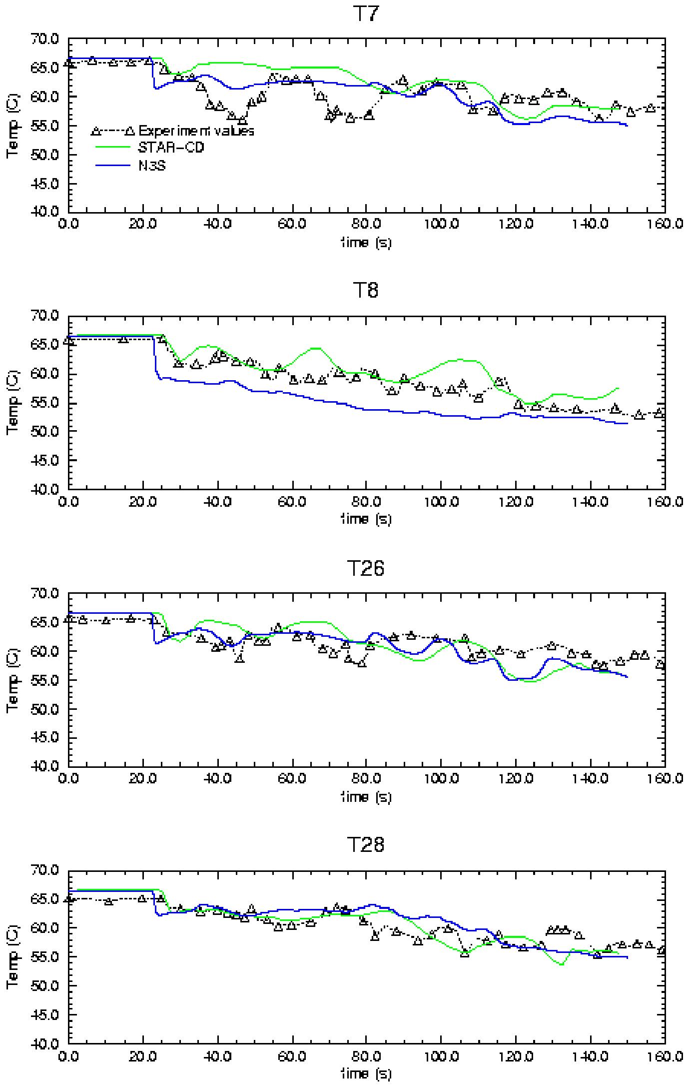


FIG. 18. Probes 7-8-26-28.

The probe number 8 located on the internal wall (core barrel) is colder than the number 7 located on the external wall (RPV) and as the figure 9, we have a good prediction of the flow separation. On this figure, we can see the unstable characteristics of the flow, which depends notably of its evolution in the cold leg. Some parameters such as the thickness or the height of the thermal layer can lead to this behaviour.

This figure also shows (probes 26-28), the good prediction of the two numerical tools about experiment values located here and there of the vertical plume issued from the cold leg. This result is important because it shows the fluctuations of the plume into the down comer and shows the code capabilities to reproduce this physical phenomenon.

In a first time, the plume flows vertically because, there are much buoyancy effects, but progressively the plume will fluctuate with the decreasing of these effects.

The same analysis can be realised on the UPTF test case. Figure 19 shows the good agreement between the different tools used for this qualification task. On this figure, experiment results are illustrated by lower and upper values except probe 6.

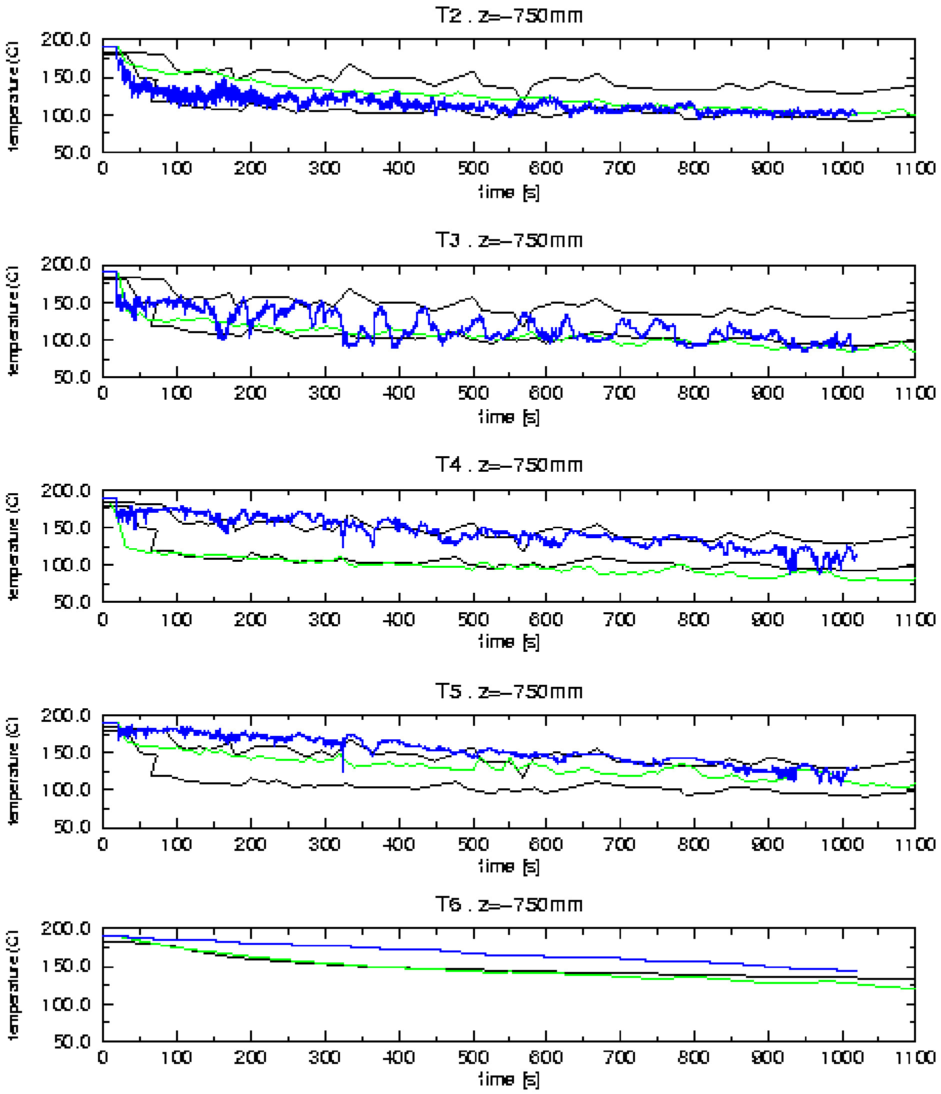


FIG. 19. Probes (UPTF) 2-3-4-5-6.

CONCLUSION

The 3D thermalhydraulic computation of the primary small break LOCA has lead us to improve our knowledge on the behaviour of the vessel during a PTS event. The purpose of this computation was to obtain the temperature distribution on the wall of the vessel during the transient. For that, we have used the coupling between the « N3S » CFD code and the « SYRTHES » solid thermal code to take into account the fluid structure interactions inside the vessel. The flow separation of the fluid along the RPV wall is a important physical phenomena which decreases the severity of the thermalhydraulic transient on the vessel. This numerical approach is a success because we fit the real transient with this modeling. On the other hand, the qualification task has lead us to assess the N3S capabilities to represent the physical phenomena linked to the cold water injection in a hot environment. The results of two test cases were available with the numerical results associated and issued from the task qualification of the Star CD code realised by FRAMATOME ANP in this context. The different results show a good agreement.

The main results are located in the connection zone into the cold leg and in the down comer just below the cold leg. The thermal layers and the temperature levels are correctly predicted and the flow separation of the fluid along the RPV wall, which is an important physical phenomenon, has been in a prominent position. This qualification shows the capacity of the EDF R&D numerical tool to simulate Safety Injection studies.

Note that the same qualification tasks are now going on at EDF R&D division using the new generation CFD tool Code_Saturne, coupled to SYRTHES code.

REFERENCES

[1] Martin A. and al.. « Assessment of the pressure vessel integrity in PTS conditions. Thermal hydraulic study of a safety injection in a three loop PWR plant with the Finite Element code N3S ». 2000 ASME PVP Seattle, Washington

[2] Martin A. and al.. « CFD tool for thermalhydraulic Pressurised Thermal Shock analysis. Qualification of the Finite Element Code N3S ». 2001 ASME PVP Atlanta, Georgia

[3] Martin A. and Bugat S... « Assessment of the pressure vessel integrity in PTS conditions. Thermal hydraulic and Thermal Mechanical study of a safety injection in a three loop PWR plant ». 2002 ASME PVP Vancouver, Canada

[4] Martin A. and Bellet S. " CFD tool for thermalhydraulic Pressurised Thermal Shock analysis. Qualification of the Code_Saturne". 2002 ASME PVP Vancouver, Canada

[5] NUREG/NP-3822 LA-10132-MS. SOLA-PTS : A Transient, Three-Dimensional Algorithm for Fluid-Thermal Mixing and Wall Heat Transfer in Complex Geometries.

[6] EPRI/NP-2312 Project 347-1 Creare Incorporated. Fluid and thermal Mixing in a Model of cold leg and downcomer with loop flow.

[7] EPRI - Transient Cooldown in a Model of Cold leg and downcomer. NP 3118 - Research Project 2122-3

[8] Fourteenth Water Reactor Safety Research Information Meeting. October, 1986, Gaithersburg, Maryland

[9] P.A. Weiss, R. J. Hertlein - Kraftwerk Union AG, Erlangen

[10] UPTF Test Results First 3 Separate Effect Tests

RESEARCH PROGRAMS ON AGEING OF REACTOR STRUCTURAL MATERIALS AT JAPAN ATOMIC ENERGY RESEARCH INSTITUTE

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Abstract

At the Japan Atomic Energy Research Institute (JAERI), the evaluation of aging degradation of structural materials related to neutron irradiation has been investigated for structures and components which are important from the safety standpoint and/or are difficult in replacing and repairing, e.g. reactor pressure vessel (RPV) and core internals. RPV aging studies are targeted to provide the new basis for the structural integrity analysis and neutron irradiation embrittlement evaluation in the long-term operation. The master curve approach has been applied to determining fracture toughness using Charpy size specimens available from a surveillance capsule. For more accurate prediction of neutron irradiation, the effect of γ-ray irradiation on the embrittlement and intergranular embrittlement have been examined. Also, we have developed the PFM analysis code named PASCAL which has new original functions such as the fracture criterion of R6 method to evaluate a conditional probability of crack initiation and failure of a pressure vessel under transient conditions. On the other hand, Irradiation assisted stress corrosion cracking (IASCC) is the main issue for the aging of core internal materials, and at JAERI the fundamental research on IASCC has been managed in this decade. In 2000, a national project for IASCC technological development had begun in which JAERI is engaged in the BWR related testing that includes the irradiation and post-irradiation examinations. Since the synergies of radiation and stress/water cannot be reproduced outside reactor core, the in-pile IASCC testing is one of the key experiments to understand IASCC behavior. A high temperature water loop facility was, therefore, installed at the Japan Materials Testing Reactor (JMTR) to carry out the in-core testing. IASCC is considerably complicated by the synergies, so that the numerical modeling and simulation techniques is substantially worthy of being developed for understanding and predicting IASCC.

1. INTRODUCTION

A few of light water reactors (LWRs) in Japan began to operate in the early 1970’s and have reached a 30-year operation. The LWRs will be the mainstream of nuclear power generation for a considerably long period. It is, therefore, necessary for the aged reactors to further operate when many aging phenomena are readily manageable. The domestic countermeasures designated to cope with the aging of LWRs have been already taken expecting a 60-year operation. Particularly important is to maintain the integrity of the safety-related structures and components subjected to aging during long-term operation. Reactor vessel, internal core components, primary coolant piping, recirculation pump, electrical cables, containment structures and concrete structures have been chosen as safety-related ones necessary for the management of aging in a report by the Ministry of Economy, Trade and Industry (METI). The relevant research and development have been comprehensively carried out at national research organizations and nuclear energy industry under the domestic coordination of METI.

The Japan Atomic Energy Research Institute (JAERI) is a governmental organization and a Japanese core institute of comprehensive nuclear energy research, which was established in 1956. At JAERI, the evaluation of aging degradation related to neutron irradiation has been investigated for structures and components which are important from the safety standpoint and/or are difficult in replacing and repairing, e.g. reactor pressure vessel (RPV), internal core structures in accordance with the national five-year safety research program issued by the Nuclear Safety Commission.

For RPV, the following research has been conducted to precisely evaluate the integrity during the long-term operation;

- Fracture toughness evaluation using the Master Curve approach for Japanese RPV steels,
- Gamma-ray effect on irradiation embrittlement,
- Grain boundary embrittlement caused by irradiation-induced phosphorus segregation

We have also developed a probabilistic fracture mechanics (PFM) code PASCAL (PFM Analysis of Structural Components in Aged LWR) to evaluate the RPV integrity under pressurized thermal hock (PTS) transients. The results of the above research will be used for a revision or an new code and/or standard related to aging management, and also incorporated in PASCAL to analyze the integrity and reliability of aged RPV.

For core internals, the main research item is irradiation assisted stress corrosion cracking (IASCC), which is caused by the synergistic effects of neutron and gamma-ray radiation, corrosion by high temperature water, and the residual and/or applied stresses. The following research and development programs have been conducted;

- Fundamental study for mechanistic understanding of IASCC,
- Development of IASCC evaluation technology for BWR plants based on post-irradiation IASCC test data as a part of METI's national project,
- In-pile IASCC test for evaluation of the synergistic effects of irradiation, corrosion and stress, and
- Computational studies on the materials database and numerical simulation.

For the programs of IASCC studies, a new high temperature water loop facility was designed and installed at the Japan Materials Testing Reactor (JMTR) of JAERI.

This paper describes the current status and typical results of the above-mentioned research programs.

2. RESEARCH ON RPV INTEGRITY

2.1 Fracture toughness evaluation by means of master curve approach

To assure the structural integrity of RPV throughout its operational period, fracture toughness after neutron irradiation must be determined from the surveillance specimens. Many efforts have been made to develop fracture toughness evaluation method in the transition temperature range, called “Master Curve” method, and this has been accepted and introduced in the standards in EU and US. ASTM established a standard test method to determine the reference temperature, T_0, of fracture toughness in the transition range (E1921-97). This method determines the temperature dependence of fracture toughness using 6 specimens at least. Since only a small number of Charpy size specimens are available from a surveillance capsule, further research has been conducted at JAERI to establish an improved testing and evaluation method applicable to those specimens. We have conducted fracture toughness tests of Japanese RPV steels using precracked Charpy-v (PCCv) specimens The specimen size effect on fracture toughness and how to select the test temperature were investigated by comparing the results with the other fracture toughness tests obtained from larger specimens.

Five kinds of ASTM A533B class 1 plates with high or low level of impurities were used in the study. The bulk Cu contents varied from 0.02wt% to 0.19wt%. The details of chemical compositions are given in elsewhere [1,2]. The testing and evaluation for fracture toughness were performed according to mainly ASTM E1921-97. Four types of specimens were used for the fracture toughness testing, namely PCCv, 0.5T-DCT, 1T-CT and 4T-CT. In most cases, the specimens were side-grooved by 10% on each side of the specimen after precracking. Neutron irradiation of PCCv specimens as well as standard Charpy-v specimens was carried out at JMTR. The values of fast neutron fluence are 2 to 13×10^{19} (n/cm^2, E>1MeV) considering typical fluence values up to the extended operation of Japanese PWRs.

Using the median value of 1T-equivalent (25 mm was used in this study) adjusted data, the temperature dependence of cleavage fracture toughness within the transition range is expressed by the following equation in ASTM E1921-97.

$$K_{JC\_1T(med)} = 30 + 70 \cdot \exp\{0.019 \cdot (T - T_0)\} \quad (1)$$

This equation has been established by statistical analyses using a huge database including unirradiated and irradiated RPV steels. This equation defines the cleavage fracture toughness transition curve by a reference temperature, T_0. Before neutron irradiation, the reference temperatures, T_0, were evaluated at several temperatures, and by PCCv and 1T-CT. The base test temperature was selected for each material according to E1921 procedure, which uses a correlation between Charpy index temperature and test temperature. Higher and lower temperatures were also chosen to check the test temperature dependence on the master curve method. The deviation of each T_0 measurement from the average of T_0 values determined by PCCv specimens is shown in Fig. 1. The data points in the lower portion of the figure mean lower T_0 values than the average values, which result in non-conservative evaluations. In the figure, filled symbols indicate T_0 values obtained from data sets including invalid data and then determined by the censoring scheme. Testing of PCCv specimens at higher temperature than the average T_0 requires a data censoring and leads to a lower, non-conservative T_0 value as indicated in Fig. 1. It is recommended from this plot that testing around (T_0-20)°C can give a comparative result of PCCv to that of 1T-CT. However, this arises a question how the test temperature is selected since T_0 value is unknown. We therefore established an empirical correlation between T_0 and Charpy 41J transition temperature as shown in Fig. 2 including reference data [3, 4]. Using the correlation, the appropriate fracture toughness test temperature for PCCv is around (T_{41J} – 60)°C.

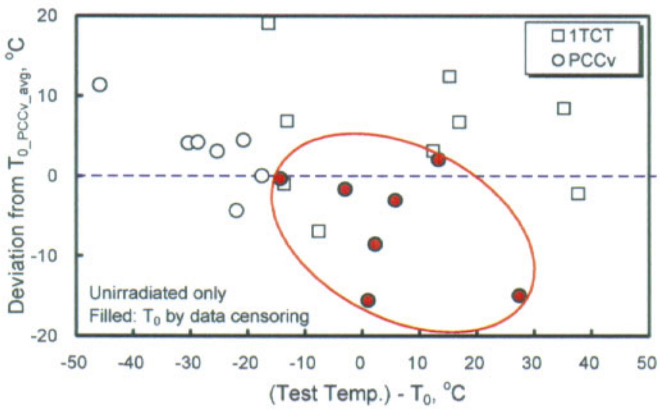

FIG. 1. Relation between test temperature and T_0 [1].

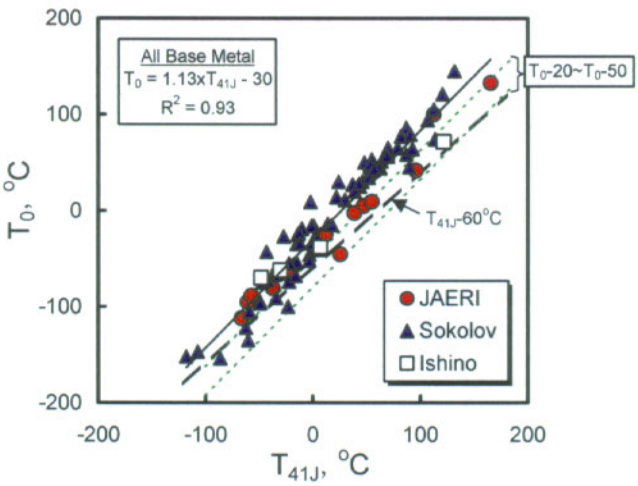

FIG. 2. Correlation of T_0 and T_{41J} with reference data [3, 4] for base metal [2].

More than 500 fracture toughness data in the transition temperature range are plotted in Fig. 3. The specimen size effect adjustment was applied to the data from other than 1T-CT specimens. The adjusted fracture toughness K_{JC} value to 1T thickness (25 mm in this study) was expressed as K_{JC_25mm}. The T_0 value for each material condition was defined by the average of T_0 by PCCv specimens. The master curve and the 5% and 95% tolerance bounds are also illustrated in the figure. Except for the data above (T_0 + 50) °C where data are sparse, fracture toughness data are agreed well with the master curve determined by T_0 of PCCv.

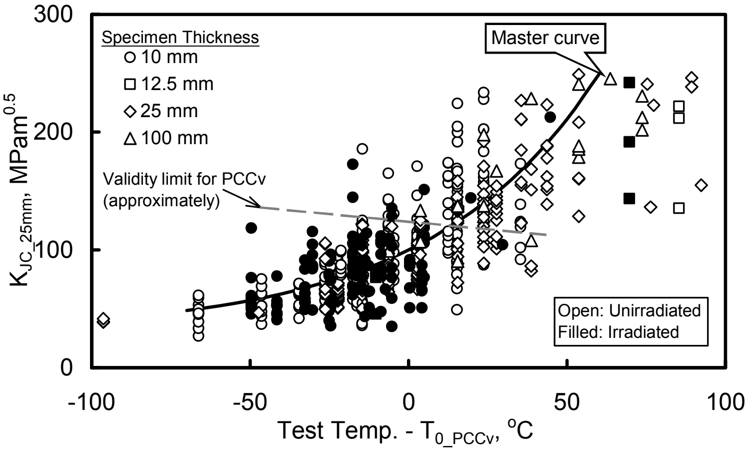

FIG. 3 . Fracture toughness data normalized by the T_0 determined from PCCv data [1].

2.2 Neutron irradiation embrittlement

The prediction of neutron irradiation embrittlement is also of great importance in assessing the structural integrity of RPV. There are several embrittlement correlations in estimating the transition temperature shift in several countries. All of these correlations contain fast neutron fluence and Cu content; most also consider P and Ni contents, and in some cases Si. Substantial effort has been made to improve on the correlation using the broader data base and to incorporate the understanding of embrittlement mechanisms. There still remained the issues which need further research. In this section the recent results of embrittlement study on the other factors which may contribute to irradiation embrittlement are presented.

2.2.1 Effect of γ-ray irradiation

The effect of γ-ray irradiation on the embrittlement has been examined to improve the accuracy of irradiation embrittlement prediction. Only fast neutron dose (E>1MeV) has been included in the present prediction of RPV steels where a ratio of γ-ray induced displacements per atom (dpa) is several percentages. However, it has been pointed out that γ-ray may effectively contribute to the irradiation embrittlement, because the survival rate of freely migrating defects (FMD) generated by simple atomic displacement via Compton scattering in γ-ray irradiation would be higher than that by cascade in fast neutron irradiation at the same dpa. The FMD induces the Cu migration, resulting in the embrittlement of the steels. In the present study [5], Fe-Cu model alloys, which aim to extract the embrittling effect of Cu as an impurity atom in RPV steel, were irradiated by high-energy electrons for simulating γ-ray irradiation and neutrons in test reactor irradiation. The irradiation-induced hardening as a measure of embrittlement was compared between these irradiations.

In Fig. 4, the electron and neutron-induced hardness increases in Vickers hardness for the Fe-Cu model alloys with high and low Cu contents (0.6 wt%, 0.02 wt%) were plotted against dpa. The significant increase in hardening for Fe-0.6%Cu alloy was mainly attributed to Cu precipitation. The difference between electron and neutron irradiation hardening was found to be small on a per dpa basis. Although the result is contrary to what we expected as described above, practically dpa works well as a scaling parameter for the estimation of irradiation embrittlement.

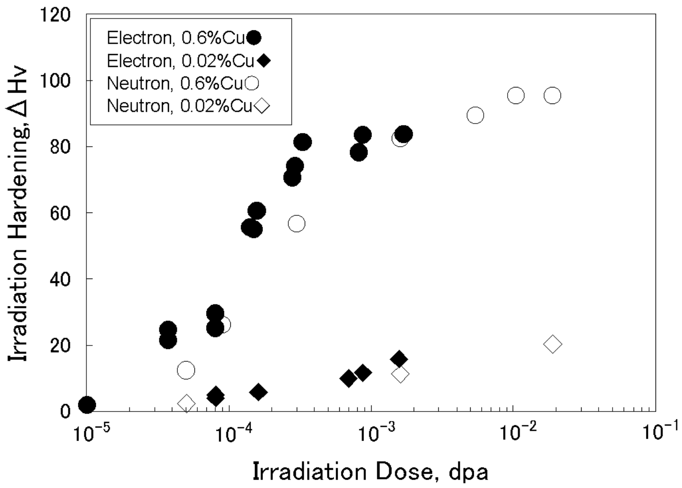

FIG. 4. Comparison of electron and neutron induced hardening after 250℃ irradiations. Both dose rate are almost the same; 7x10^{-9} dpa/s [5].

2.2.2 Integranular embrittlement

Neutron irradiation induces hardening, thereby increasing the ductile-brittle transition temperature (DBTT) in ferritic alloys. The present embrittlement prediction is based only on this hardening mechanism. It has been recently recognized that one of the other possible embrittlement mechanism of RPV steels is intergranular P segregation [6]. It is known that P segregation is promoted by neutron irradiation and the presence of P at grain boundaries weakens the cohesive strength of grain boundaries, leading to an increase in DBTT through intergranular failure. These are related to coupled fluxes of impurity and point defects near grain boundaries generated by irradiation [6]. However, the mechanism of irradiation-induced non-equilibrium segregation and embrittlement has not been fully understood.

The effect of bulk P contents on hardening, non-equilibrium intergranular segregation and embrittlement has been studied in three Fe-based model alloys doped with Mn and different bulk levels of P [7]. We also highlighted the role of intergranular C as a grain boundary strengthening element. The P-doped alloys containing 0.016 wt.%, 0.117 wt.% and 0.38 wt.% were used and designated as PL, PM and PH alloys, respectively. The chemical compositions and heat treatment conditions are given in elsewhere [7]. The C contents of the alloys varied from 0.0011 to 0.0027 wt%. These alloys were subjected to neutron irradiation (E>0.1MeV: fluence of 1 x 10^{25} n/m^2 at 711K for 2120 h) or irradiation-equivalent thermal aging (ETA). The DBTT was determined by Small Punch (SP) test [8]. The average peak height ratio (PHR) of first derivative Auger signals of several elements to the iron peak (703 keV) was determined from 25-40 data points. The grain boundary concentration (fraction of monolayer coverage) of these elements was estimated from the PHR [9].

The effects of the segregated impurities and microhardness on the intergranular embrittlement are now presented. The relationships of the DBTT to the P segregation, C segregation and/or microhardness in the variously treated alloys are shown in Fig. 5. The DBTT largely increased with increasing microhardness and slightly increased with increasing P segregation. On the contrary, there is an increasing trend of the DBTT when the segregated C becomes low. The irradiation decreased the segregated C, thereby weakening the grain boundary toughening effect produced by segregated C.

We derived an empirical equation of the DBTT as a function of the segregated impurities and hardness incorporating the present results together with the earlier data from the irradiated and post-irradiation annealing (PIA) treated Mn-free ferritic alloys [10]. Using P-doped alloys subjected to PIA at 788 K for 100 h as the reference point, the slope against individual parameters was first determined for alloys with similar values of other variables. Then least-square fitting was carried out in the equation superimposing the several linear terms. The equation can be given by [7],

$$(DBTT)_{SP}\ (K) = 36C^{\phi}_{P} + 573C^{\phi}_{S} - 244C^{\phi}_{C} + 1.1Hv + 61 \qquad (2)$$

where C^{ϕ}_{P}, C^{ϕ}_{C} and C^{ϕ}_{C} are the concentration of P, S and C, respectively, and Hv is the microhardness. The coefficient of the individual terms represents the strength of metallurgical variables controlling the DBTT shift. It is recognized that the embrittling potency of P is much less that of S. In addition, segregated C has a strong grain boundary toughening effect. The suppression of segregated C as well as the P segregation led to intergranular embrittlement in the irradiated PL and PM alloys although the reason for the DBTT decrease in the irradiated PH alloy is not clear. It should be pointed out that the shift in the $(DBTT)_{CVN}$ obtained by the Charpy impact test is about third times greater relative to that in $(DBTT)_{SP}$ by the SP one [14].

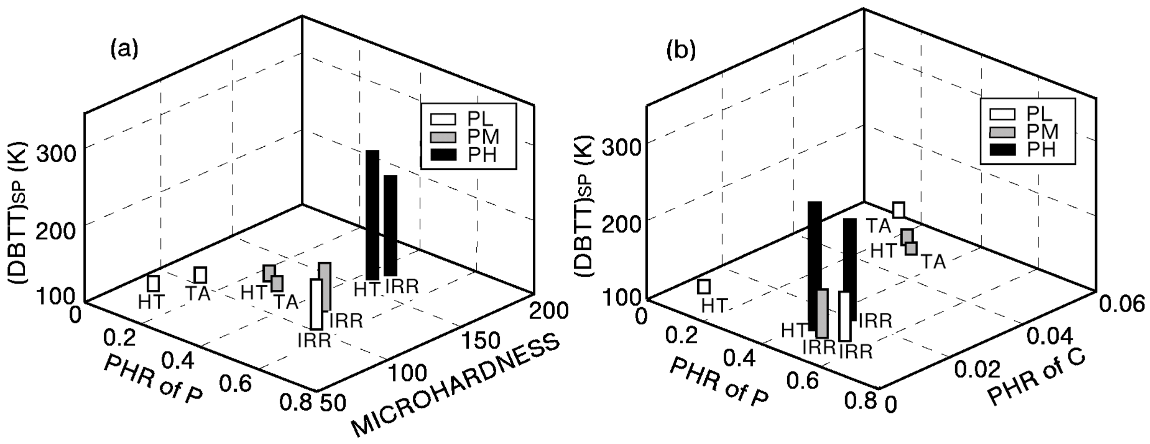

FIG. 5. Correlation of DBTT with segregated P and (a) microhardness, and (b) segregated C in as- heat-treated (HT), thermally aged (TA) and irradiated (IRR) alloy [7].

2.3 Development of probabilistic fracture mechanics analysis code

Probabilistic fracture mechanics (PFM) has been used in the fields of reliability analysis, life extension assessment and risk management for important structural components as a promising and rational evaluation methodology. The probabilistic approaches are also being introduced into regulations and standards related to structural integrity such as pressurized thermal shock (PTS), leak before break (LBB) and etc in the USA.

We have developed the PFM analysis code named PASCAL (PFM Analysis of Structural Components in Aging LWR) [11-14]. This code can evaluate the conditional probability of crack initiation and failure of a pressure vessel under transient conditions such as a PTS and possesses many improved input functions in relative to those in existing codes. The probabilistic simulation methods used in this code are the importance sampling Monte Carlo method for an infinite length surface crack and the stratified sampling Monte Carlo method for a semi-elliptical surface crack. The initial crack size, chemical composition, neutron fluence, fracture toughness and ductile to brittle transition temperature are all treated as probabilistic variables.

PASCAL has many original functions such as the fracture criterion of R6 method, the initial semi-elliptical surface crack with distributions of both depth and aspect ratio, the model to evaluate the decrease of upper shelf fracture toughness, the model to evaluate the effect of thermal annealing, the equations of stress intensity factor, the models of semi-elliptical surface crack extension, and the procedures for optimized sampling. The accuracy and reliability of this code have been verified in some earlier analyses [13,14]. The NRC/EPRI PTS benchmark analysis using R6 method shown in Fig. 6 has indicated that the conditional fracture probability by R6 method category 3 is much lower than that by K_{IC}/K_{Ia} criterion. Such efforts for improving the reliability and efficiency of the PFM analysis are being continued towards establishing a utility tool for integrity evaluation of aged LWR components. The master curve approach described in the earlier section will be also introduced in PASCAL.

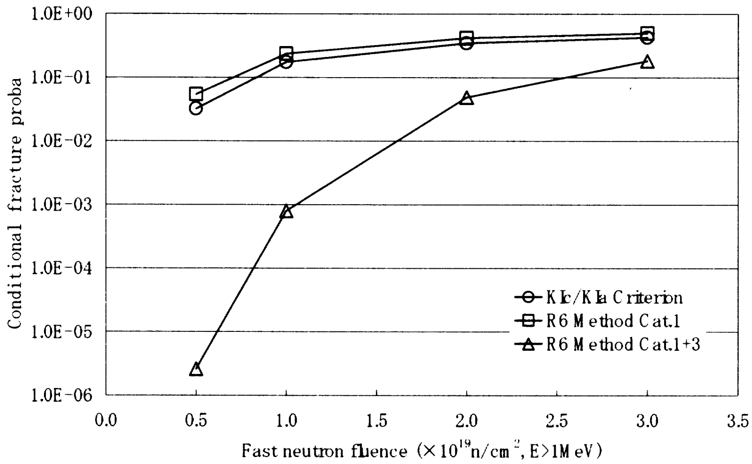

FIG. 6. Influence of fracture criterion on conditional fracture probability.

3. RESEARCH ON IASCC OF CORE INTERNAL MATERIALS

Core internals are not the pressure boundarymand many of them are replaceable.However, their functions in core are significant to keep the safety operation of LWRs, because core internals align the fuel and control rod assemblies properly in core and control the coolant flow in vessel. Degradation of the core internals could cause an irregular flow of coolant water or loose part which could bring about the damage to other components. Recently in Japan, many LWRs are suffering from the stress corrosion cracking (SCC) at core shroud and its support of BWR. To assess the cause of SCC at core internals, it is important to clarify effects of neutron radiation on SCC, because the radiation effects are accumulating in the core internal materials and therefore it is the aging matter. The specific SCC of core internal is the irradiation assisted stress corrosion cracking (IASCC).

When the LWR has been running over a long period, IASCC might occur in the core structure due to the synergistic effects of the high-level neutron irradiation and the pressurized high temperature water environment. Consequently fundamental understanding of the mechanism and countermeasures against IASCC are the important issues with regard to the long-term operation of the LWR. IASCC has been studied since the beginning of the 1980s and the phenomenological knowledge on IASCC is accrued extensively. However, mainly due to the experimental difficulties, data for the mechanistic understanding and prediction of failures of the specific in-vessel components are still insufficient and further well-controlled experiments are needed.

JAERI had initiated SCC research in the late 1960s on the occasion of SCC incidents experienced at the Japan Power Demonstration Reactor (JPDR) which were found at the stainless steel overlay lining of the pressure vessel [15]. In 1989, JAERI started an IASCC study of core structural material based on the experience and experimental techniques developed for two decades using its hot-laboratories and irradiation facilities. IASCC is one of the most complicate degradation phenomena experienced by the reactor structural materials. To elucidate the mechanism of IASCC, therefore, three approached are necessary which are (1) fundamental studies to separate the primary processes causing IASCC, (2) IASCC testing under the experimental conditions which simulate the actual LWR core environment, e.g. in-pile IASCC testing, (3) numerical modeling and simulation.

3.1 Fundamental studies for mechanistic understanding of IASCC

At JAERI in the last decade, for the fundamental research on IASCC of core internal materials, i.e. mainly austenitic stainless steels, various experimental facilities have been installed for the post- irradiation examinations on neutron irradiated materials at hot-laboratories. The facilities are including slow strain rate testing (SSRT) or crack propagation testing machines which can simulate BWR coolant conditions, electrochemical corrosion equipments, high resolution transmission electron microscope (FE-TEM), scanning electron microscope with electron back-scattering (EBSP) system, etc. Neutron irradiation of specimens have been carried out at JMTR and the Japan Research Reactor No.3, JRR-3 operated by JAERI, besides ion irradiation can be performed by means of the ion accelerators at Takasaki Establishment of JAERI. The following example of experimental results was obtained through irradiation and post-irradiation examinations using above facilities.

Effects of alloying and minor elements of alloys on IASCC behavior are essential to understand IASCC and to develop IASCC resistant material, because the evolution of defect microstructure and radiation induced segregation are influenced sensitively by chemical composition of alloys. Therefore, to study the effect of elements on IASCC, we irradiated the model stainless steels which are based high-purity 304 and 316 type experimental alloys and their heats doped with minor elements, i.e. C, Si, P, S, Ti [16,17]. Specimens of the solution annealed alloys were irradiated at JRR-3 at 513 K up to a neutron fluence of $6x10^{24}$ n/m^2 (E>1MeV). After the irradiation, specimens were examined by SSRT technique to evaluate susceptibility to IASCC in oxygenated high temperature water. In addition, microstructures of irradiated alloys were examined by FE-TEM.

Fig. 7 [17, 18] summarizes results of SSRT where the susceptibility to IASCC is indicated as %IASCC evaluated from areal fractions of intergranular (IG) and transgranular (TG) SCC portions on fracture surface of tested specimens. It is known from field experiences that IASCC in LWRs appears as IG cracking, therefore, %IASCC evaluated by IG fraction is more important. Remarkable effects of C, Mo and S can be derived from Fig. 7. In a series of 304 type model alloys, an effect of C addition can be obviously seen on fracture mode. A dominant fracture mode of alloys without C addition was IG cracking but alloys doped with C caused a suppression of IG cracking and increased fractions of TG cracking. Comparing HP304 and HP316, or HP304/C and HP316/C (/C notifies an addition of C), one can conclude that an addition of Mo entirely suppress IASCC susceptibility at the present neutron fluence.

Among the tested alloys, alloys doped with S, i.e. HP304/S and HP316/C/Ti/S showed relatively high susceptibilities to IG or TG cracking, therefore it is suggested that existence of S around 0.04% is very injurious to IASCC.

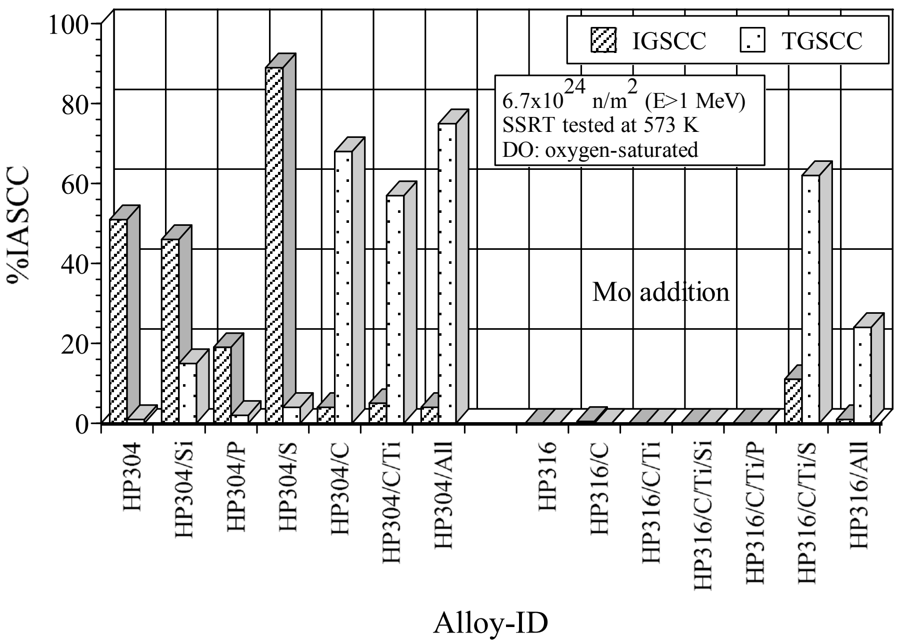

FIG. 7. IASCC susceptibility of irradiated alloys evaluated by SSRT [18].

To investigate the relation between effects of minor elements on IASCC behavior and radiation induced microstructure, analyses by FE-TEM were carried out on the irradiated specimens [19]. Fig. 8 [19] shows nominal stress-strain curves from SSRT for HP304, HP304/C and HP304/Si. Former two alloys showed similar maximum stresses and %IASCC of IG type as seen in Fig. 7. However, these alloys showed different total elongations and microstructures. As seen in TEM photographs of Fig. 9 [19], in HP304 Frank loops and small clusters were observed and the addition of C caused an increase in number of Frank loops after irradiation, while the addition of Si almost completely suppressed the nucleation and growth of Frank loops. These results suggest that addition of Si into HP304 alloy can not suppress IASCC but it delayed the fracture due to IASCC and it is cased by the effect of Si addition on the defect microstructure, on the other hand, an addition of C suppress IG type IASCC and it may be caused by the increase of radiation hardening.

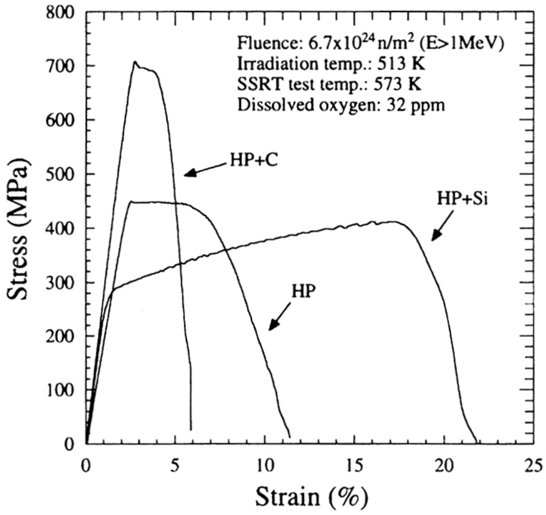

FIG. 8. Stress-strain curves of irradiated stainless steels from SSRT in high temperature water [19].

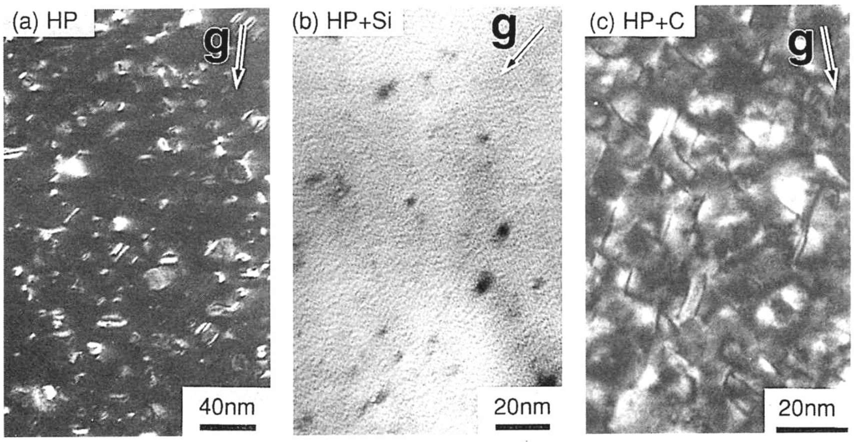

FIG. 9. Microstructures of irradiated model stainless steels [19].

3.2 Development of IASCC evaluation technology for BWR plants based on post-irradiation tests

Referring to the report by MITI [20] and focusing on the core internals issue, it was suggested that IASCC and fatigue were essential for assessing their degradation [21]. In 2000, METI has begun the project for IASCC technological development as a part of the more comprehensive program for technological development of countermeasures for aged of LWRs [21]. In the project for IASCC technological development, the basic essential data were planned to be prepared for the maintenance standard and safety evaluation of core internals under a promotion of the Japan Power Engineering and Inspection Corporation (JAPEIC). In JAPEIC, to focus on the aging issues from a broad perspective, the Nuclear Power Plant Life Engineering Center (PLEC) was established in response to the request of METI that was independent from the other activities of JAPEIC to secure its independency and specialty. In the IASCC project, materials irradiation and post-irradiation examinations are undertaken in cooperation with industry, governmental organizations including JAERI and academics.

The project for IASCC technological development consists of BWR and PWR relevant works. JAERI is engaged in the BWR related testing and research program, which includes the neutron irradiation at JMTR and post-irradiation examinations shared by the participants. In a part of the METI project aiming at IASCC of BWR, the objective is to evaluate IASCC initiation and propagation data necessary for making IASCC database satisfactory to contribute for preparing the maintenance standard against IASCC, which will be used for the assessment of timing and intervals of the inspections of core internals. In the IASCC project for BWR, core internal materials used for BWR, i.e. types 304, 304L and 316 stainless steels will be examined at hot laboratories after irradiation at JMTR up to four levels of neutron fluence of $5x10^{24}$, $1x10^{25}$, $3x10^{25}$ and $1x10^{26}$ n/m^2 (E>1 MeV) to assess dependency of IASCC behavior on the neutron fluence. Experimental data to be obtained through the post-irradiation examinations are the crack propagation rate data, IASCC susceptibility data form slow strain rate testing (SSRT) and uni-axial constant load (UCL) tests in high temperature water, irradiation-induced stress relaxation data, fracture toughness data, etc. The IASCC national project will be completed in FY2008.

3.3 In-pile IASCC test for evaluation of the synergistic effects of irradiation, corrosion and stress

In core of the LWRs, IASCC can be caused by the various synergistic effects of neutron radiation, stress/strain and high temperature water on structural materials of austenitic stainless steels. On the other hand, it is well known that IASCC can be reproduced on alloys irradiated over threshold neutron fluence by post-irradiation examinations [22]. Through post-irradiation examinations, we have understood that a primary cause of IASCC is the generation/migration and aggregation of radiation defects and especially the radiation induced segregation (RIS) at grain boundaries plays an essential role in IASCC phenomena. However, it is considered that the reproduced IASCC by post-irradiation examinations must be carefully distinguished from the IASCC that occurs in the core under simultaneous effects of neutron radiation and high temperature water environment because it is known that the evolution of radiation damage microstructure in alloys is influenced by the applied stress [23]. Irradiation under stress loading may potentially lead to the different microchemistry, i.e. RIS, and mechanical properties, and consequently IASCC behavior. Although it is an inverse viewpoint of this synergy, the radiation induced stress relaxation is acting to reduce the internal stress in alloy and susceptibility to IASCC only in core. From a viewpoint of water chemistry, the radiolysis and radical species have influences on the corrosion process of materials in core. Since these synergies of radiation and stress/water are significant to understand IASCC behavior in the core, but not reproducible outside reactor core. The in-pile IASCC testing is, therefore, one of the key experiments to understand IASCC behavior of core internal materials. A high temperature water loop facility was designed to be installed at JMTR to carry out material irradiations and in-core testing for the study of IASCC [24]. JMTR has a high flexibility of reactor core arrangement and high accessibility to the core for installation of irradiation facilities. The water loop system for IASCC study was designed to simulate BWR environment and supply high temperature pure water into five irradiation capsules at a time. Its construction has been completed in FY2001 in the framework of cooperative research program between JAER and the Japanese electric power companies [25].

JMTR is a tank-in-pool type reactor cooled and moderated by light water with thermal power of 50 MW that has been operated for the material and fuel irradiation since 1969 at the Oarai establishment of JAERI. The maximum neutron flux in fuel region at the core is $4x10^{16}$

n/m^2/s (E>1.0 MeV). Coolant temperature is approximately 60°C. At JMTR, in FY 1999 a design activity for the new water loop system was started focusing on the irradiation and in-pile testing for IASCC studies. Fig. 10 shows a schematic diagram of the loop system. Since the loop facility is designed for material irradiation under BWR conditions, the maximum operational parameters are as follows; temperature: 593 K, pressure: 10 MPa and flow rate: 1 m^3/h. Temperature of inlet water to irradiation capsule can be controlled to keep temperature inside the capsule typically at 561 K. Pressure of water will be controlled the temperature inside capsule by means of saturation temperature control technique for base irradiation of specimens. Flow rate is an important parameter to be controlled at a specimen surface because it affects the electrochemical potential (ECP) and heat balance in the loop system. To simulate the normal water chemistry (NWC) and hydrogen water chemistry (HWC) conditions of BWR, the dissolved oxygen and dissolved hydrogen concentrations can be controlled at levels up to 200 ppb and 1 ppm respectively.

Using the loop facility at JMTR, in-pile IASCC tests will be carried out which are including the IASCC initiation testing by applying uni-axial constant loading test technique and the IASCC propagation or crack growth testing with compact tension (CT) type specimens. At present in FY2002, mockup tests of in-pile testing equipments to be installed in irradiation capsules are on going, and the tests in the JMTR core will be started in FY2003. Result to be obtained from the in-pile IASCC initiation and propagation tests will be compared with test results from post-irradiation examinations and it will reveal the synergy of neutron/gamma radiation and stress/water environment.

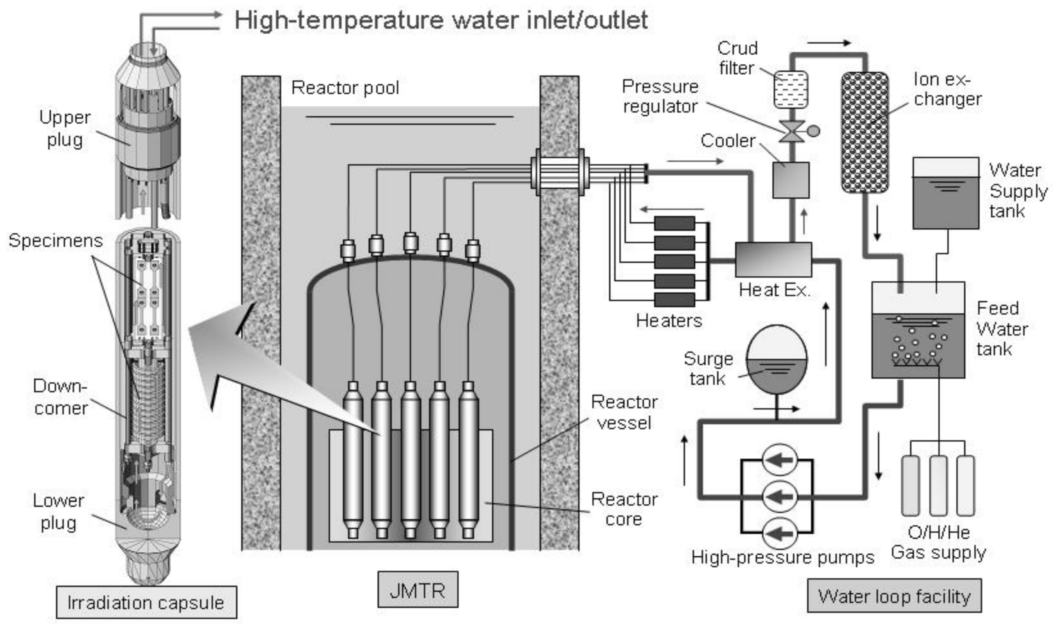

FIG. 10. Schematic drawing of loop facility for IASCC study constructed at JMTR..

3.4 Computational studies on materials database and numerical simulation

A material performance database, which was named JAERI Material Performance Database (JMPD) [26, 27], has been developed since 1986 focusing on the storage of data regarding research and development promoted at JAERI. The JMPD was designed for effective utilization of material data especially for the environmentally assisted degradation, e.g., fatigue or SCC behavior in aqueous or gaseous environments. As for a part of IASCC database, about 300 data of post irradiation SSRT from our experimental work and 20 open published papers were input. The IASCC data consist of those of type 304 and 316 materials at irradiation temperatures between 333 and 573 K. The fast neutron fluencies to the materials are in the range of from $1x10^{22}$ n/m^2 to $8x10^{26}$ n/m^2 (E>1MeV). The IASCC susceptibility of the materials has been examined by SSRT at around 573 K in high-temperature water containing dissolved oxygen concentration between 1ppb and 32ppm. Data analyses were performed with the knowledge on the factors controlling IASCC obtained by our results of the post irradiation SSRT [16, 17, 28, 29].

For an example, the IASCC susceptibility data compiled into the JMPD were plotted against fast neutron fluence (E>1MeV) in Fig. 11 [18, 22]. Though the percent IG cracking in the SSRT is a variable parameter, the susceptibility evaluation was made in terms of the IG cracking area in the SSRT for the common indicator in these database analyses. The data were scattered over a wide range of susceptibility against the neutron fluence. Since the dissolved oxygen (DO) content in high-temperature water is an essential factor for SCC phenomena, all data are classified into two groups by levels of DO content during SSRT. There is a tendency that the percent IG cracking of alloys tested in lower DO environment is smaller. According to the results of the post irradiation SSRT, the addition of molybdenum to 304 SS caused a drastic suppression of IASCC [17].

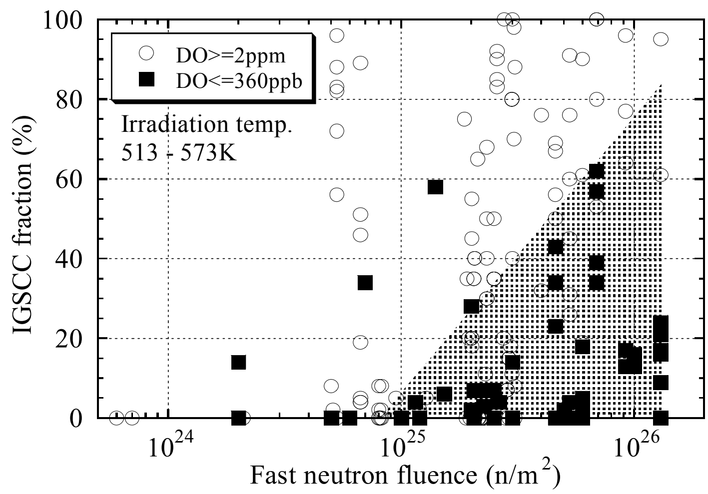

FIG. 11. Effect of DO on IASCC susceptibility of SSRT database in JMPD [18].

At JAERI, we are developing a finite element method (FEM) program simulating crack growth behavior of IASCC of stainless steels in pressurized high temperature water, by combining the crack growth model. The crack growth model consists of the two processes as shown in Fig. 12 [21]: (1) Growth and fracture of oxide film by model for crevice chemistry and (2) Dissolution of metal by slip dissolution model. Moreover, effects of the radiation induced segregation (RIS) at grain boundary and grain orientation etc. on the crack growth behavior of IASCC are considered in this model. Stress and strain distributions at crack tip are obtained by FEM analysis. In RIS analysis subprogram, RIS behavior is simulated taking into account fast neutron fluence, characteristics of grain boundary and stress distribution. In subprogram of model for crevice chemistry, growth and fracture behavior of oxide film is simulated by considering the RIS simulating results and stress distribution. Therefore, crack advance is numerically reproduced by the node release technique in FEM program by taking account of RIS and crevice chemistry subprograms.

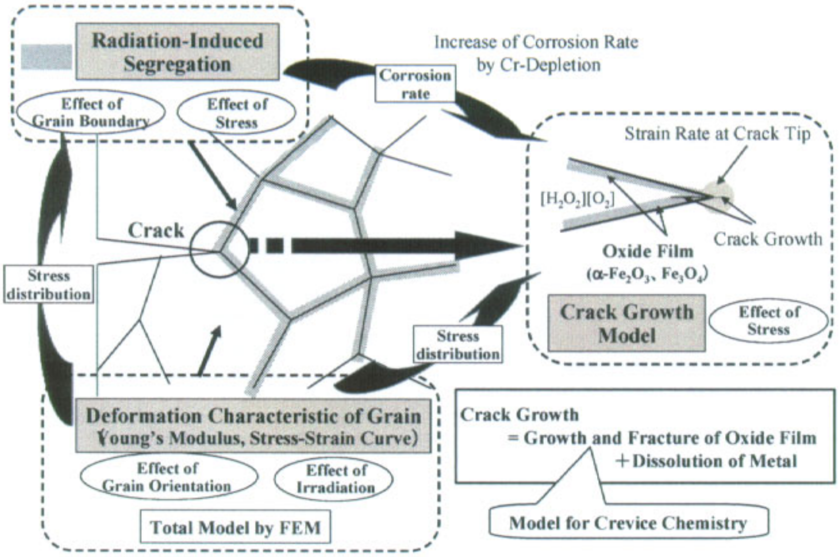

FIG. 12. Conceptual diagram for IASCC modeling [21].

4. SUMMARY

This paper described the research at JAERI on the aging degradation caused by neutron irradiation for structures and components which are important from the safety standpoint and/or are difficult in replacing and repairing, e.g., reactor pressure vessel (RPV) and core internals.

The RPV study is targeted to provide the new basis for the structural integrity analysis and neutron irradiation embrittlement evaluation in the long-term operation. We have been devoted to use precracked Charpy specimens (PCCv) available from surveillance specimens to directly determine the fracture toughness by the master curve approach. Fracture toughness tests using PCCv specimens and the analyses were performed in the transition range for unirradiated and irradiated Japanese RPV steels. For the irradiation embrittlement, the γ-ray effect and intergranular embrittlement which have potential to contribute to the present embrittlement correlations were investigated. It has been shown that both γ-ray and neutron induced hardening can be well scaled by calculated dpa and that the C desegregation as well as the P segregation due to neutron irradiation led to intergranular embrittlement. Also, we have developed the PFM analysis code named PASCAL. One of the features in this code is that the ductile crack extension analysis model is introduced. Results of some case studies suggested that the conditional fracture probability by R6 method category 3 was much lower than that by K_{IC}/K_{Ia} criterion, especially in the low fluence range.

IASCC is the main target of the research on aging of core internal materials at JAERI. For the fundamental research on IASCC, various experimental facilities have been installed for the post-irradiation examinations at hot-laboratories. Neutron irradiation of specimens were carried out at JMTR and JRR-3 operated by JAERI. As an example of experimental result, effects of alloying and minor elements of model stainless steels on IASCC behavior was described. In 2000, METI has begun the project for IASCC technological development as a part of the more comprehensive program for technological development of countermeasures for aging of LWRs. In the project, the basic essential data were planned to be prepared for the maintenance standard and safety evaluation of core internals. JAERI is engaged in the BWR related testing and research program, which includes the neutron irradiation at JMTR and post-irradiation examinations shared by the participants. Since the synergies of radiation and stress/water in the core are not reproducible outside reactor core, the in-pile IASCC testing is one of the key experiments to understand IASCC behavior. A high temperature water loop facility was, therefore, installed at JMTR to carry out material irradiations and in-core IASCC testing. Its construction has been completed in FY2001 and irradiation of specimens for IASCC studies is ongoing. IASCC is considerably complicated to elucidate its mechanism by the synergy of material, environmental and radiation: therefore an approach through the numerical modeling and simulation techniques is substantially worthy of being developed for understanding and predicting IASCC.

REFERENCES

[1] ONIZAWA, K. and SUZUKI, M., "Comparison of Transition Temperature Shifts Between Static Fracture Toughness and Charpy-v Impact Properties Due to Irradiation and Post-Irradiation Annealing for Japanese A533B-1 Steels," Effects of Radiation on Materials: 20th International Symposium, ASTM STP 1405, S. T. Rosinski, M. L. Grossbeck, T. R. Allen, and A. S. Kumar, Eds., American Society for Testing and Materials, West Conshohocken, PA (2001) 79-96.

[2] ONIZAWA, K. and SUZUKI, M., "Correlation Between Cleavage Fracture Toughness and Charpy Impact Properties in the Transition Range of Reactor Pressure Vessel Steels," Proceedings of APCFS and ATEM '01, The Japan Society of Mechanical Engineers, Sendai (2001) 140-145.

[3] SOKOLOV, M. and NANSTAD, R., "Comparison of Irradiation-Induced Shifts of K_{JC} and Charpy Impact Toughness for Reactor Pressure Vessel Steels," NUREG/CR-6609, U. S. NRC, November 2000.

[4] ISHINO, S., et al., "The Effect of Chemical Composition on Irradiation Embrittlement," Nuclear Engineering and Design, 119 (1990) 139-148.

[5] TOBITA, T., et al., "Hardening of Fe-Cu alloys at elevated temperatures by electron and neutron irradiations," Journal of Nuclear Materials, 299 (2001) 267-270.

[6] Review of Phosphorus Segregation and Intergranular Embrittlement in Reactor Pressure Vessel Steels (PWRMRP-19): PWR Materials Reliability Project, EPRI, Final Report, May 2000.

[7] NISHIYAMA, Y., BLOOMER, T.E. and KAMEDA, J. "Effect of Bulk Phosphorus Content on Hardening, Non-Equilibrium Segregation and Embrittlement in Neutron Irradiated Iron-Based Alloys, Materials Research Society Symposium Proceedings Vol.650, Microstructural Processes in Irradiated Materials, (2000) R6.10.1-6.

[8] KAMEDA, J., MAO, X. and BEVOLO, A. J., "Intergranular Impurity Segregation and Embrittlement in Neutron-Irradiated Iron Alloys," Journal of Nuclear Materials, 179-181(1991) 1034-1037

[9] BRIGGS, D. and SEAH, M. P., "Practical Surface Analysis," John Wiley & Sons (1995)

[10] KAMEDA, J. and BEVOLO, A. J., "Neutron Irradiation-Induced Intergranular Solute Segregation in Iron Base Alloys," Acta Metall., Vol. 37 No.12 (1989) 3283-3296.

[11] SHIBATA, K. et al., "Research and Development Related to PFM for Aged Nuclear Components," Nuclear Engineering and Design 214 (2002) 113-126.

[12] LI, Y. S. et al., "Improvements to a Probabilistic Fracture Mechanics Code for Evaluation the Integrity of a RPV Under Transient Loading," International Journal of Pressure Vessels and Piping 78 (2001) 271-282.

[13] SHIBATA, K. et al., "Development of PFM Code for Evaluating Reliability of Pressure Components Subjected to Transients Loading," Nuclear Engineering and Design 208 (2001) 1-13.

[14] SHIBATA, K., LI, Y. and KATO, D., "Introduction of Effect of Annealing into Probabilistic Fracture Mechanics Code and Results of Benchmark Analyses, ASME PVP-Vol.400 (2000) 49-54.
[15] KONDO, T., NAKAJIMA, H. and NAGASAKI, R., "Metallographic Investigation on the Cladding Failure in The Pressure Vessel of a BWR," Nuclear Engineering and Design 16 (1971) 205-222.
[16] TSUKADA, T., MIWA, Y. and NAKAJIMA, H., "Stress Corrosion Cracking of Irradiated Type 304 Stainless Steels," Proc. 7th Int. Sympo. Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Breckenridge, CO, NACE, p.1009 (1995).
[17] TSUKADA, T. et al., "Effects of Minor Elements on IASCC of Type 316 Model Stainless Steels," Proc. 8th Int. Sympo. Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Amelia Island, FL, ANS, p.795 (1997).
[18] KAJI, Y. et al., "Status of JAERI Material Performance Database (JMPD) and Its Use for Analyses of Aqueous Environmentally Assisted Cracking Data," ASTM STP 1401, West Conshohocken, PA (2000) 191-209.
[19] MIWA, Y. et al., "Effect of Minor Elements on Irradiation Assisted Stress Corrosion Cracking of Model Austenitic Stainless Steels," J. Nucl. Mater., 233-237 (1996) 1393-1396.
[20] MINISTRY OF INTERNATIONAL TRADE AND INDUSTRIES MITI, "Basic Policy on Aged Nuclear Power Plant," 1996
[21] NAKAJIMA, H. et al., "Recent Japanese experiences on SCC of structural materials in LWRs and future direction of relevant R&D," Proc. Quadripartite meeting of the national advisory committees of France, German, Japan and USA (Berlin, Germany, October 23-25, 2002) to be published.
[22] KAJI, Y. et al., "Development of A Comprehensive Material Performance Database (JMPD) and Analyses of Irradiation Assisted Stress Corrosion Cracking Data," Proc. 9th Int. Sympo. Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Newport Beach, CA (1999) 987-994.
[23] TANIGAWA, H., KOHYAMA, A. and KATOH, Y., "A Modeling of Radiation Induced Microstructural Evolution under Applied Stress in Austenitic Alloys," J. Nucl. Mater., 239 (1996) 80-84.
[24] TSUKADA, T. et al., "New In-Pile Water Loop Facility for IASCC Studies at JMTR," Proc. Int. Conf. Water Chemistry in Nuclear Reactor Systems – Operational Optimization and New Developments (Avignon, France, April 22-26, 2002) Paper No.91.
[25] JAPAN ATOMIC ENERGY RESEARCH INSTITUTE New Release, "Reliability of Irradiation Facility Simulating LWR Environment Confirmed on JMTR - IASCC irradiation test conducted to improve the reliability of the LWR," (April 8, 2002) URL: http://www.jaeri.go.jp/english/press/index.html.
[26] TSUJI, H. et al., "Development of Comprehensive Material Performance Database for Nuclear Applications," J. Nucl. Sci. Technol., 30, (1993) 1234-1242.
[27] YOKOYAMA, N. et al., "Development of Comprehensive Material Performance Database for Nuclear Applications," *Computerization and Networking of Materials Database: Fifth Volume*, ASTM STP 1311 (1997) 261-272.
[28] TSUKADA, T. et al., "Stress Corrosion Cracking Susceptibility of Neutron Irradiated Stainless Steels in Aqueous Environment," Proc. 7th Int. Conf. on Nucl. Eng. (ICONE-7) (1999) L4-3.
[29] KAJI, Y. et al., "Status of JAERI Material Performance Database (JMPD) and Analysis of Irradiation Assisted Stress Corrosion Cracking (IASCC) Data," Journal of Nuclear Science and Technology, Vol.37, No.11 (2000) 949-958.

AGEING OF AUSTENITIC PIPES IN IGNALINA NPP

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Abstract

The Ignalina NPP contains two RBMK-1500 reactors. Operation of the first unit of Ignalina NPP started in December 1983, the second unit in August 1987. After some operation period a lot of defects were detected in the primary circuit piping, produced from austenitic stainless steel 08X18H10T. The outside diameter of piping is 325 mm, the wall thickness – 16 mm. At normal operation conditions the internal pressure is 6.9-8.4 MPa and the temperature of coolant is 260-270°C. Defects in welded joints were detected during In-service inspections. Metalographic investigations defined that crack growth mechanism is Intergranular Stress Corrosion Cracking (IGSCC). All cracks appearing at the inner surface in heat affected zone near to weld root and grow to outside close to fusion line. The presentation covers current performances and further "Ageing Management" related actions and plans as well as experience (lessons learned) on solving IGSCC phenomenon, which is currently under investigations and no yet comprehensive answer how to avoid it.

1. INTRODUCTION

The cracking in Ignalina NPP main circulation circuit (MCC) piping welds, produced from austenitic stainless steel of type 08X18H10T is caused by IGSCC. IGSCC is a combination of the following three factors: material, stresses and environment. All these factors are important for cracking.

In 2000-2001 performed IAEA Extrabudgetary Programme on Mitigation of Intergranular Stress Corrosion Cracking in RBMK Reactors an engineering judgement on the parameters affecting the observed cracking is given based on received information and on previous experience and knowledge. The main root causes of intergranular stress corrosion cracking in the stainless steel of type 08X18H10T can be summarised as follows [1]:

- sensitisation, which is caused by a high degree of free carbon and a low stabilisation ratio in the material and high heat input during welding;
- deformation of the pipe inner surface due to weld preparation;
- geometrical weld imperfections accelerating crack initiation;
- deformation of the material in the heat affected zone (HAZ) due to weld shrinkage;
- high tensile stresses (residual and/or operational), indicated by a large opening of the cracks;
- environmental parameters, indicated by chlorides on the fracture surface, known condenser leakage incidents, possible sulphate intrusions, which cannot be ruled out, water impurities and the oxidising power of the water;
- operational fluctuating stresses indicated by observation of fatigue striations on the fracture surfaces.

The IGSCC cracks in MCC austenitic piping of Ignalina NPP appear at the inner surface in HAZ near to weld root and grow to outside close to fusion line. The HAZ material is susceptible to IGSCC and sensitised in most cases. The sensitisation occur due to overheating during welding and is an important factor in the cracking behaviour. According to investigations [2], the crack growth stops reaching zone of low sensitisation approximately in the middle of pipe wall. It determines the maximal height of detected cracks of 8-10 mm. However, it is necessary to mention that is not known with a certainty that only degree of sensitisation stops a crack grow. If one waits long enough without inspections there are no guaranties that the crack will not penetrate the wall thickness. Up to now no leakage has occurred.

The stresses in MCC piping occuring due to internal pressure (membrane stress), deadweight and thermal expantion (bending stress) as well as deformation during welding due to weld shrinkage (weld residual stresses). A numerical finite element simulation of welding process showed, that the axial weld residual stresses are tensile at the inside of the pipe and compressive at the outside of the pipe [3]. The operating and residual stresses are high enough to contribute the growth of cracks.

The environment inside MCC piping is oxidising. In Ignalina NPP at present time water chemistry is considerably improved and during normal operation oxygen content is very low, however it increases during start up and shutdown of the reactor, and during outages.

2. DATA ABOUT MCC AUSTENITIC PIPING OF IGNALINA NPP

The Ignalina NPP contains two RBMK-1500 reactors. Operation of the first unit of Ignalina NPP started in December 1983, the second unit in August 1987. Construction of the third unit was terminated in 1989 because of political reasons. The MCC consists of two loops, whose components are arranged symmetrically with respect to the vertical axis of the reactor. The MCC piping is produced from austenitic stainless steel 08X18H10T. The outside diameter of piping is 325 mm, the wall thickness – 16 mm. At normal operation conditions the internal pressure is 6.9-8.4 MPa and the temperature of coolant is 260-270°C [4]. The one loop of Ignalina NPP main circulation circuit austenitic piping is shown in Fig.1.

There are 3 types of MCC austenitic piping welds: shop, assembly, and repair. These welds were produced using different welding methodologies. The first IGSCC cracks in unit 1 was detectect in 1987, and in unit 2 – in 1991. In both units the first cracks were detected in water equilizing piping. After some time of operation the IGSCC crack also were detected in other MCC piping systemes. After repairs in 2000 totally 1431 weld existed in Unit 1, and 1240 – in Unit 2. Total 335 IGSCC-cases were identified in both units: 278 cases in the Unit 1 during 17 years of operation and 57 cases in the Unit 2 during 13 years of operation. Cumulative number of IGSCC cases in Ignalina NPP after repair in 2000 is shown in Fig.2. Straight lines in the figure represent medium number of IGSCC-cases per year. For the Unit 1 this number is equal to 16.4, while for the Unit 2 – 4.4. Frequencies of IGSCC defect occurrence in shop welds, assembly plus repair welds and all welds are presnted in Fig.3. Shop welds is separated because the amount of IGSCC defects in these welds is larger (except WEP) than in assembly and repair welds. Also shop welds were welded with higher heat input than assembly and repair welds. Frequencies were calculated assuming that IGSCC defect occurrence is a linear function, because during the latest years in-service inspections results of both units, when more advanced inspections were started to use, show a linear defect occurrence. In Fig.2. it can be noticed a significant increase of IGSCC defect occurrence in the latest years for unit 1. The reasons of such an increase are probably implementation of more advanced inspection methods and increased in-service inspections volume.

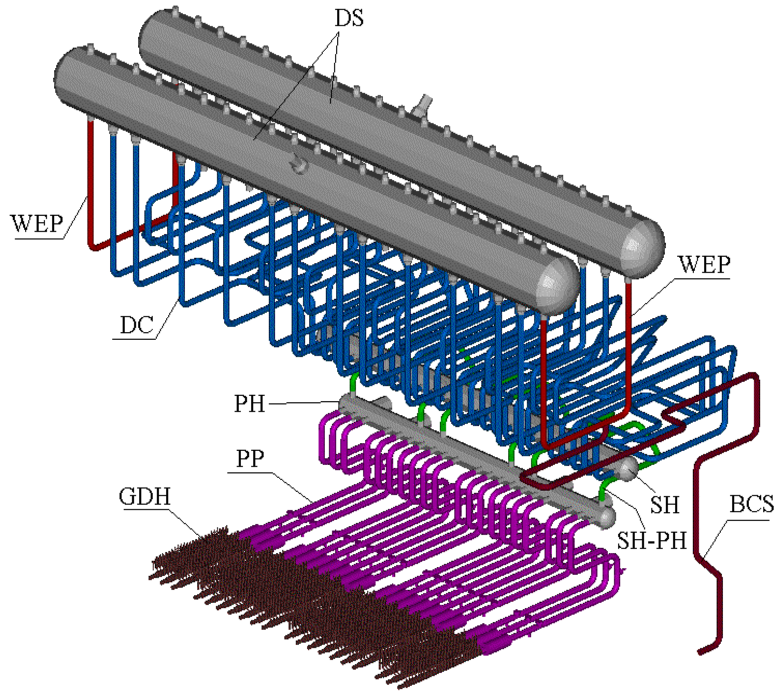


FIG. 1. MCC austenitic piping of one loop (WEP – water equalizing piping; DC – downcomers; SH-PH – bypass between suction header (SH) and pressure header (PH); PP – pressure piping, connecting pressure header (PH) and group distribution header (GDH); GDH – group distribution headers; BCS – blowdown & cooldown system piping).

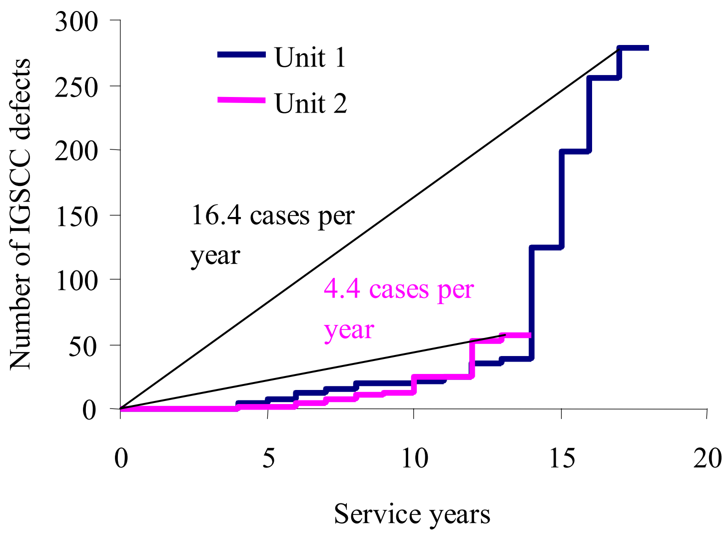
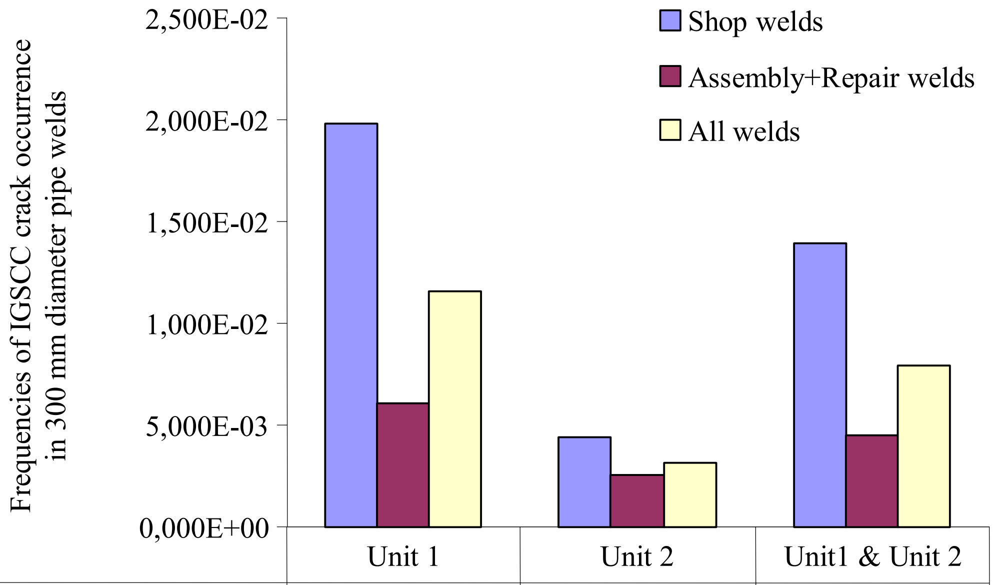


FIG. 2. Cumulative number of IGSCC cases in Ignalina NPP after repair in 2000.



	Unit 1	Unit 2	Unit1 & Unit 2
Shop welds	1,976E-02	4,368E-03	1,388E-02
Assembly+Repair welds	6,059E-03	2,582E-03	4,486E-03
All welds	1,152E-02	3,174E-03	7,958E-03

FIG. 3. Frequencies of IGSCC crack occurrence.

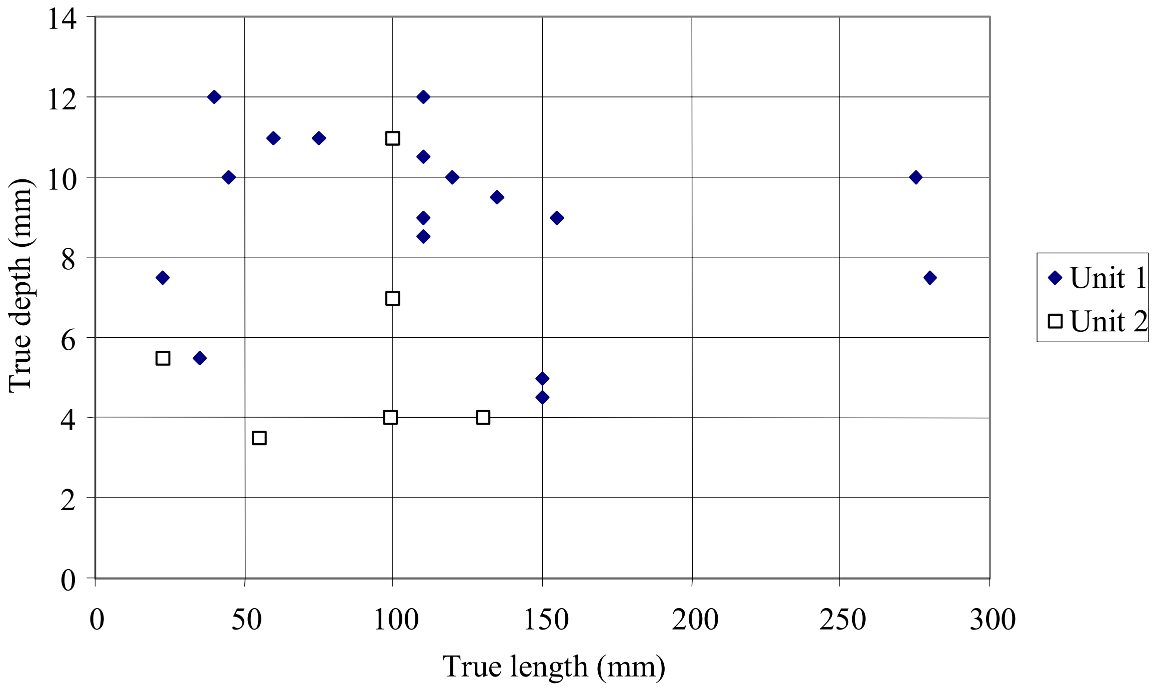


FIG. 4. True length and depth (measured in destructive examinations) of IGSCC cracks in Ignalina NPP austenitic steel pipelines with relative diameter of 300 mm.

In Fig. 4 comparison of defect sizes, measured during destructive examinations are presented. Comparison of defect sizes sowed that here is no relation between IGSCC crack depth and length. Depth of the deepest cracks reaches up to 12 mm, i.e. 75% of wall thickness. Length of the longest cracks reaches up to 280 mm, i.e. approximately 27% of circumference. Through-wall cracks were not detected in austenitic 300 mm relative diameter piping.

3. IN-SERVICE INSPECTION OF MCC AUSTENITIC PIPING

In Ignalina NPP X-ray technique for austenitic steel pipelines with relative diameter of 300 mm has been used up to 1997. In 1997 at Ignalina NPP Unit 2 some initial ultrasonic tests were used to confirm ISI results collected with X-ray probe. In 1998, X-ray and KRAB-I probe (manual UT) was used only at Ignalina NPP Unit 1. From 1999, the enhanced manual UT methodology with improved KRAB-II probes and various equipment of automatic UT was used.

In 1999 100% inspection of austenitic steel pipelines with relative diameter of 300 mm was performed. From this year the inspections are scheduled with an interval of 4 years, i.e. during 4 year interval in total 100% inspections will be performed (for example in unit 2 in 2001– 50% and in 2003– other 50%) [5].

A criterion based on crack length is used. If the detected crack length is above 100 mm then a repair is made and next year the inspection of new welds should be performed. If the detected crack length is below 100 mm, it is left for continued operation without repair with an inspection interval of one year.

4. IGSCC MANAGEMENT AND MITIGATION ACTIONS AND PLANS

(1) In-service inspections. At present time the manual and automatic UT techniques is used in the Ignalina NPP for piping base metal and weld metal inspection. In some cases the manual inspection was more uncertain than automatic inspection, so now mostly automatic UT inspection is used if possible (not all welds is accessible for automatic inspection). Procedures used in the Ignalina NPP at present time allows to determine the flaw length only, and procedure which allows to determine the depth of the flaw is under development.

(2) Repairs of defected welds. Repairs of austenitic piping are performing cutting out defected weld with heat affected zone and inserting a new piece of pipe. This means, that each repair increases the amount of welds. Operation experience indicate that cracking occur also in repaired welds. For repairing mainly manual welding is used. Only a few repair welds are welded using automatic welding, but in future automatic welding is planed to be used to an increasing extent. At present time Ignalina NPP developing a new welding methodology which involves minimal heat input and minimal number of passes. Hopefully it will reduce sensitization level and weld residual stresses.

(3) Leak-Before-Break deterministic analysis. Analysis was performed for group distribution header, downcommer piping, and pressure piping connecting pressure header and group distribution header of Ignalina NPP unit 2. The aim was to calculate the maximal leak rates in acceptance with LBB requirements for all welds of analysed piping, and to analyse change of surrounding parameters in case of leak. Basing on analysis results, additional leak detection systems (as additional safety systems) are in installation process.

(4) Partuicipation in IAEA Extrabudgetary Programme on Mitigation of Intergranular Stress Corrosion Cracking in RBMK Reactors (2000-2001). The aim of the project was to help countries operating RBMK reactors to solve IGSCC mitigation problems using international experience. The general recommendations on remedial actions to be taken to significantly lower the risk for IGSCC are given, but each recommendation needs efforts to validate the remedial action before take into use.

(5) Risk Based Inspection Pilot Study of Ignalina NPP Unit 2 (2000-2001). It was a part of IAEA Extrabudgetary Programme on Mitigation of Intergranular Stress Corrosion Cracking in RBMK Reactors. The aim of study was optimisation of ISI program of 300 mm relative diameter piping produced from austenitic steel. The study showed, that it is possible to combine a 44% reduction of the number of future inspections with a 35% reduction of overall risk. This is possible due to a proposed shorter inspection interval for the high risk welds. In the higher risk levels, a shorter inspection interval than 4 years is suggested. Many low risk locations are suggested not to be included in the new ISI-selection. This means that the radiation exposure to plant personnel can be reduced and resources can be redirected to other safety related issues.

5. LESSONS LEARNED SOLVING IGSCC PROBLEM

(1) Welding technology has evident effect on IGSCC resistance of HAZ metal of welded joints made from steel 08Ch18N10T. The influence of welding technological factors (weld joint type, number of passes, welding regimes, welding methods and cooling procedure) is complex and each factor should be taken into account.
(2) Advanced non-destructive testing technique is necessary to found IGSCC cracks.
(3) IGSCC cracks in Ignalina NPP appearing at the inner surface in HAZ near weld root.
(4) There is no relation between length and depth of IGSCC cracks.
(5) Crack growth rate varies at separate stages of propagation because the HAZ metal sensitisation degree is different through a pipe wall.
(6) Water chemical composition should be neutral to IGSCC in all operation regimes.

Concluding it is necessary to mention, that the IGSCC phenomenon is currently under investigations and there is no yet comprehensive answer how to avoid this ageing issue.

ACKNOWLEDGEMENTS

The authors acknowleges administration and technical staff of Ignalina NPP for providing information.

REFERENCES

[1] IAEA extrabudgetary programme on mitigation of intergranular stress corrosion cracking in RBMK reactors. Working group 2 on comprehensive assessment techniques. Final report, IAEA (2002).
[2] Technical Report No. 66173 "Determination of Downcomer Piping Joints Damage Reasons, Development of Technological Measures and Experimental Evaluation of Damaged Joints Manufactured of 08Ch18N10T Steel Durability After Definite Operation and Repair Period", PROMETEY, Sankt-Peterburg 1997.
[3] BRICKSTAD, B., LETZTER, A., KLIMASAUSKAS, A., ALZBUTAS, R., NEDZINSKAS, L. and KOPUSTINSKAS, V., "Risk Based Inspection Pilot Study of Ignalina Nuclear Power Plant, Unit 2, Final Report", Report No. 10048500-4, Det Norske Veritas, Stockholm, 2001.

[4] ALMENAS, K., KALIATKA, A., USPURAS, E., Ignalinas RBMK-1500. A Source Book, Ignalina Safety Analysis Group, Lithuanian Energy institute, Kaunas (1998).
[5] IGNALINA NUCLEAR POWER PLANT, The Rules of the In-service inspection of the condition of metal of the equipment and pipelines of systems, important for safety of the Ignalina nuclear power plant with reactors RBMK-1500, PTOed-1125-B2, Ignalina NPP (in Russian).

Session 4
TECHNOLOGICAL AND OPERATIONAL ASPECTS OF PLIM

Sub-Session 4.3

Chairpersons

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NEW IN-CORE INSTRUMENTATION FOR ATUCHA I NUCLEAR POWER PLANT

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Abstract

In-core neutron flux monitoring assemblies were developed and constructed for Atucha I Nuclear Power Plant (CNA-I) in the Argentine National Atomic Energy Commission (CNEA) to replace the German original ones that had to be removed for maintenance reasons. Each assembly contains seven Self-Powered Neutron Detectors (SPND) in different measurement positions, protected with an external Zy-4 tube. Detector's mineral insulated wires pass the reactor pressure vessel (RPV) cover head through seven holes made in a removable steel-sealing plug. A braze welding is made between the plug and the wires to seal such holes. The SPND were totally developed and constructed in CNEA. Vanadium is used as neutron sensitive material, Inconel 600 as wire conductors and magnesium oxide as isolation. Instead of making the cable-detector junction by welding, in the present fabrication method, the electric contact between the vanadium and the central conductor is obtained by plastic deformation and the external sheath is a unique tube, covering both detector and cable zones. In this way the internal welding, that is a cause of failure, and the welding between the sheaths are avoided. The method is patented and allows obtaining integral cable-detector devices. The main advantage of these detectors is their greater reliability than the welded ones. Six assemblies have been supplied to CNA-I and one of them was installed in March of 2001 and is working with no failures up to now.

1. INTRODUCTION

As in most nuclear power plants, CNA-I uses in-core SPND to obtain information about neutron flux spatial distribution. Particularly this Plant has six long assemblies with seven detectors each one and two short assemblies with three detectors each one. In these assemblies detectors are located with their sensitive parts at different heights and the assemblies are located at different angular and radial positions. The original long assemblies, from Germany, had to be cut for maintenance reasons and it was necessary to replace them. So the company Nucleoléctrica Argentina Sociedad Anónima (NASA) hired CNEA to construct and supply these components.

The manufacture was divided into two big processes. One of them was the fabrication and testing of the detectors and the other was the construction and checking of the assembly.

a) To obtain the detectors, a first step was carried out to verify the construction feasibility and to validate the fabrication process [1]. In order to do this, a Prototype SPND was constructed for electrical, radiographic and neutron tests and a Simile SPND without vanadium was constructed to evaluate the evolution of dimensions and metallurgical properties during the plastic deformation sequence. In the second step the definitive detectors were fabricated according to the validated sequence. These sensors were electrically (continuity and isolation) and radiographic evaluated and a final neutron calibration was done [2].

b) To obtain an adequate welding between the detector cables and the sealing plug of the RPV cover head perforation, several trials were made with different conditions in the relevant variables to reach satisfactory and repetitive results. Qualification specimens were welded and evaluated. Leakage and metallographic tests were made. Finally the assembly welding was made and checked for helium leakage and high temperature hydrostatic test. The criteria to finish the rest of the assembly were to reproduce as much as possible the original design.

2. CONSTRUCTIVE METHOD

2.1. Detectors fabrication

A composite bar is assembled at the beginning of the process. Such bar is composed by an Inconel 600 external tube, an Inconel 600 central bar, a small tube of vanadium (neutron sensitive material) placed around the middle of the central bar and tubular MgO pellets, between the external tube and the central bar, as isolation. A sketch of the composite bar is shown in figure 1.

The pellets are made pressing MgO powder in a special matrix designed to get an annular geometry with the right dimensions and density. For this process an automatic hydraulic press is used. After pressing, the pellets are dried out at high temperature.

The composite bar is deformed by cold working with intermediate thermal treatments. The external diameter is reduced in several steps with the consequent increase in length. Between the initial diameter and the final diameter of the sensitive zone the composite bar is cold worked with a tube reducer. Then the cable zone is reduced with a cold drawer, specially adapted for that operation, and so the final diameter of the cable is obtained.

The intermediate thermal treatments are made in vacuum with a movable tubular electric furnace and so the heating is made in the zone where the furnace is passing. A sketch can be seen in figure 2.

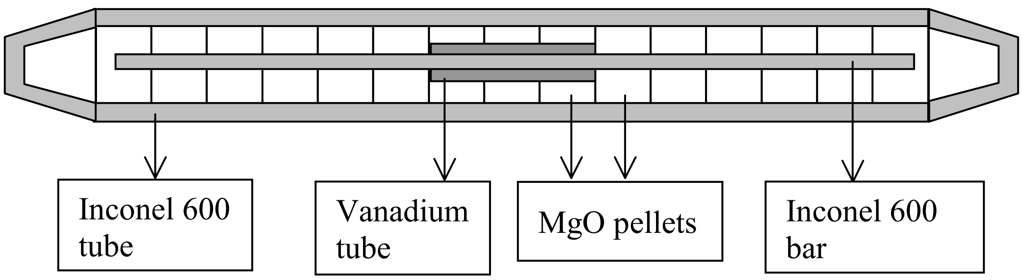

FIG. 1. Cross-sectional view of the initial composite bar.

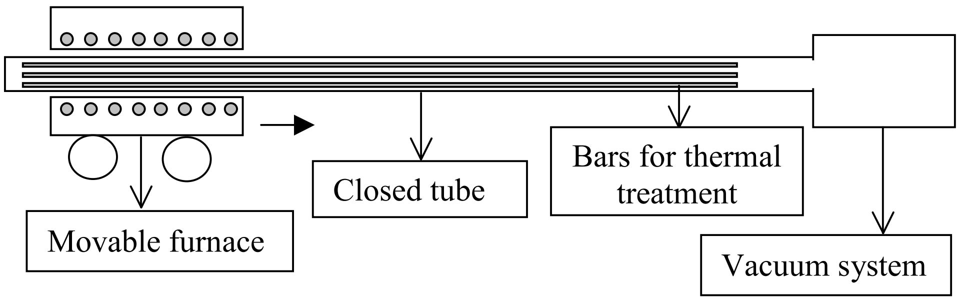

FIG. 2. Sketch of the system for bars thermal treatments.

When the final diameter of the sensitive zone is obtained, the vanadium ends are localized by external visual inspection and the bar is cut in the middle of these ends. So two detectors are obtained from only one bar. After cutting, the cable zone (without vanadium) is reduced by cold drawing operation, introducing the cable end through conical dies. Then the bar is pulled using a tensile load and the entire cable zone passes through the die. The drawing process is stopped at the beginning of the vanadium zone. Then the bar is extracted in the opposite direction of drawing. This reduction is made in several steps and the final diameter of the cable is obtained at the end of this sequence. A sketch of this process can be seen in figure 3.

This method is patented and allows obtaining integral cable-detector devices. The electric contact between the vanadium and the central conductor is obtained by plastic deformation and the external sheath is continuous with a conic reduction from the detector up to the cable. In this way the internal welding, that is a cause of failure, and the welding between the sheaths are avoided. The main advantage of these detectors is to have greater reliability than the welded ones. The small disadvantage is to have lower neutron sensitivity due to the fact that the neutron sensitive material, in this case vanadium, is a tube with an Inconel core and not a solid bar.

After the plastic deformation process the ends are sealed. Special care should be taken at the vanadium end seal because this end has to withstand the operational condition of the reactor core. To close this end a few millimetres of the materials inside the sheath are removed and a small Inconel cap is inserted in this place and welded with the sheath with TIG process. Figure 4 shows a sketch of the end seal before welding.

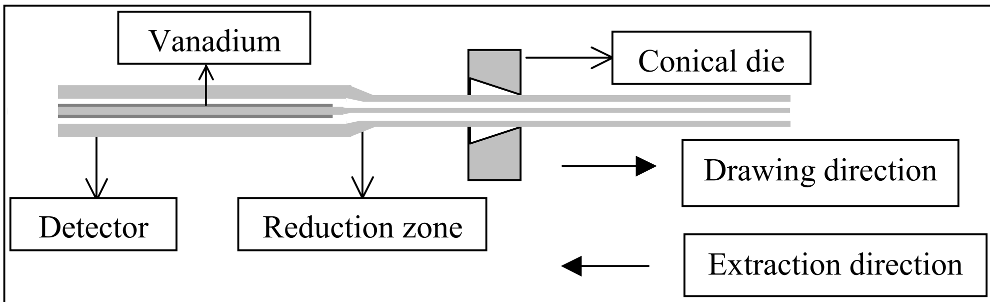

FIG. 3. Sketch of drawing process for integral detector.

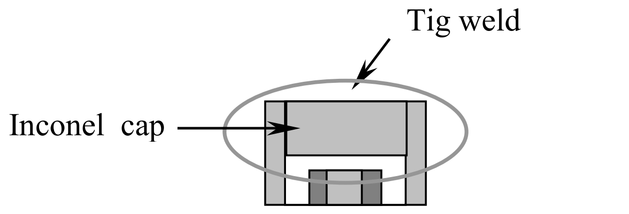

FIG. 4. Sketch of the end seal before welding.

The evaluation of this welding includes an hydrostatic test followed by an electrical isolation test and radiographic test. In the other end a temporary seal is made in order to make the electrical continuity and isolation tests. Finally, a neutron calibration is done in the RA-1 or RA-3 research nuclear reactors of CNEA.

2.2. Assemblies fabrication.

The main technological problem during the manufacture of the assemblies is the welding between the detector cables and the sealing plug of the RPV cover head perforation. This welding has to withstand high pressure and high temperature operational conditions and has to remain watertight to avoid losing of primary coolant (small LOCA). The welding process is brazing, so a filler material is introduced between cables and the sealing plug. During the welding only the filler material is melted and this liquid fills the space between the cables and the sealing plug obtaining a watertight junction. Heating is produced with an induction furnace using an adequate atmosphere of passing flux. The filler material used is a silver-copper-palladium alloy. A sketch of the system used to make the welding can be seen in figure 5.

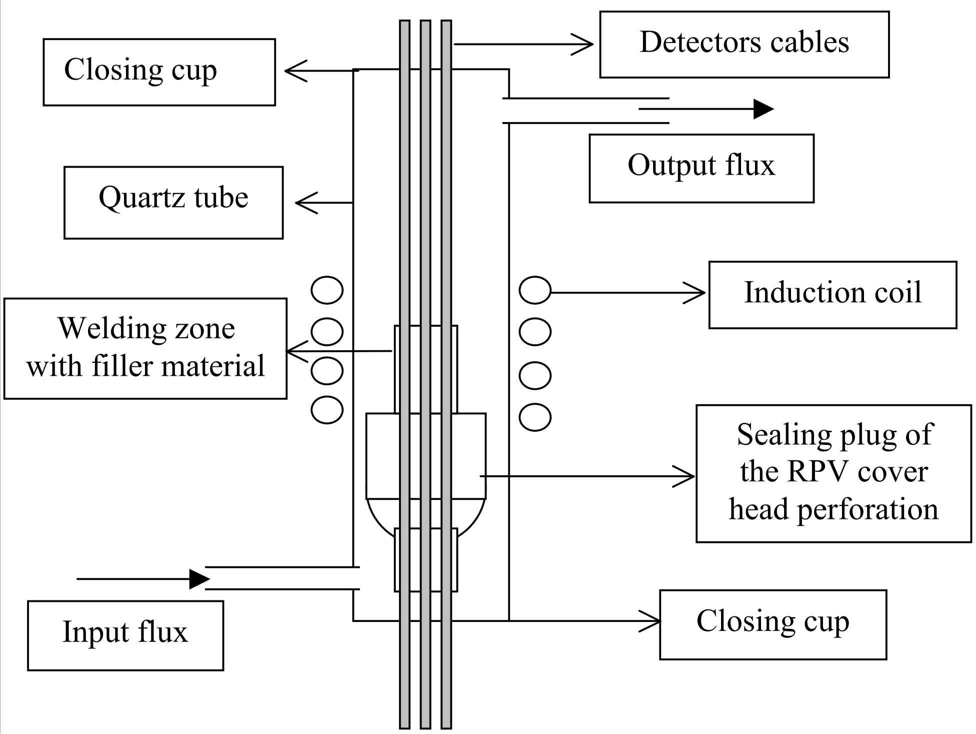

FIG. 5. Longitudinal cut of the system used to make the welding between cables and sealing plug.

During the construction of the assembly, the seven cables of the detectors are passed through seven corresponding holes made in the sealing plug. The relative positions between the sensitive zones and the sealing plug are fixed according the original diagrams in order to determine the correct axial position of the detectors inside the core. The filler material is put around the cables and over the sealing plug. The assembly is passed trough the quartz tube of figure 5 and the welding zone is fixed in an adequate position into the induction coil. The ends of the quartz tube are sealed with specially designed caps and vacuum is made. Then the quartz tube is pressurized with and adequate atmosphere with a small continuous flow. Finally the heating process is made in the welding zone with the induction coil. The quartz tube allows the observation of the melting and wetting of the filler material. In this form the moment when the process has to be stopped can be determined. Figures 6 and 7 show the initial mounting for the welding and the heating process respectively.

Helium leakage and high temperature hydrostatic tests are performed on the welded assembly. Figures 8 and 9 show photographs of these tests.

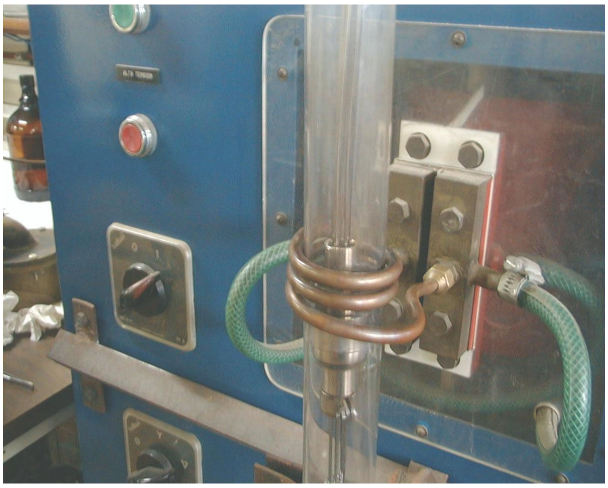

FIG. 6. Initial mounting for the welding.

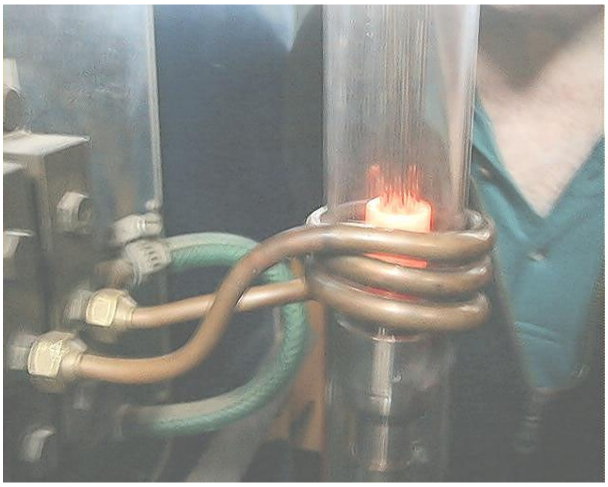

FIG. 7. Heating process

FIG. 8. Helium leakage test

FIG. 9. High temperature hydrostatic test

After these leakage tests the upper part of the detector is protected with a stainless steel flexible tube which is fixed to the upper part of the sealing plug, the cables ends are definitively sealed and connectors installed. Ceramic-metal feedthrouhgs are used to seal the ends of the cables. All the weldings used to seal these ends are helium leakage tested. Electrical isolation and continuity tests are repeated with the definitive connectors installed.

Finally, the part of the detectors that will be installed inside the RPV is protected with a Zy-4 tube. This tube is also fixed to the sealing plug. Additional electrical isolation tests are made with the detectors of the final assembly and if they satisfy, the assembly is released to be supplied.

3. RESULTS

The detectors passed the continuity and isolation electrical tests after hydrostatic immersion. The neutron response was adequate. Obtained sensitivities were compatible with the original design with a ±6% dispersion.

The first assembly installed in March of 2001 is working with no failures up to now. The seven detectors present good responses and this is an indication that there is electrical contact between the vanadium and the central conductor and that no humidity entered into the sensors. No primary coolant leakage was detected through the sealing plug of the RPV cover head perforation. This is a consequence of the cables-plug welding watertightness and the metallic contact watertightness between the plug and the RPV cover head. The montage of the assembly in the Nuclear Power Plant was made with no problems and this is an indication of a good compatibility with the original design.

The other five assemblies have already been supplied to CNA-I but they have not been installed yet.

4. CONCLUSIONS

It can be affirmed that, as a consequence of the developed work and the successful instrumentation provision, the Argentine National Atomic Energy Commission has the knowledge and the technological capability to construct and provide this kind of components or similar ones, according to a specific requirement. All this under strict safety controls and a Quality System matching ISO 9001. For the future, CNEA will continue the international diffusion, among nuclear reactor operators (PWR, BWR or HWR), of its capability to solve this aspect of NPP life management.

REFERENCES

[1] MILLER, M., BANCHIK, D., BIANCHI, D., FLORES, A., MARAJOFSKY, A., PÉREZ, L., SABIO, M., Validation Procedure for the Fabrication Sequence of Self-Powered Neutron Detectors for NPP Developed by CNEA, XIII ENFIR, Río de Janeiro, Brazil.

[2] MILLER, M., ESTRYK, G., ROQUETA, D., QUINTANA, J., AND ZARZA, L., Sensitivity and Linearity Measurements of Vanadium Self-Powered Neutron Detectors Made in CNEA, Argentina, 21° International Meeting on Reduced Enrichment for Research and Test Reactors, San Pablo, Brazil.

CANDU® STEAM GENERATOR AGEING MANAGEMENT: SOME PERSPECTIVES AFTER 20 YEARS IN-SERVICE EXPERIENCE

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Abstract:

AECL, in collaboration with utilities, has carried out several steam generator (SG) condition assessments to assure reliable remaining life and to provide a prognosis for extended life capability. With in-service dates ranging from 1972 to 1983, and a variety of tubing materials, the assessments have identified challenges for an effective SG aging management program. Factors found to limit SG life and to challenge aging management, are identification of critical degradation mechanisms for the key structural materials, evaluation and determination of the life-limiting sub-components, the need to develop improved inspection technology and tracking of life-limiting components, the need to create and use databases of plant information (in particular that from commissioning) and the development of recommendations for future operation and maintenance. At CANDU 6 plants, there has been an excellent SG service record with little or no significant active degradation to date and several of the utilities are actively pursuing the option of planning for extended operation. At three of the older CANDU 6 plants, detailed and comprehensive life assessment studies of the steam generating equipment, the interfacing systems, the external support structure, the tubing, and all the key internal sub-components, has been completed. Despite the excellent CANDU 6 SG record, it is well known that SGs provide challenges for the assurance of continued good health, as operation continues through to design life and for a significant period of extended operation. While the prognosis for life attainment and for extended operation of CANDU 6 SGs is good, it has also been found that this conclusion is dependent upon implementation of the recommended enhancements of inspections, maintenance, chemistry control and assessment of the future field data. This paper outlines the results of several CANDU SG condition assessments, and places the recommendations from these assessments in context with effective Aging Management programs.

1. INTRODUCTION

AECL, in collaboration with a number of utilities, has carried out several steam generator (SG) condition assessments with the objective to assure reliable remaining life and to provide a prognosis for extended life capability. With in-service dates ranging from 1972 to 1983, and a variety of tubing materials, the condition assessments have identified a wide range of challenges for an effective SG aging management program. All of these SG condition assessments were carried out in conjunction with overall plant condition assessments, performed either by AECL, the utility, or both in partnership.

Significant factors found to limit SG life and to challenge aging management are identification of critical degradation mechanisms for each of the key structural materials, evaluation and determination of the life limiting sub-components, the need to develop improved inspection technology and plans that track the life limiting components, the need to create and use databases of plant information (in particular data from commissioning), and the development of recommendations for future operation and maintenance.

Effective condition assessment and aging management (AM) requires a co-ordinated application of research and development (R&D) knowledge, plant or component design, and a thorough understanding of the operational history and the current plant programs including monitoring, inspection and maintenance related to aging. In fact, the R&D program at AECL has been restructured to provide tools and databases for condition assessment and AM. An example is the use of our knowledge of component design and function and system chemistry to interpret eddy current data obtained from SG tubing and predict primary side fouling behaviour. That knowledge, coupled with our understanding of heat transport and nuclear steam supply systems performance provides input not only for current condition assessment, but also for planning of remedial actions.

Several Canadian CANDU units have been placed in lay-up while condition assessments and rehabilitation programs are carried out. These rehabilitations are justified economically. The economics require careful evaluation of current condition and, especially challenging, of remaining life. Review of the current condition of the SGs that have been laid up demonstrates the importance of lay-up chemistry control practices, and the variability in degradation behaviour introduced by the different tube materials. Questions arising from the lay-up condition, and its impact on future life, are difficult to address quantitatively. Further R&D is required to define the linkage between lay-up condition and future aging degradation.

At CANDU 6 plants there has been an excellent steam generator service record with little or no significant active degradation to date and several of the utilities are actively pursuing the option of planning for extended operation. At three of the older CANDU 6 plants, detailed and comprehensive life assessment studies of the steam generating equipment have been completed, along with studies of the interfacing systems, the external support structure, the tubing, and all the key internal sub-components. The studies involved a very thorough assessment of tubing corrosion mechanisms that can occur in various forms in nuclear steam generators. CANDU 6 SGs are tubed with Alloy 800M (M means “modified”) and have experienced relatively little SG tube corrosion to date.

Despite this excellent record, it is well known that SGs provide challenges for the assurance of continued on-going good health, as operation continues through to design life and particularly for a significant period of extended operation beyond. While the prognosis for life attainment and for extended operation of CANDU 6 SGs is good, it has also been found that this conclusion is very dependent upon implementation of the recommended program enhancements of inspections, maintenance, chemistry control and assessment of the future field data. Subtle changes to plant operation and also the unique plant-specific details of materials and design often can have a significant impact on the life capability of this complicated equipment. From the plant-specific assessments, and the experience from our R&D program and utility interaction, AECL has developed a detailed proactive SG aging management strategy for long and reliable life capability (to 50 years). This proactive strategy has a number of important elements that are recommended to be incorporated into current plant programs.

With CANDU plants continuing operation with aging equipment, such as SGs, AECL is continuing to undertake comprehensive programs to support operations and provide enhanced technologies for performance attainment and improvement in inspection, surveillance and performance/safety analysis methodologies. These technologies are also continuously improved by programs that are targeted to important in-service degradation mechanisms, understanding of the system/equipment tolerance to this degradation, and to measures that can be applied to both new reactors and back fitted to existing plants. Systematic and continuous feedback of experience from operations to the performance programs is another important part of the approach.

This paper outlines the results of several CANDU SG condition assessments, and places the recommendations from these assessments in context with effective AM programs that need to be followed to achieve economic life. For CANDU 6 utilities, which have had a very good SG service record with little or no significant active degradation to date, the paper will outline the approach to proactive AM that will see equipment through to long and reliable extended life. Also mentioned are examples of AECL performance improvement and operations support programs for SGs.

2. SUMMARY OF THE CANDU SG CONDITION ASSESSMENTS

2.1. An Overview

Assessments of the SGs of several CANDU plants have been completed. In summary, the basic steps include:

(1) Identification of areas within the SG that require assessment;
(2) Identification of applicable degradation mechanisms and related stressors for each component in each area;
(3) Review of design, fabrication, installation, maintenance and inspection records;
(4) Review of operation records, including transients and chemistry data;
(5) Assessment of current condition;
(6) Assessment of life attainment (remaining life) and life extension options.

This comprehensive review leads to a set of recommendations that are required to achieve station life, and for maintaining the option for life extension. Note that station life and extended life may not be identical to the original design life (30 years for CANDU 6); for CANDU stations pressure tube replacement, the major life extension requirement, may occur from 25 years on, depending somewhat on economic factors. For Point Lepreau Generating Station (PLGS), life attainment is planned to be until 2008 (25 years) and life extension is until 2033 (an additional 25 years). For Gentilly-2 Generating Station (G-2) GS, life attainment currently is 2013 (30 years), with life extension to 2033 (an additional 20 years).

2.2. Summary of Results to Date

There has been very little degradation of CANDU 6 SG tubing or secondary side internals to date; most of the tubing degradation has been at PLGS and is a consequence of early condenser in-leakage and the use of phosphate chemistry. PLGS changed to all-volatile treatment (AVT) chemistry in 2000, and it is anticipated that there will be no further significant phosphate wastage. At all other CANDU 6 stations, the only SG tube degradation noted has been isolated shallow U-bend fretting. This experience is similar to that worldwide in PWR SGs tubed with Alloy 800. The early CANDU 6 SG primary side divider plates were floating segmented designs that proved susceptible to leakage through the gaps between the segments and the bolt holes holding the segments in place. At PLGS and G-2 these divider plates have been replaced.

The results of the condition assessments carried out for PLGS, Wolsong-1 and G-2 can be summarized as follows:

- the tube bundles and supports are expected to achieve 50 year life if good chemistry practices are maintained;
- the primary and secondary pressure boundaries are in good condition and expected to achieve a 50 year life;
- the replaced divider plates, and the original Wolsong-1 divider plate (an all-welded fixed design) are not expected to degrade significantly over the remaining SG life;
- the internal and external supports are in good condition and are expected to have a 50 year life capability and hence not to compromise overall SG life
- the condition of many secondary side internals is unknown; some of these are critical components, for instance the feedwater box and associated components. Many of these components are difficult to inspect. After reviewing the most critical components, a number were recommended for further inspections in order to better define current condition.

In addition, review of the SG inspection databases and reports indicates that they have not been designed with a life management focus. For instance, it is often difficult to track previous degradation or inspection results for a given tube, and relatively little attention has been given to correlating inspection data with other plant/component data relevant to SG aging degradation.

3. CANDU SG DEGRADATION AND LIFE PREDICTION

To a large extent the conclusions summarized in Section 2 depend on engineering judgement since there is no 50-year service experience with SGs, and particularly with CANDU 6 SG materials, for direct comparison. Current condition is dependent on the inspection program in place at any given station; typically this is most complete for the tube bundles and less so for the secondary side internals. Remaining life, in the absence of comparative in-service data, then depends significantly upon R&D knowledge and extrapolations.

The R&D emphasis at AECL recently has been on chemistry, degradation of tubing materials, fouling and inspection, and is currently focussed on the need to provide the extrapolations needed for life management and life extension assessments. Until recently the primary focus of the inspection program was to provide the tools required to ensure that CANDU 6 (and other CANDU) SGs can be inspected for all plausible degradation mechanisms, that the inspections are comprehensive and fast (for instance, X-probe development), and that the probes are qualified for all defects detected. More recently the technology has been developed further to provide information required for plant life management (PLiM), and be better integrated with the materials, thermalhydraulics and vibration/fretting work.

For both the tube bundle and the secondary side internals, the major factors related to life attainment and life extension are corrosion and fouling. In effect, fouling is a primary focus for any SG operator since tube bundle fouling impacts SG tube bundle thermal performance, support fouling impacts hydraulic performance and can impart high “lock-up” stress to the tubes, and the crevices between deposits and heat transfer surfaces (primarily the tubing) can develop chemistries highly corrosive to the contacting surfaces. Knowledge of SG fouling behaviour, and the ability to predict its impact on SG operation and life, is critical to a life management program. Hence it is evident that field experience and R&D knowledge must be integrated if life management is to be able to use this information effectively.

3.1. Integration of SG R&D with the CANDU PLiM Program

The SG is the interface between the primary and secondary systems at nuclear power stations. Consequently the SG R&D often has interfaces beyond the SG and the feedwater/steam supply system. For instance, a model has been developed to predict fouling in the primary side of the SG tubing, as a function of chemistry, temperature, hydraulics and time. This model, incorporated into the THIRST and SLUDGE codes, is also used to predict fouling throughout the heat transport system (HTS), and this can be used to monitor and predict system performance, as well as that for the SG, and to assess the impact of remedial actions. SG primary side fouling can be shown to be a major contributor to HTS system aging (decreased thermal performance with time), using the SG/HTS fouling models. The model then was used to show that removal of the SG tubing primary side deposits would result in a significant improvement. These predictions were verified by recent cleaning activities at G-2 GS and more recently, at Embalse GS.

Linked to this area of activity is the use of eddy current bobbin inspection data (see Section 4) to measure in-situ the deposit distributions on the primary side of the SG tube bundle, and so verify the model predictions as well as the efficiency of any cleaning process.

Similarly, inspection technology is being developed to provide the PLiM program with tools to predict SG tube fretting behaviour. It has been found that the tube-to-support gaps, measured in-situ, are critical to life predictions where fretting may be expected to be a concern.

The chemistry, thermalhydraulics, fouling and materials R&D has been integrated to provide the foundation for a monitoring and diagnostic chemistry control system (ChemAND), currently installed at G-2. This same technology also provides the PLiM program with important tools for predicting tube bundle life, and the impact of changes in chemistry control.

4. R&D IN SUPPORT OF SG TUBE BUNDLE LIFE

Typically it is tube bundle life that has limited SG life to date, and primarily this has been an issue affecting only those SGs with Alloy 600 tubing. CANDU 6 SGs are tubed with Alloy 800 (with Ti/C ≥ 12; so-called “modified” Alloy 800M), with no record of significant in-service corrosion degradation failures to date. Within the PWR community, the only other significant Alloy 800 SG tubing experience is in Germany, where some units have operated for more than 25 calendar years with no corrosion degradation. In order to be able to predict tube bundle life beyond this current in-service experience, R&D is being carried out in several areas. This R&D is briefly summarized below.

4.1. Predicting Corrosion Degradation of Alloy 800 SG Tubing

It is known that Alloy 800, similarly to all other SG tubing, is not immune to corrosion, particularly in the presence of aggressive contaminants such as Pb. Alloy 800 is also susceptible to under-deposit pitting in the presence of chlorides, and to phosphate wastage.

For CANDU SG tube life predictions some assumptions have to be made about potentially corrosive environments; it should be reasonable to expect that significantly off-specification chemistries will not be a chronic situation, but possibly occurring as short acute events. Thus electrochemical measurements were chosen as a screening method to establish the "safe" electrochemical potential (ECP) zones of operation, with the expectation that operation in these safe zones will provide 50 years' of economic service. Detailed results of this electrochemical work are reported elsewhere [1]. Electrochemical studies were carried out in various contaminated crevice chemistries and AVT water. Previous work has shown that Alloy 800 is not susceptible to crevice corrosion, pitting or stress-corrosion cracking (SCC) in normal AVT chemistry, either in tube-to-support crevices or in the free span [2], but that lead oxide inhibits the repassivation process in Alloy 800 following breakdown of the passive film; this can result in either pitting or SCC in significantly aggressive crevice chemistries. On the other hand, silica contamination appears to cancel the effect of lead and hinder the lead-induced corrosion. This type of information can be collected into a schematic showing acceptable (safe) and unacceptable zones of operation, as illustrated in Figures 1 and 2. This information is also being co-ordinated with the crevice/SG chemistry model (Section 4.2) to provide a qualitative guide for on-line monitoring and diagnosis.

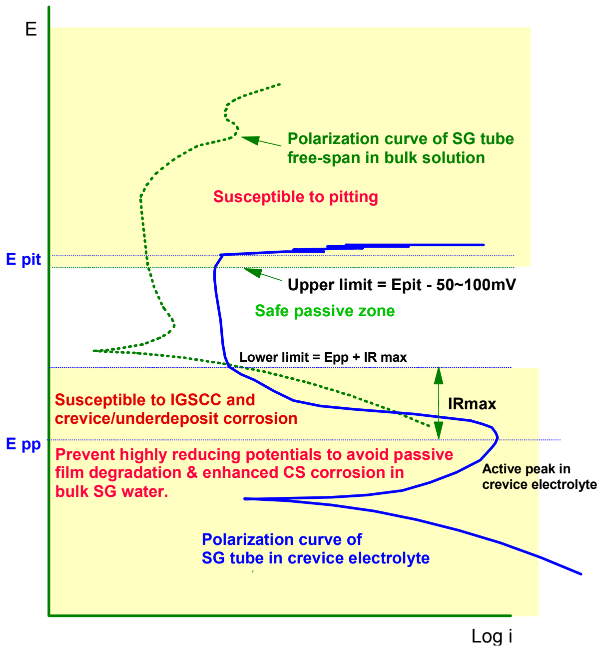

FIG. 1. Schematic presentation on how to determine safe potential zone for preventing corrosion degradation of SG tubing in crevice areas.

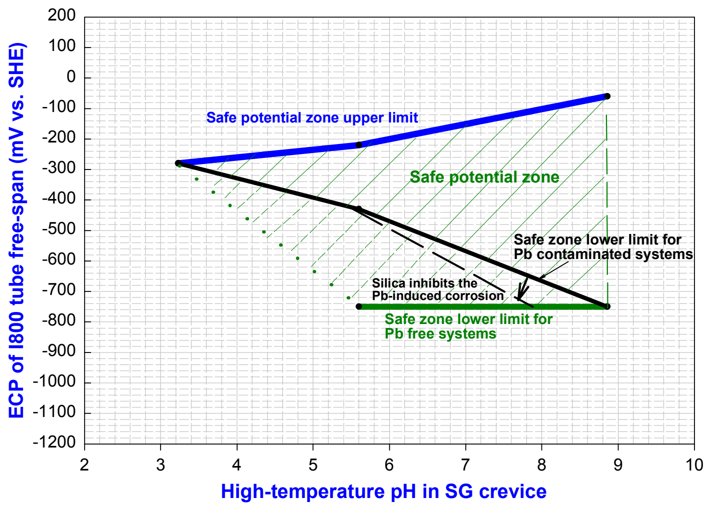

FIG. 2 Safe potential/pH zone for Alloy 800 SG tubing under CANDU full power operating conditions.

4.2. Crevice Chemistry Diagnostics for CANDU

Most of the SG tubing corrosion worldwide has occurred in crevices between the tubing and supports, the tubesheet or the sludge pile. At least some of the free-span tube corrosion can be ascribed to the presence of deposits on the tubes. It is known from experimental studies that impurities such as chloride and sodium can concentrate in SG crevices by many orders of magnitude over bulk water concentrations [3]. More recently, laboratory studies have been carried out in order to better correlate SG bulk water chemistry to that in crevices, including determination of crevice pH and ECP, and to provide updated justification for CANDU SG chemistry control. This experimental information is also used to validate the ChemSOLV code (ChemSOLV is a kinetics-based code for modelling SG and feedwater chemistry). ChemSOLV, which is incorporated into ChemAND, is used for on-line diagnosis of the crevice chemistry and subsequent corrosivity of current SG and feedwater chemistry to SG tubing.

The recent studies carried out at conditions expected under CANDU SG operation have provided an improved understanding of the factors determining crevice chemistry, and the required actions to prevent crevice corrosion of the tubing. In summary, hideout/hideout return studies of Na and Cl from Na Cl solutions show that oxidants responsible for the cathodic reaction in the crevice become depleted by local corrosion there, which lowers the potential of the corroding metal in the crevice. The normal flow of anions into the crevice to compensate for the electronic flow through metal tube from crevice site to the exposed area leads to a positive potential gradient between the crevice and the bulk solution. Consequently, anions migrate preferentially to the crevice during hideout, if oxidants are present in the bulk water, and lower the crevice pH. Hideout return, on the other hand, is diffusion controlled and the potential gradient between the crevice and the bulk, in the presence of oxidants in the bulk water, acts against the concentration gradient and slows the release of anions from the crevices compared to that of cations. This leads to increasing acidity of the crevice as long as the potential gradient exists. It was also found that the crevice chemistry is determined by the concentration of solutes in the bulk and by the heat input, and is not significantly affected by chloride volatility [4]. Thus it is crevice corrosion which affects the crevice concentration process most and leads to alkaline hideout return (because anion return is hindered by the crevice potential), not the presence of alkalinity in the crevices.

The major mitigating factor that can be used to prevent this crevice acidification and provide assurance for life attainment and extension is to ensure that sufficient hydrazine is present, both during operation and at shutdown. Hydrazine was shown to suppress the crevice corrosion, by suppressing the cathodic reactions outside the crevice. This helps ensure return of both anions and cations at the same rate.

4.3. Fouling Studies and Thermalhydraulics for Life Management

Over the past 9 years considerable R&D has been carried out at Chalk River Laboratories on the role of secondary side chemistry on SG fouling, including the use of dispersants and alternative amines to control the fouling. Much of the work over the past 5 years has been co-funded by EPRI. There have also been studies to determine the effect of various primary side and secondary side deposits on heat transfer, and of the application of various commercial and proprietary cleaning technologies, both mechanical and chemical, in removing these deposits and the subsequent impact on heat transfer. Studies have also been carried out on the deposition of magnetite, hematite, and lepidocrocite, the factors leading to deposit consolidation and on fouling of support structures. This work was preceded by studies to determine the physical and chemical basis for SG fouling, which resulted in development of the SLUDGE code to predict SG fouling as a function of chemistry, thermalhydraulics, time and location [5].

The impact of tube bundle aging, principally by fouling, is estimated using the SLUDGE and THIRST codes to model the thermalhydraulic response of the SG to fouling. The THIRST code predicts the SG thermalhydraulic behaviour under normal and abnormal conditions, including given or internally calculated scenarios of fouling. The SLUDGE code is able to predict the rate of fouling, areas most susceptible to fouling, and the distribution of deposit thickness on the tubes and supports. These results help in re-estimating the effective clearances between tubes and tube supports, which leads to the capability of being able to re-evaluate fretting-wear rates to take into account the impact of fouling of supports.

The most recent work has compared fouling rates and the rates of deposit consolidation for water chemistry controlled with so-called “alternative amines”, e.g., morpholine, ethanolamine, and dimethylamine. This work has demonstrated, in high-temperature loop tests, that fouling can be successfully mitigated (i.e., tube-bundle fouling rates can be lowered by up to a factor of 5), under conditions representative of CANDU SGs, by the use of appropriately optimized mixtures of amines, as illustrated in Figure 3.

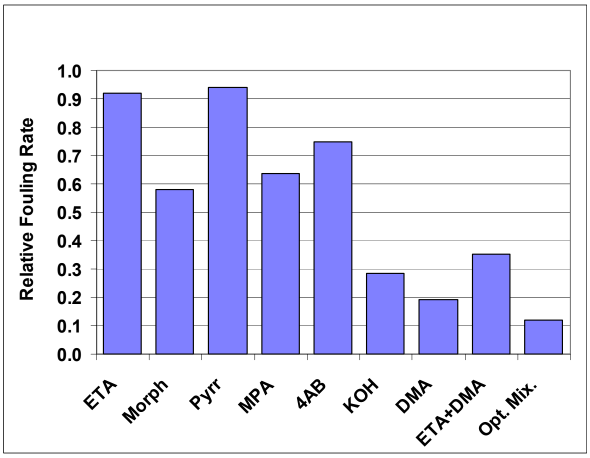

FIG. 3 Relative (to an arbitrary and historical fouling rate database) tube-bundle fouling rate of magnetite particles for various water treatment chemicals. ETA = ethanolamine; Morph = morpholine; Pyrr = pyrrolidine; MPA = methoxypropylamine; 4AB = 4 aminobutanol; DMA = dimethylamine; Opt. Mix. = optimized mixture of alternative amines (currently proprietary information).

4.4. Chemistry Monitoring and Diagnosis (ChemAND) for Life Management

The R&D knowledge briefly summarized above has been incorporated into a chemistry monitoring and diagnosis system called "ChemAND". This system has been installed at G-2 GS. Currently ChemAND monitors major station chemistry systems, but diagnostic routines are available only for the feedwater/SG system, including the primary side of the SG tubing. Although ChemAND was originally designed to improve chemistry operation and control, it is evident that it also has significant life management applications. This is a consequence of the diagnostic tools used in ChemAND, currently the THIRST, SLUDGE and ChemSOLV codes, and the chemistry and operations database generated by ChemAND. By tracking chemistry and operational parameters, assessing the impacts of any excursions on SG tube life, and monitoring systems fouling with time, an on-line aging assessment can be developed. The impacts of any remediation strategies, for instance SG cleaning, can be directly measured.

An example of the immediate value of ChemAND's diagnostic routines is the prediction that the SG crevices at G-2 GS were becoming corrosive during Summer months when the makeup water from the water treatment plant showed low, but significant, levels of chloride contamination. This chloride was residual from the chlorination of the intake water to control algal and bacterial growth that occurs in the summer months. The residual chlorine was predicted to be sufficient to reduce the crevice pH from being alkaline (normal winter condition) to being acidic in the summer months.

4.5. Vibration and Fretting Management

Fretting has not been a significant SG tubing degradation mechanism at CANDU 6 stations. However, there are some significant U-bend fretting issues at the Bruce B and Darlington GS units. Currently fretting is managed by inspection and plugging as required. Predictions of fretting-wear rate at CANDU stations are based primarily, and conservatively, on extrapolation of inspection data and on a linear rate of defect volumetric penetration. Fretting-wear rate modelling typically assumes as-designed clearances between tubes and supports, but it is becoming evident that as-built clearances sometimes differ from design.

Thus the R&D emphasis is moving towards an inspection method to measure tube-to-support gaps, and to develop vibration and fretting tools to predict fretting-wear locations and rates in as-built SGs, in particular focussing on the tube bundle U-bend area. For this purpose, the PIPO and VIBIC computer codes are used. The codes calculate the vibrational response of a tube in a given environment and the wear rate for the contacting tube. The calculated wear rate is based on experiments under realistic conditions for the contacting materials. The codes can also be used to estimate the effect of an ineffective support because of an increased clearance caused by corrosion or erosion.

4.6. Eddy Current Inspection and Analysis for SGs

In many outages SG tube inspection is a critical path activity and results are of critical importance to condition assessment. The AECL R&D in eddy current technology (ET) has traditionally focussed on developing fast degradation-specific probes for CANDU SGs, culminating with the X-probe, which can detect volumetric defects and axial and circumferential cracks at essentially the same speed of probe travel as the bobbin probes.

These probes are tested in the laboratory against laboratory-produced defects that account for deposits, supports, tubesheet joints and other artefacts that may compromise analysis, and compared with field experience and removed tube results.

The increasing reliance on eddy current inspection data for SG condition assessment and life prediction has led to a strong interface between the ET technology, materials technology, design engineering and PLiM/PLEX R&D activities. An example demonstrates the effectiveness of this approach.

CANDU 6 stations contain significant surface areas of carbon steel in the primary heat transport system (PHTS). This leads to deposition of magnetite in the core. As noted previously, a CANDU PHTS chemistry model was developed that predicts deposition throughout the PHTS, including the SG tube bundle. It has been noted for some time that SG primary side magnetite deposits generate a background eddy current signal. R&D on various simulated fouled tubes and careful comparisons of before and after primary side cleaning inspection data (PLGS, G-2 GS) led to a quantitative estimate of deposit distribution and thickness based on eddy current bobbin data and laboratory measurements of deposit density. This included data on both removed tubes and on laboratory-prepared tubes, and correlations were made with heat transfer resistance so that predictions of thermal performance could also be made. Figure 4 shows the data derived from the G-2 GS inspections before and after cleaning.

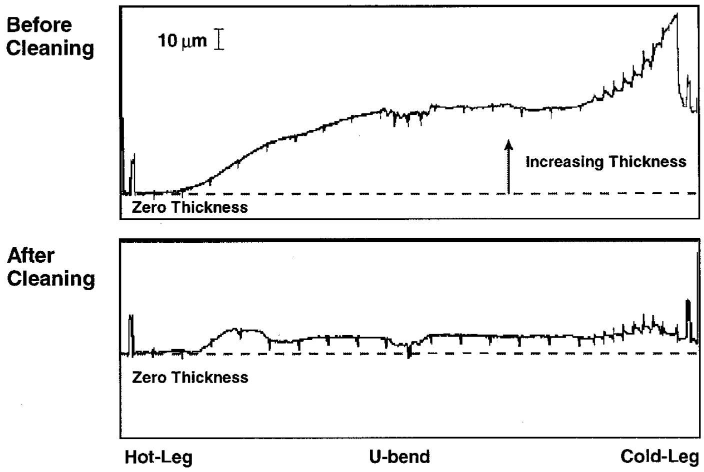

FIG. 4 Magnetite thickness before and after cleaning at G-2

4.7. Life Management R&D Relevant to SG Internals

As noted in Section 2, the primary side internals of the CANDU SGs, which are largely fabricated from carbon steels, with the exception of the older segmented and bolted primary divider plates, are not susceptible to corrosion in CANDU HTS chemistry, and to date no degradation has been noted other than in the older divider plates, which have been replaced or repaired. There is considerable relevant R&D underway at AECL focussed on non-SG HTS components, and this knowledge is available for SG applications.

Secondary side internals may be more susceptible to aging-related degradation. There is industry experience with flow-assisted corrosion (FAC) of a number of secondary side components, including separators, support plates and feedwater inlet structures. FAC diagnostic tools such as CHECWORKS can be used to assess FAC of some SG internal components and structures, and AECL has carried out laboratory tests under SG operating conditions to better quantify the FAC rates for various geometries and chemistries relevant to CANDU SGs.

5. IMPACTS OF SHUTDOWNS, LAY-UPS AND MAINTENANCE

Typically the impact of scheduled outages for maintenance and inspection are not thought to lead to conditions that might be significant in SG life. However, recent R&D results from AECL studies shows that high hydrazine concentrations (>10 ppm) under shutdown conditions can compromise the passivity of SG tubing. Thus any life assessment study must consider the impact of shutdown chemistries on SG life, especially tube life, and recommend appropriate procedures to minimize hydrazine concentrations without compromising the reducing chemistry required during the shutdown.

Lay-ups, defined here to be shutdowns longer than a normal inspection and maintenance outage, typically greater than ~30 days, are of greater concern. There are a number of examples world-wide of long lay-ups, very recently both Bruce Units 3 and 4 and Pickering Units 1 to 4 have been laid up for several years. Refurbishment is underway at both plants to restart these units.

Depending on tube material, long lay-ups can present significant risks to SG life, especially tube life. All SG tube materials (with the possible exception of the Alloy 400 tubing used for the Pickering A SGs) are susceptible to lay-up-induced degradation under oxidizing conditions, based on laboratory data and field experience. Of the commonly-used materials, Alloy 600 is most susceptible, particularly to sulphur-induced attack in the presence of air or copper oxides. Although lay-up-induced tube corrosion may not propagate under operating conditions (when chemistries revert to reducing conditions), there is little field or laboratory data available to be able to accurately predict the impact of such experience on future SG life. It is a significant challenge to maintenance staff to maintain non-oxidizing chemistries for extended lay-ups, for instance such as those required for retubing a CANDU reactor, but a challenge, if not met, which can significantly compromise SG tube bundle condition.

Related to the above discussion is the need to maintain clean SGs for predictable extended life, but also the concern that maintenance activities such as cleaning and inspection do not compromise SG condition. Included here are loose parts left in the SG, degradation of support structures by chemical cleaning solvents, etc. Implicit in any life prediction is the need to quantify the impact of chemical cleaning on design allowances. It is not always straightforward to obtain the "true" design corrosion allowances of all SG internal components, nor the actual corrosion allowances that can be assigned to chemical cleaning. These are essential data for life extension assessments, as well as accurate predictions of the need for future cleaning activity.

6. A SG AGE MANAGEMENT AND PERFORMANCE IMPROVEMENT STRATEGY FOR LIFE EXTENSION

Despite the excellent record of CANDU SGs at many plants, it is well known that the SGs will provide challenges for the assurance of continued good health for a significant period of extended operation. Many important secondary side internal components are very difficult to inspect and as a result, little is known about their current condition. Subtle changes to plant operation may have a significant impact on the tubing corrosion potential under deposits that have built up, and in crevices between the tubing and support structures. As an outcome of the SG work at a number of CANDU plants, it has been concluded that each plant and its SGs have unique aspects that could impact on life attainment or extended operation. The life assessment recommendations typically focus on specific aspects of chemistry control, proactive inspection and monitoring and periodic cleaning. While the prognosis for life attainment and for extended operation of CANDU SGs is good, it has also been found that this conclusion is very dependent upon implementation of the recommended program enhancements of inspections, maintenance, chemistry control and assessment of the future field data [6], [7], and is dependent on assumptions about the condition of uninspected components, particularly those on the secondary side of the SG. It is also important to consider that the SG is the interface between two systems, the HTS and the feedwater system, and that effective function of these depends on effective function of the SG for the life of the station. Conversely, life extension of the SGs requires that the HTS and feedwater systems operate without compromising extended SG life.

From the studies undertaken to date, a typical proactive SG age management strategy for life extension would include the following elements.

6.1. Enhanced Tube Bundle Inspection/Interpretation

In recent years, there have been considerable advancements in tubing eddy current testing technology and also much better knowledge of tubing degradation mechanisms and which EC inspection techniques can be best used for detection. Also, improvements in analysis and interpretation of eddy current data, and use of these data for predicting early signs of tubing degradation have been developed. A proactive SG aging management program uses the results of the life assessment work, couples it with these advanced inspection and interpretation techniques, and then develops an enhanced SG tubing inspection program for plausible tubing aging degradation. The objective is to have early-as-possible identification of any possible tubing degradation by focussing inspection effect on the "age-sensitive" regions of the tube bundle with appropriate techniques capable of detecting the plausible degradation. Typical examples of information not previously available from eddy current inspection, but now available, as noted in Section 4, is quantification of the depth and extent of deposits on the tubing primary side, and detection of tube-to-support gaps for use in vibration and fretting-wear assessments.

6.2. SG Surveillance Tubes

An important proactive life management technique in many CANDUs is a program of regular tube removal and subsequent metallurgical evaluation. Such examination of removed tubes is a requirement of the Canadian Standards Association. This examination is an important supplement to the NDE inspections and provides a more sensitive confirmation of tube wall condition and an insight into the local operating environment on the tube surface. This is particularly useful for the secondary side tube surfaces that have been exposed to under-deposit conditions (such as in tubesheet sludge piles).

6.3. Secondary Side Internals

The importance of many key internal components to successful long-term operation of the SGs, and the relative lack of information on in-service condition, is a typical outcome of the life assessment work. A detailed risk assessment of these non-tube components based on design function and operational experience, leads to identification of those specific internals that should be subject to inspection in a proactive and comprehensive age management program for life extension. As noted earlier, there have been several instances, both in CANDU SGs and those of PWRs, of degradation of internal structures resulting from flow-assisted corrosion. This experience indicates that the secondary side internals are an area where some inspection for component integrity is essential for assurance of life extension.

6.4. Secondary Side Crevice Conditions

Most plants place considerable emphasis on controlling the operational chemistry of the secondary side so that it is consistently within acceptable levels. In the SG, tubing life is directly related to local chemistry conditions at the tube secondary side surfaces. In the crevices at tube supports and in the tubesheet sludge pile region, successful long life requires maintaining the crevice chemistry within ranges that minimize tubing corrosion. Subtle changes that can significantly affect tubing life may result from variations in feedwater impurity. If these impurity fluctuations are within normal bulk water specifications, their potential to increase tubing corrosion damage, particularly in secondary side SG crevices, could go unnoticed until extensive damage becomes evident.

To assess crevice conditions for corrosion damage potential with given operational chemistry parameters, plant staff need knowledge of the local chemistry in the SG tube bundle crevices, in addition to the bulk chemistry of the water surrounding the tubing. In the past this was a rather difficult and time intensive task that could only be done by chemistry and corrosion experts not usually found at the plants. However, recently, tools have been developed to provide a CANDU SG crevice chemistry prediction that can be used by the plant operator on-line. The effects of impurity in-flows to the secondary side water, on local crevice chemistry and fouling in the steam generator are identified and, where of concern, flagged.

This type of on-line monitoring and prediction system gives the plant operator an important life management tool for maintaining good SG health and for attaining long life, by providing early indication of any change in chemistry parameters that could result in damage to the SG tubing. Ready on-line access by the operators to current and past chemistry conditions, including chemistry predictions in the critical crevice regions of the SG, enables appropriate responses while on-line (diagnosis of any change in corrosion susceptibility) and planning of shutdown maintenance actions (such as inspections to verify local conditions, and the need for cleaning specific areas).

6.5. Proactive SG Cleaning Program

Even with the best secondary side water chemistry control, corrosion products from the secondary side systems will continue to be carried into the SG tube bundle during its lifetime. A large percentage of these corrosion products come to rest in the SG as deposits, mostly on the tube surfaces. Those deposits that end up on horizontal surfaces, and particularly the tubesheet, can be difficult or impossible to remove while the SG is operational. While considerable effort is made to maintain good bulk water chemistry in the SG itself, and advanced chemistry control methods are now available to reduce such fouling, these deposits, if allowed to accumulate, become hard (consolidated) and create crevice conditions at the tubesheet-to-tube interface, as well as fouling the tube-to-support gaps. Feedwater impurities diffuse to these crevices and, as a consequence of boiling in the crevice, can concentrate by factors of up to 10^6. Hence, a proactive aging management program for SGs should include secondary side cleaning (particularly tubesheet flushing or lancing), with regular application, even before the presence of significant SG deposit is detected.

Although there is considerable world experience to support this activity, plant operation and maintenance staff sometimes question the benefits of cleaning because of the considerable costs involved. To provide some enhanced tools (and the science behind them) that will aid cleaning decisions, AECL is further developing models to predict tubing corrosion damage using a variety of laboratory and field data.

7. AN INTEGRATED APPROACH: HEAT TRANSPORT SYSTEM PERFORMANCE IMPROVEMENT

As noted in Section 6, the SG is a critical interface with the HTS, and hence impacts the generation of electricity as well as life extension considerations for the HTS.

Analysis of the combined effects of aging of these two systems is necessary to ensure that the plant is operating within the original design envelope, to demonstrate that there has been no deterioration of the operational or safety margins, and to ensure mitigation methods are effective in managing aging.

When the first CANDU 6 stations were originally designed, AECL recognized the need for a detailed thermalhydraulic predictive modelling capability for the CANDU Heat Transport System (HTS). A number of aging mechanisms were anticipated and compensating margins were provided to cater for the in-service aging degradation that would occur. Accurate predictions of HTS thermal and hydraulic parameters were recognized as an important capability. Hence, predictive codes were not only developed but also extensively validated with both commissioning and operational data from the early CANDU 6 experience.

This program of code development and refinement for prediction of HTS aging behaviour has continued throughout the operational period in cooperation with the utilities with particular attention on assessing the impact of primary side fouling, and the effect of such fouling on the thermalhydraulics of SG tubing, fuel and other system piping surfaces. Also, in parallel, the AECL R&D program systematically addressed the understanding of the thermal and hydraulic behaviour of deposits that accumulate on various surfaces in the HTS, including the primary side of the SG tubing. Combined with utility data and reactor specimens provided for laboratory analysis at our Chalk River Laboratories, tube specimens from several in-service SGs were examined to gain in-depth knowledge of the HTS deposits on the internal diameter (ID) surface of the tubes. Experiments were performed at CANDU HTS conditions to measure deposit thermal resistance. Additional hydraulic experiments and measurements were conducted to determine the effect of deposit roughness. This effort has provided additional important data and modelling parameters that have been subsequently incorporated into the prediction codes result in an enhanced predictive capability for HTS aging.

As noted earlier, an eddy current method has also been developed to measure the extent and distribution of SG tube primary side fouling using this capability, it is now possible to predict the effectiveness of a cleaning process, and, in conjunction with the HTS performance models, to assist operators, maintainers and technical staff in planning when such cleaning should be carried out. This development has been applied in the field at several plants and is providing valuable field data on deposit behaviour. Recent primary side mechanical cleaning at G-2 GS restored ~+5% of core flow and ~-3ºC in reactor inlet header temperature (RIHT) values which were very close to the improvements predicted using the NUCIRC code and the data developed from the work noted above. Use of these refined codes cannot only accurately reflect the current HTS condition of the plant, but can also be used as an important PLiM technique to predict the benefit of aging mitigation processes.

8. CONCLUSIONS

The objective of the CANDU utilities and AECL is to maintain the plants as a safe, economic and reliable means of electricity production through their design life and to preserve their capability for extended life operation. A comprehensive and integrated PLiM program has been developed and is in the process of being implemented at several CANDU stations.

The CANDU PLiM program is already helping other reactor owners achieve their goals for safe, economic and reliable life attainment and to preserve the option for extended operation. It has provided CANDU utilities with in-depth assessments and promising life prognosis for the key critical components, structures and systems in the plant. This has been an important input into Canadian utility decisions to embark upon a detailed CANDU 6 plant extended operation program.

To date, the SGs in the CANDU 6 units have had a very good service record with essentially no significant active degradation reported. In performing the systematic and detailed life assessment of this critical equipment, a key activity is diagnosis of the operational history for aging indicators, as well as a thorough understanding of applicable degradation behaviour. With little degradation data, a heavy reliance is placed on the thorough understanding of the applicable degradation mechanisms and the associated “stressors”. This understanding derives from research and development programs, integrated with knowledge from relevant field data of other plants.

REFERENCES

[1] Y. Lu, M.D. Wright and R.D. Cleland, “Effect of Lead and Silicon on Localized Corrosion of Alloy 800 in Steam Generator Crevice Environments”, Proceedings of 10th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, 2001 August 5-9, Lake Tahoe, Nevada.

[2] Y.C. Lu, “The Influence of Hydrazine on the Cathodic Kinetics and Localized Corrosion of Incoloy 800 Under Steam Generator Lay-up Conditions, Proceedings of 8th International Symposium on Passivity of Metals and Semiconductors, Jasper, Alberta, 1999 May 9-14.

[3] P.V. Balakrishnan, “Crevice Concentration Studies Using Radiochemical Techniques, presented at the Seminar on Crevice Chemistry, Studsvik, Sweden, STUDSVIK/M-96/151, SV/STUDSVIK-M52:5, 1996 October 21-22.

[4] P.V. Balakrishnan, “Hideout Return and Crevice Chemistry, 2001 EPRI PWR/BWR Plant Chemistry Meeting, San Antonio, USA, 2001 January 9-12.

[5] C.W. Turner and S. Yiang, “Deposition Simulation in CANDU and PWR Steam Generators using ATHOS and SLUDGE”, Proceedings of Steam Generator Sludge Management Workshop, Myrtle Beach, N.C. 1996.

[6] J.H. Nickerson, M. Soulard, R.L. Tapping and J.R. Hopkins, “CANDU Plant Performance Improvements and Life Management, 2000 KAIF Conference, Seoul, Korea, 2000 March.

[7] R.M. Lepp, C.A. Kittmer, R.A. Speranzini and R.L. Tapping, “R&D Programs in Support of CANDU Plant Life Management”, presented at 1998 ASME PVP Conference, 1998 July 27-21, San Diego, California.

AGEING COUNTERMEASURES FOR NUCLEAR POWER PLANTS IN JAPAN AND PLEC's ACTIVITIES

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Abstract

This paper summarizes an outline of research on nuclear power plant aging in Japan. It also provides an outline of the aging related research projects currently conducted by Japan Power Engineering and Inspection Corporation (JAPEIC) under the auspices of the Ministry of Economy, Trade and Industry (METI), and provides application examples of the results of these research projects. The activities of the Nuclear Power Plant Life Engineering Center (PLEC) are also summarized in this paper.

1. INTRODUCTION

As of March 2002, 52 commercial nuclear power plants have been operating in Japan. Three (3) plants constructed in the early 1970s have been operated for more than 30 years. Twenty-two (22) plants, including the above three plants, have been in operation for over 20 years and represent about 35% of the gross capacity of 45.7 GWe. Thus, aging management has become a matter of increasing concern in Japan.

The regulatory bodies of Japan have, therefore, endeavored to enhance the reliability and safety of aged nuclear power plants. The details are described in the paper entitled ‘Current Approaches to Nuclear Power Plant Life Management in Japan (IAEA-CN-92/61)’ for this symposium [1].

Regarding recent technical evaluations of aging, the electric utility companies conducted technical evaluations in their periodical safety reviews concerning the aging phenomena of all safety related components and structures. From the viewpoint of aging management, the companies identified additional maintenance measures to reinforce their current maintenance procedures and summarized these as their long-term maintenance programs including the current and additional maintenance activities. Figure 1-1 shows an example of aging phenomena of the major components and structures for evaluation in PWR nuclear power plants.

2. AGING RELATED RESEARCH AND DEVELOPMENT PROJECTS

2.1 Outline

The report issued by the regulatory bodies in April 1996 addressed the importance of research and development to enhance the reliability and safety of aged nuclear plants as described in reference [1].

Consigned by regulatory bodies, JAPEIC has been conducting aging related research (see Figure 2.1-1). These research are composed of three areas including ‘evaluation methods for aging’ concerning aging phenomena such as ‘fatigue’ and ‘irradiation embrittlement’, ‘inspection and monitoring’ for the detection of flaws such as stress corrosion crack, and ‘preventive maintenance and refurbishment’ for treatment of material surface improvement and repair welding. The results of these research projects will be applied to methods and criteria for the technical evaluation of aging phenomena.

An outline of the research projects concerning the aging phenomena of reactor pressure vessels is described below.

2.2 Fatigue

‘Fatigue’ is one of the most important aging phenomena evaluated for components and structures, including reactor pressure vessels. In the Environmental Fatigue Test of Nuclear Power Plants Materials for Reliability Verification Project (EFT Project), the fatigue tests for carbon steel, low alloy steel and austenitic stainless steel under simulated water environment conditions of the light water reactor (LWR) have been conducted [2], [3], [4]. Figure 2.2-1 shows an example of the results of the fatigue test for carbon steel. Based on the results of fatigue tests under simulated LWR environments, the EFT Project developed tentative evaluation methods for evaluation of reduction of fatigue initiation life [2]. These methods were issued as a guideline for assessment of the environmental fatigue damage of aged LWR nuclear power plants by the regulatory authorities. The guideline is applied to the fatigue evaluation of components and structures on the technical evaluation in the periodic safety review conducted by the electric utilities. Examples of the methods are shown in Figures 2.2-2 and 2.2-3. In order to verify the validity of the tentative methods, additional fatigue tests will be conducted for another several more years.

2.3 Neutron irradiation embrittlement

Regarding the effects of neutron irradiation on reactor pressure vessel steel, there is two measures in tendency of shift in ductile-brittle transition temperature at the transition energy region and decrease in fracture toughness at the upper shelf energy (USE) region after neutron irradiation.

The evaluation methods for integrity of components and structures at transition temperature region and prediction equation for degree of transition temperature shift were already standardized in the Japan Electricity Association Code JEAC4206 (1991) based on the results of the Pressurized Thermal Shock Test Project (PTS Project) [5], [6]. This standard is applied for the technical evaluation of neutron irradiation embrittlement at transition temperature in PWR plants.

Regarding evaluation of the upper shelf energy region, the JAPEIC has been conducting the Nuclear Power Plant Integrated Management Project (PLIM Project) which aims at developing the prediction equation of USE reduction and establishing the equation of correlation between USE and fracture toughness values by using various heats of test materials equivalent to actual RPV materials, (see Figure 2.3-1) [7], [8]. Regarding fracture toughness monitoring for reactor pressure vessels, the PLIM Project also conducts reconstitution of the surveillance test specimen by using irradiated materials to establish the applicability of the methods. (Figure 2.3-2) [9], [10].

Another project is the Structural Assessment of Flawed Equipment Projects (SAF Project) in which vessel model tests were conducted to confirm consistency between the experimental results and the results of elastic-plastic fracture mechanics analysis for verification on the integrity of the vessels [11], [12], [13], [14]. The vessel model test in the SAF Project was to confirm ductile crack growth and propagation behavior in tested materials with lower upper shelf energy values around 50J). The safety margin for actual equipment was also evaluated by using small and large vessel test models providing small flaws in the pressure vessel with lower USE value. Figures 2.3-3 and 2.3-4 show the test equipment employed and the test results respectively.

2.4 Stress corrosion cracking in nickel based alloy

Recently, it is reported that incidents of stress corrosion cracking (SCC) in nickel based alloy welds had been observed in portions including PWR reactor head penetrations, PWR nozzle safe end welded joints, BWR control rod drive (CRD) stub tubes, and BWR shroud support structure.

Regarding the inspection and monitoring technology for SCC in nickel based alloy, the Nondestructive Inspection Technologies on the Nickel Alloy Welded Joint Project (NNW Project) was started this year. The objectives of the Project is to verify the application of inspection technology for flaw detection and sizing performance in ultrasonic testing (UT) and eddy current testing (ECT) of the portions, thereby improving the safety and reliability of aged nuclear power plants (see Figure 2.4-1). The investigations into the following subjects related to nickel based alloy welded joint are to formulate the plan to develop the test conditions;

(a) Inquire of the plant data (welding materials and design dimension for welded joint)
(b) Surveillance of the failure incidents
(c) State-of-the-art technology for fabrication of test assembly with natural flaw
(d) Research for the ultrasonic wave propagation characteristics and electromagnetic characteristics in the weldment
(e) Verification for nondestructive inspection technologies for the nickel based alloy weldment

In the case of aging evaluation technology, the Evaluation Technology for Stress Corrosion Crack Growth of Nickel Based Alloy (NiSCC Project) is being conducted. The purposes of the project are to systematically obtain the data of SCC growth rate for nickel based alloys under the simulated environments of LWR and to contribute integrity evaluation for major components with nickel based alloys for the reactor coolant pressure boundary (see Figure 2.4-2). The tests for SCC crack growth rate are currently conducted under constant loading conditions and cyclic loading conditions.

Failure incidents were observed in the components with complicated configurations of structural discontinuity or bimetal weld joints. It is important to establish Ni-SCC crack growth evaluation methods and to verify the integrity of these components. The project for the Integrity Assessment of Flawed Components with Structural Discontinuity, IAF Project, aims at establishing and verifying evaluation methods for the residual stresses and crack growth behavior of these components during the in-service period (see Figure 2.4-3).

3. PLEC ACTIVITIES

3.1 Mission and Roles

For the implementation and promotion of aging-related technology research, the Nuclear Power Plant Life Engineering Center (PLEC) was established in April 2000 within JAPEIC. The PLEC, that specializes its mission and role for aging, is assigned four items as follows:

(a) Coordinate planning of technology research on aging in cooperation with industry, government and academia.

(b) Integrate technology knowledge of aging for management and assessment.

(c) Promote practical application of technical research results

(d) Promote the sharing of technology information.

The PLEC will perform these roles in a neutral and transparent manner, independently from other JAPEIC activities to secure its independency and specialty.

3.2 Organization for PLEC Activities

Focusing on the aging issues from a broad perspective, a Technology Advisory Committee was organized to advise on the PLEC's activities. Figure 3.2-1 shows the organization for PLEC activities schematically. The members consist of professors, specialists and experts from the electric utilities, research institutes and manufacturers, and representatives of the prefectural and municipal governments relevant to nuclear power plant sites. This committee meets in open session. Under the leadership of the technology advisory committee, the specialized subcommittees on Technology Research and Technology Application are organized.

3.3 Planning of research on aging in cooperation with industry, academia and government

It is important to systematically promote research for various factors such as aging events, systems, structures, components and materials, from a long-term viewpoint. The PLEC is engaged in the planning and coordination of aging related research in cooperation with the government, academia and industry (see Figure 3.3-1).

3.3.1 Preparation of map for aging research

Technical survey for research and development related to aging countermeasures are currently conducted and summarized in a type of matrix table that lists the existing research related to aging countermeasures of 'evaluation method for aging', 'inspection and monitoring' and 'preventive maintenances and refurbishment' v.s. aging phenomena of 'fatigue', 'neutron irradiation embrittlement', 'stress corrosion crack', 'thermal aging', 'corrosion' and so on. These matrix tables identified major structural materials and parts in which these aging phenomena have the potential to occur. Figure 3.3.1.1-1 shows a format example of the technical map.

Using these matrix tables, the PLEC evaluates and identifies the research items necessary in the future from the viewpoints of safety regulatory aspects and facility maintenance operation management. The PLEC will check this map annually using the latest knowledge and the results of research and development.

3.3.2 Developing medium and long-term perspective

It is important to carry out the research effectively, efficiently and in a timely manner from a medium and long-term perspective, under the restriction of resources and period. The basic concepts are as follows;

(a) To establish the medium and long-term perspective and plan based on 'technical development items identified from the technical map', 'needs for codes and standards necessary for aging countermeasures' and 'trends of technology related to aging countermeasures domestically and overseas'.

(b) To check the contents of the medium and long-term perspective and plan in order to use the latest knowledge.

3.3.3 Identification and evaluation of aging related research

On surveying, evaluating and identifying a new candidate, the PLEC is making every effort to survey the needs and to thoroughly utilize the technical knowledge of concerned organizations including research organizations, universities, electric utility companies and plant manufacturers engaged in research and development. The identified candidate themes in relation to verification tests are as follows;

(a) Research on Nondestructive Inspection Technologies on the Ni Based Alloy Weld Joints

This research project was started this year as the NNW Project. The outline of the project is described in paragraph 2.4.

(b) Research on fatigue crack growth rate under a simulated reactor water environment

This research is to examine fatigue crack growth rate tests for stainless steel under a simulated PWR reactor water environment to establish fatigue crack growth rate curves for the parameters including stress ratio, frequency and temperatures and to reflect the curves on the related codes and standards.

(c) Research on advanced technology to evaluate environmentally assisted crack growth rate under simulated stresses in nuclear power plant components

It is recognized that the stress corrosion crack growth rate for nickel based alloys corresponds to the data of stress intensity factor (K)-value increasing mode under the constant load tests. It is, however, vital to use the data of the K-value decreasing mode under the constant displacement tests in order to conduct the appropriate evaluation of crack growth rate for actual components. The research is to examine SCC crack growth rate tests for nickel based alloys by constant displacement tests, to develop crack growth rate diagrams and to apply them to related codes and standards.

4. SUMMARY

The number of nuclear power plants operating more than 30 years has increased. Various aging related activities have been performed in Japan. To assure and improve the safety and reliability of aged nuclear power plants, it is important to continue and conduct research activities for evaluation methods of aging, inspection, monitoring, preventive maintenance and refurbishment, in an effective manner and to apply the results of these research to aging management strategy.

REFERENCES

[1] T. Noda et. al., 'Current Approaches to Nuclear Power Plant Life Management in Japan', IAEA-CN-92/61, IAEA International Symposium on Aging Management of Nuclear Power Plants, Budapest, Hungary, 4-8 November 2002.

[2] K. Iida et. al., 'Comparison of Japanese MITI Guideline and Other Methods for Evaluation of Environmental Fatigue Life Reduction', Proc. of the ASME Pressure Vessel and Piping (PVP), July 22-26, 2001

[3] M. Higuchi et. al., 'Effects of Strain Rate Fluctuation and Strain Holding of Fatigue Life Reduction for LWR Structural Steels in Simulated LWR Water', Proc. of the ASME Pressure Vessel and Piping (PVP), July 22-26, 2001

[4] A. Hirano et. al., 'Effects of water Flow Rate on Fatigue Life of Carbon Steel in Simulated LWR Environment under Low Strain Rate conditions', Proc. of the ASME Pressure Vessel and Piping (PVP), July 22-26, 2001

[5] S. Ishino et. al., 'The effect of chemical Composition Irradiation Embrittlement', Nuclear Engineering and Design 119 (1990) pp 134-148.

[6] M. Mishima et. al., 'PTS Integrity study in Japan', Int. J. Pres. Ves. & Piping 58 (1994) pp 91-101

[7] K. Sakamoto et. al., 'Development of Evaluation Method on Integrity of RPV at the Upper Shelf Region', ASME PVP-2002.

[8] E. Otsuka et. al., 'Development of Aged Nuclear Power Plants Management Technology in Japan Power Engineering and Inspection Corporation', PLIM+PLEX'99.

[9] K.Sakamoto et. al., 'Development of the plant life management technology for RPV steels [-Current Status of Surveillance Test Specimen Reconstitution Program-]', ICONE-9.

[10] Y. Atago et. al., 'Development of Reconstitution Technology for Surveillance Specimens', ICONE-10.

[11] K. Hosaka et. al., 'Structural Assessment of Flawed Equipment for Nuclear Power Plants' Maintenance Criteria', Japan-Germany Seminar, 2001

[12] K. Hosaka et. al., 'A Ductile Crack Growth Behavior of A Pressure Vessel with Low Upper Shelf Energy', ASME PVP, 2001

[13] K. Hosaka, 'Structural Assessment of Flawed Equipment Project (SAF)', Houkoku To Kouen No Kai (original in Japanese, Proceeding of the 14th JAPEIC Seminar), Oct. 2001

[14] K. Hosaka, et. al., 'A Ductile Crack Growth Behavior of A Pressure Vessel with Low Upper Shelf Energy (original in Japanese)', M&M Seminar of JSME, 7-8 October 2000.

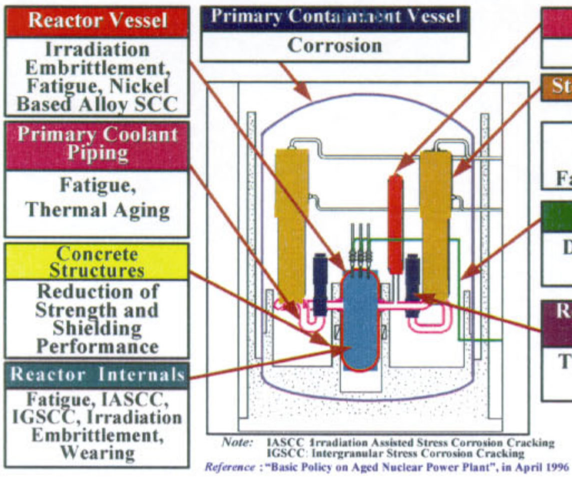


FIG. 1-1. Example of Aging Phenomena and the Major Components and Structures (PWR).

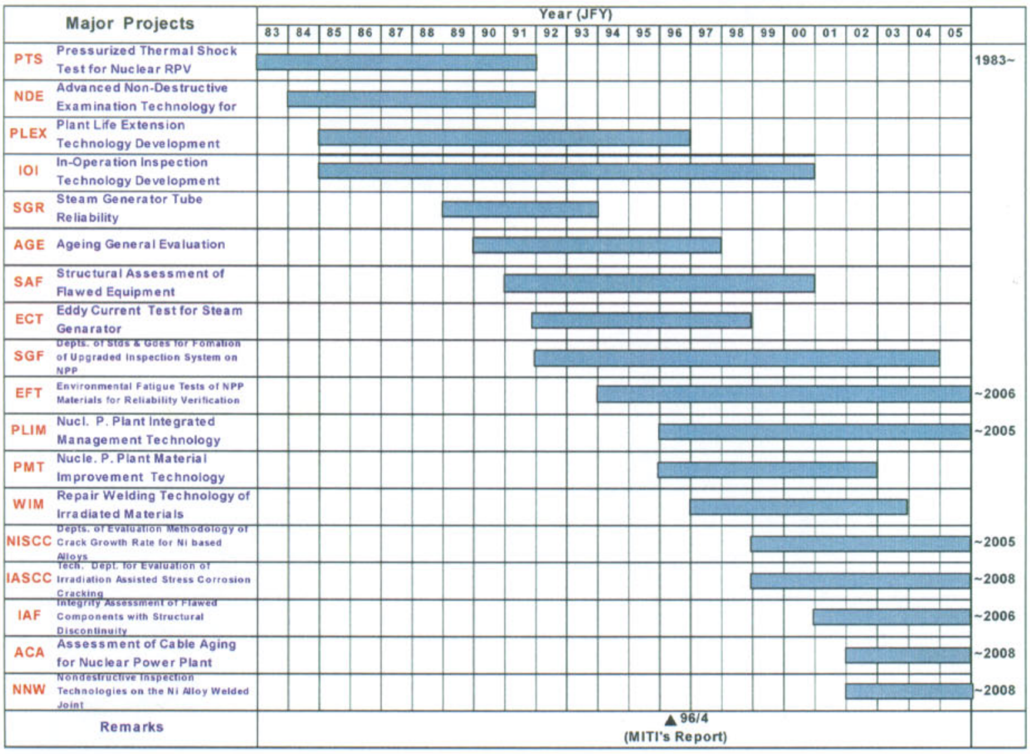


FIG. 2.1-1. Major Research Projects related to Aging.

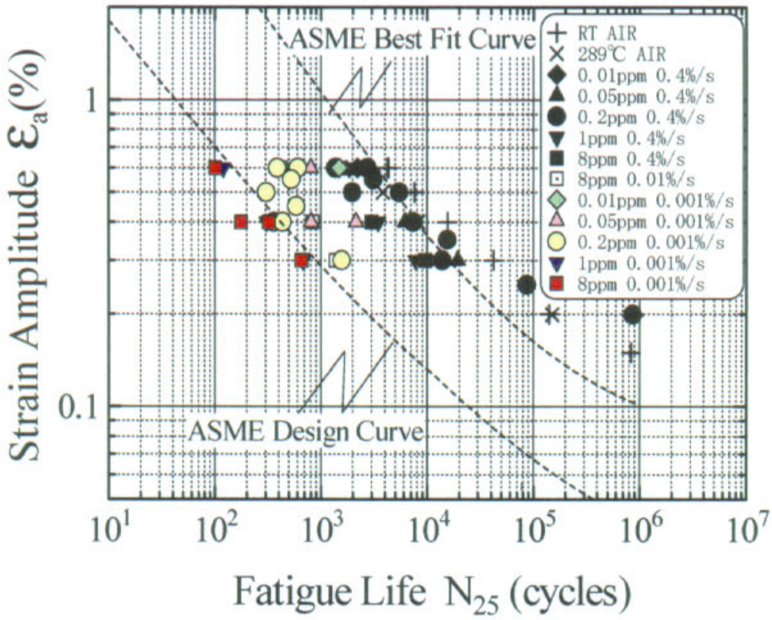


FIG. 2.2-1. Example of the results of Fatigue Test for Carbon Steels (BWR) Some data are below the ASME Design Curve.

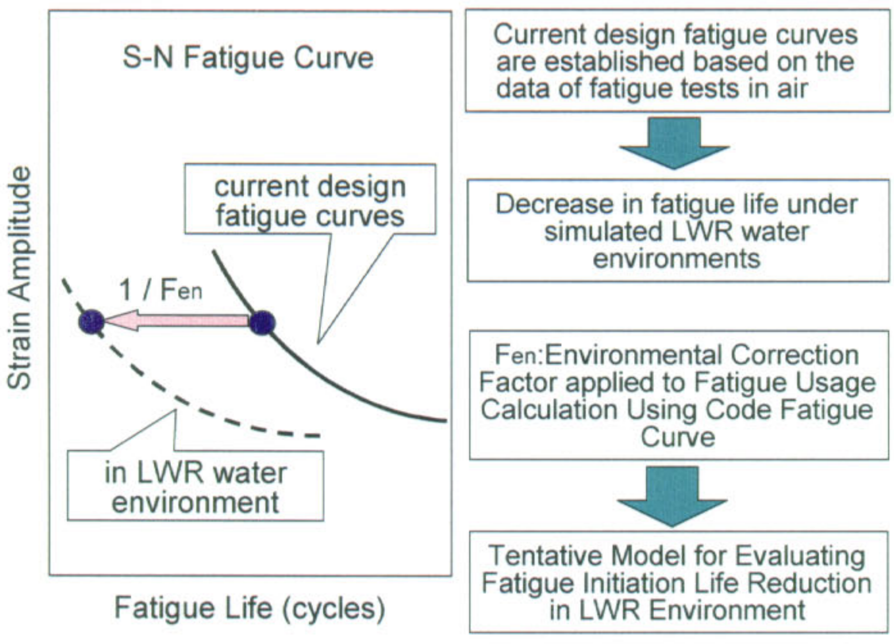


FIG. 2.2-2. Schematic Description of Fen.

$N_W = N_A$ / Fen

ln (Fen)=-(0.199 x T* x O*+ 0.112) x S* x $\dot{\varepsilon}$*

where,

$\dot{\varepsilon}^*$ = 0.0	($\dot{\varepsilon}$ > 1.0% / sec)
$\dot{\varepsilon}^*$ = ln ($\dot{\varepsilon}$)	(1.0% / sec ≦ $\dot{\varepsilon}$ ≦ 0.0004% / sec)
$\dot{\varepsilon}^*$ = ln (0.0004)	($\dot{\varepsilon}$ < 0.0004% / sec
T* = 0.00531 x T – 0.7396	(T ≧ 180 ℃)
T* = 0.216	(T < 180 ℃)
O* = ln (DO/ 0.03)	(0.03 ≦ DO ≦ 0.5ppm)
O* = 0.0	(DO < 0.03ppm)
O* = ln (0.5 / 0.03)	(DO > 0.5ppm)
S* = 17.23 x S + 0.777	

Fen : Environmental fatigue life reduction factor

N_A : Fatigue life in air at (cycles)

N_W : Fatiguelife in water (cycles)

$\dot{\varepsilon}$: Strain rate on the strain increasing leg of cycle (%/s),

$\dot{\varepsilon}^*$: parameter depending on strain rate,

DO : dissolved oxygen content (ppm),

O* : parameter depending on dissolved oxygen content

S : Sulfur content of steels (weight%),

S* : parameter depending on Sulfur content of steels

T : Temperature (℃),

T* : parameter depending on Temperature

FIG. 2.2-3. Fen: Evaluation Formula for Carbon Steels and Low Alloy Steels.

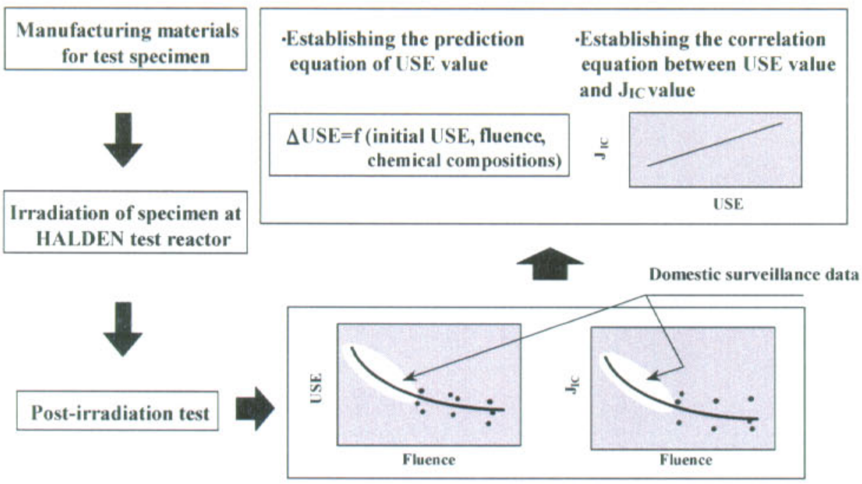


FIG. 2.3-1. Evaluation Method of Neutron Irradiation Embrittlement (predict USE reduction and to establish correlation between USE and fracture.

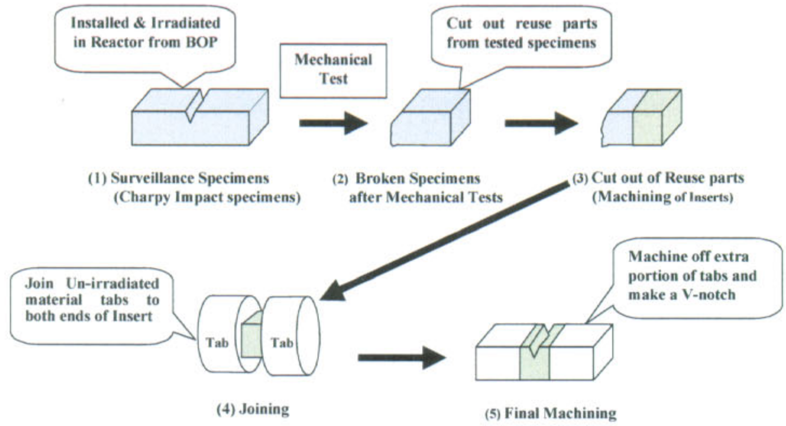


FIG. 2.3-2. Monitoring Technology for Neutron Irradiation Embrittlement (Re-constitution of surveillance test specimens of reactor pressure vessel).

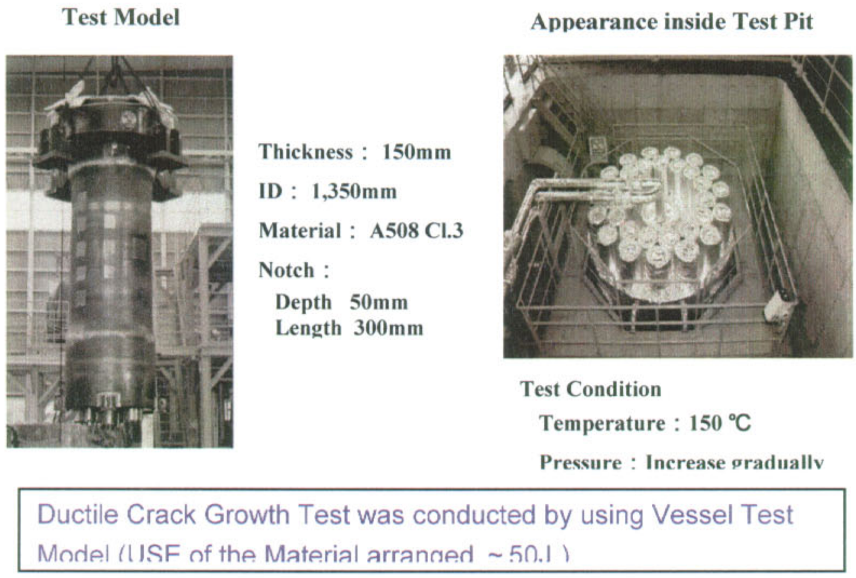


FIG. 2.3-3. Ductile Crack Growth Test in Vessel.

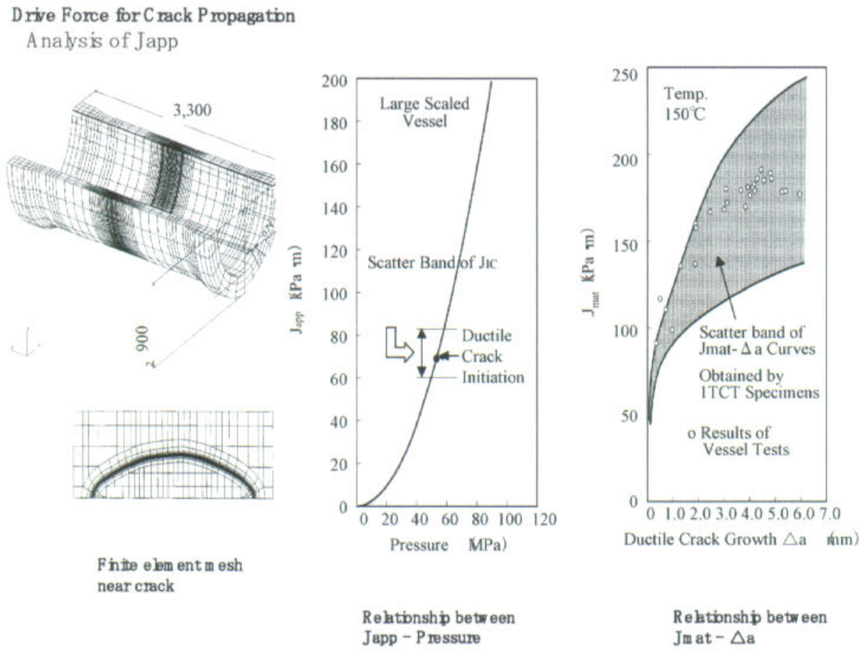


FIG. 2.3-4. Results of Ductile Crack Growth Test in Vessel.

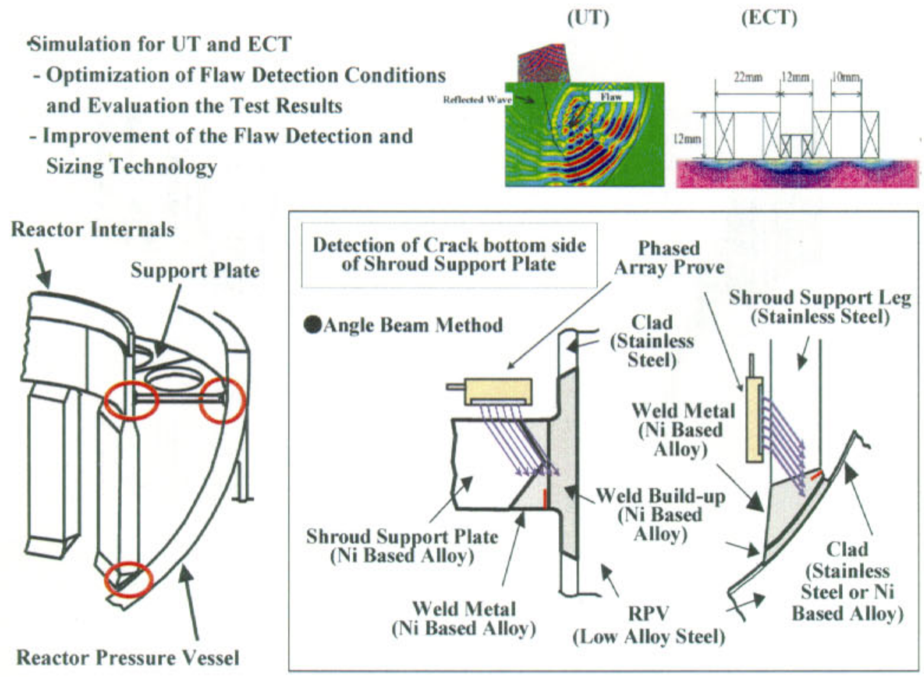
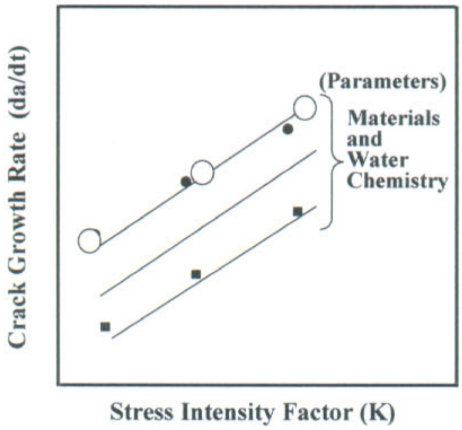


FIG. 2.4-1. Nondestructive Inspection Technologies on the Ni Based Alloy Welded Joint.



Schematic Figure of Crack Growth Rate (K – da/dt)

FIG. 2.4-2. Evaluation Technology for Stress Corrosion Crack Growth of Nickel Based Alloy.

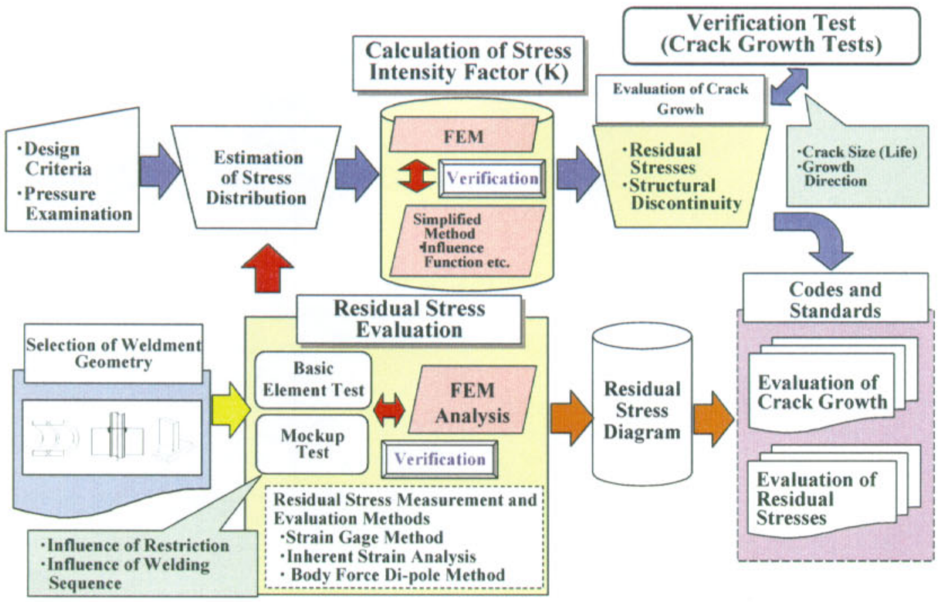


FIG. 2.4-3. Evaluation Technology for Stress Corrosion Crack Growth of Nickel Based Alloy (Evaluation of Residual Stresses, Stress Intensity Factors and Crack Growth Tests).

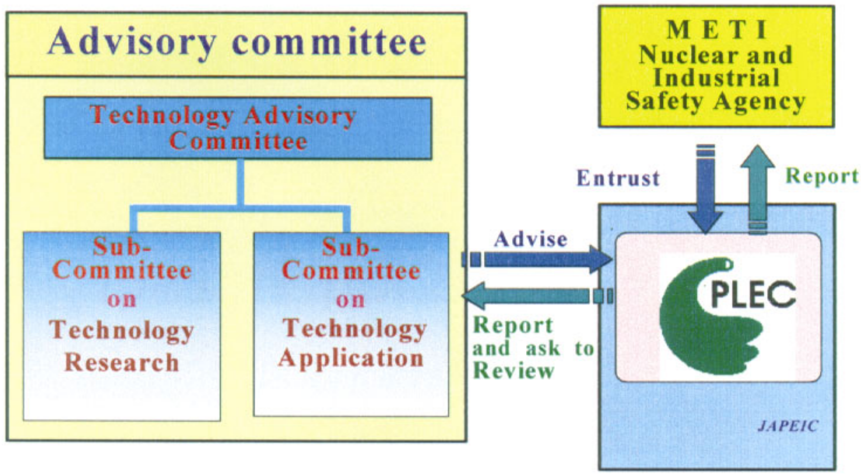


FIG. 3.2-1. Organization for PLEC Activities.

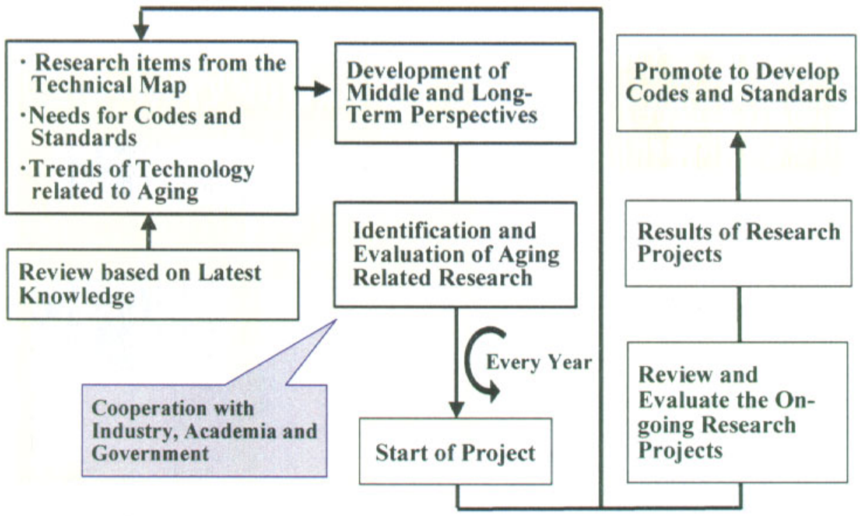


FIG. 3.3-1. Planning of Research on Aging.

Aging Phenomena			Irradiation Assisted Stress Corrosion Cracking							
Components	Parts	Environment (Irradiation)	Structural Materials	Current Maintenance Program and Countermeasures	Inspection and Monitoring	Evaluation Methods			Preventive Maintenance and Refurbishment	Remarks
						Crack Initiation Evaluation	Crack Growth Evaluation	Integrity Evaluation		
Reactor Internals										

FIG. 3.3.1-1. Format Example of the Technical Map for Aging Research.

LEAK/BURST TEST ON SCC DEGRADED STEAM GENERATOR TUBINGS

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Abstract

Forced outage due to a steam generator tube leak in Korean nuclear power plants has been reported [1]. Primary water stress corrosion cracking of the steam generator tubings occurred on many tubes in a plant, and they were repaired using sleeves or plugs. In order to develop proper repair criteria, it is necessary to know the leak behavior of the tubes, which have stress corrosion cracks. Crack development tests were carried out on the tubes at room temperature, and leak rate and burst pressure were measured on the degraded tubes at room temperature and high temperature. The tubes having 100 % through wall crack did not show a leakage at 10.8 MPa, which is an operating pressure difference of pressurized water reactors(PWRs). In some tests, leak rates of the tubes increased with time at a constant internal water pressure. A test tube showed a very small leakage at 18.6 MPa in high temperature pressure test at 282 ^{o}C, but it disappeared after the pressure increased a little. Even though cracks are 100 % through wall, they need to be open in order to reach a certain amount of leak rate at the operating pressure difference.

1. INTRODUCTION

For many years, steam generators of PWR have suffered from many types of corrosion, such as pitting, wastage and stress corrosion cracking (SCC) in primary and secondary side. In order to prevent primary coolant from leaking to the secondary side, the tubes are repaired by sleeving or plugging. It is important to establish repair criteria to maintain the plugging ratio within the limit to operate it well.

In the international steam generator tube integrity program (ISG TIP) supported by US NRC(Nuclear Regulatory Commission), works such as in-service inspection technology development, study on steam generator tube degradation mode have been undertaken. Some Korean archive tubes were tested to understand the leak behaviors under operating and accident condition of PWRs.

This article aims to investigate the leak behavior of the archive tubes at normal plant operating pressure, accident pressure and burst pressure.

2. EXPERIMENTAL

2.1. Laboratory stress corrosion cracking of SG tubes

Laboratory induced stress corrosion cracks were introduced into steam generator tubes using the ANL (Argonne National Laboratory) SCC production facility and techniques described in reference [2]. Two kinds of Korean archive alloy 600 tubes were prepared for the work. The alloy 600 tubings, which have 19.05 mm of out diameter and 1.07 mm wall thickness, are high temperature mill annealed. All the tubes of 356 mm long were sensitized at 600^{o}C for 48 hours in a vacuum tube furnace. In order to avoid the tubes from oxidizing on the surface, the furnace was vacuumed and filled with helium and hydrogen mixture gas three times. The surface of the tubes was abraded by emery paper # 600 followed by acetone cleaning before being subjected to the SCC development.

The tube specimens were exposed to 1 M sodium tetrathionate solution at room temperature and nitrogen gas pressurized inside the tubes. All tests stopped when a leakage was found on each tube by the indication of pressure drop. The test time varied from 69 hours to 607 hours depending on the tube. Crack development test conditions and results on each tube are shown in Table 1. The length and depth of the defects of tubes were checked roughly by the eddy current method, and the tubes were transferred to the leak and pressure test step.

2.2. Leak rate measurement

Leak rates at a certain internal pressure in the degraded tubes were measured by using the ANL facility at room temperature and high temperature.

Table 1. Laboratory Stress Corrosion Cracking of the Korean Archive Tubes

Tube ID	Heat	Crack type	Solution /Temp	SCC Time	Leak proved?	Cursory ECT
SGH001	UC4 (600 HTMA Sensitized for 48 hrs @ 600C	OD axial	1 M Na2S4O6 @room temp.	68.8 hrs	20.7 MPa bubble: Yes!	OD, 95%/5.08mm, 90%/11.43mm
SGH002	"	"	"	67.8 hrs.	"	Axial indication
SGH005	YG5,6 (600 HTMA Sensitized for 48 hrs @ 600C)	"	"	68.8 hrs	"	90%/15.2mm
SGH006	"	"	"	68.8 hrs.	"	Axial indication(1)
SGH009	"	"	"	404 hrs.	"	OD, 80%/10.2mm
SGH010	"	"	"	607 hrs.	"	OD, 85%.5.08mm
SGH012	"	ID axial	"	264 hrs.	"	ID, 100%. 24mm

The room temperature test facility is equipped with a water pressurizing pump, and test specimen section and control unit as shown in the report NUREG/CR-6511 [3]. The first leak from the tube was detected through the transparent plastic window by eye, and the leak rate at a certain pressure was measured by weighing water flown out from the crack. The pressure was held at 10.8, 13.8, 17.2, 20.7, 27.6 MPa for 5 to 10 minutes to measure a time dependent variation of the leak rate. On some tubes, higher pressures were applied to see a burst pressure.

During the first stage of the room temperature pressure test, it was attempted to obtain first leak pressure, leak rate at certain pressures and burst pressures. Two tubes of SGH002, SGH006 were prepared for this work. In the second stage of the test, 0.7 MPa pressure increment and holding for 4 to 10 minutes were adapted to see first leak pressure and leak rate changes with time. The pressurization stopped at a certain value at which a measurable leak rate could be obtained, and this process let the tubes avoid an unstable burst, which makes the metallography harder. Tubes SGH001, SGH005, SGH009, SGH010 were used for this analysis.

A high temperature pressure test was undertaken to investigate the leak behavior at the PWR operating temperature of 282 ^{o}C, the facility consists of a high temperature water reservoir and test section where the temperature of the specimen and pressurizing water was controlled. The maximum water pressure of this test is around 19.3 MPa. Detail of the facility is described in the report NUREG CR-6511 also. For the high temperature pressure test, tube SGH005 and SGH012 were used. The pressure was raised to 8.3 MPa and held for 45 minutes and then increased to 19.0 MPa slowly while checking if a leak symptom was detected at the muffler from which steam came when it leaked.

2.3. Leak rate prediction method

Leak rate based on the prediction model developed by ANL was adapted in order to analyze the leak behavior of Korean archive tubes (Ulchin 4:UC4 and Younggwang 5,6:YG5,6) [4]. The equation used in the analysis is as below.

$$Q = 50.9 \times 10^6 \mathbf{A} (\Delta P/\rho)^{0.5} \text{ [l/min]} \quad (1)$$

Where A is crack opening area in m^2, ΔP is the pressure difference across the tube wall in MPa, ρ is the density of water in kg/m^3(1000 kg/m^3 at room temperature, 735 kg/m^3 at 282^{o}C). In order to calculate the crack opening area, variables such as yield stress, mean radius, thickness and young's modulus were used, and details are described in the report NUREG/CR-6664. In the case of the Korean tubes heat-treated for producing SCC, yield stress was considered to be the same as the received materials. Yield stress for high temperature calculation was estimated as low as 10% from the room temperature value [4]. Yield stress of 265.5 MPa and 248.2 MPa for UC4 and YG5,6 respectively was used for room temperature evaluation. In case of high temperature calculation for SGH005, 221.3 MPa was used as the reduced yield stress in the leak rate calculation at 282^{o}C.

3. RESULTS AND DISCUSSIONS

3.1. Room temperature pressure test

The first leak was detected at 17.2 and 24.3 MPa on the tube SGH002, SGH006 respectively, which was similar to the crack developing pressure, 20.7 MPa at which the tube was confirmed as 100 % through wall by nitrogen gas. A droplet on the tube SGH002 was formed at 17.2MPa, but it did not grow any more for 5 minutes. At 22.0 MPa, one drop every 3 seconds was formed from the crack on the tube SGH002. It showed a leak rate of 0.24 liter/min at 29.6 MPa and 4.28 liter/min at 34.5 MPa, and ruptured at 35.9 MPa. Fig. 1 shows the crack of SGH002 after the pressure and leak test done at room temperature. The crack was torn like a fish mouth, which was formed during the rupture at 35.9 MPa, and the length is about 11 mm.

Tube SGH006 revealed a water spray at 23.4 MPa, it turned into a droplet, which was made for every 4 seconds. It showed a leak rate of 0.27 liter/min at 27.6 MPa and 3.87 liter/min at 34.5 MPa and ruptured at 39.3 MPa. This tube did not show a leakage at 10.8 MPa and 17.2 MPa.

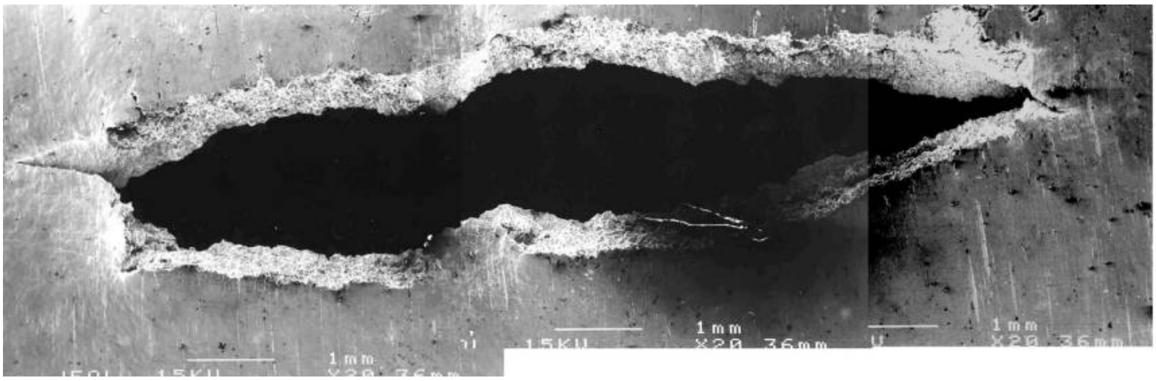


FIG. 1. Crack morphology of SGH002 taken by SEM after the pressure/leak test at room temperature.

History of the pressure test results is shown in Table 2. Though both tubes were 100 % through wall penetration, they did not leak at the operating pressure of 10.8 MPa. So, it is hard to say that the criterion of the detectable leakage is the fact that a crack is wall through. On the contrary, the crack opening or tightness is more important to show a certain amount of leak rate.

In the second stage of test, the three tubes SGH001, SGH009, and SGH010 showed a similar first leak pressure except SGH005, which was deformed during the high temperature pressure test. They did not leak at 10.8 MPa for 5 minutes, which is considered as a normal operating pressure difference in PWRs. First leak pressures of each tube SGH001, SGH009, SGH010 were 25.0, 20.7, 22.8 MPa respectively, even though these tubes showed nitrogen gas leak at 20.7 MPa during crack development. The first leaks were in the form of a single water drop forming in every 4 to 8 seconds. The droplet changed into a water jet after the pressure increased to 2.4 MPa higher after the first leak. Leak rates were measurable above the jet pressure, leak rates measured first on each tube were 0.19 l/min at 31.7 MPa for the tube SGH001, 0.019 l/min at 24.1 MPa for the tube SGH009, 0.13 l/min at 29.6 MPa for the tube SGH010. The leak rate changed with time, which came from a crack opening at high pressure, was observed for all three tubes, and then they showed a constant leak rate after about 30 minutes.

Fig. 2 shows flaws of the tubes SGH001 after the pressure test at room temperature. Two or more cracks are linked with each other in an axial direction, and they showed a different opening.

Table 2. History of the pressure test on Korean archive tubes(1000psi=6.9MPa).

Tube ID	Heat	Type, SCC time	Leak proved?	Cursory ECT(Effecive crack length, mm)	Leak test temp.	Leak @1560 psi(4)	Leak @2500 psi(5)	First leak, psi	Spray starts psi	Leak rate 1, l/min	Leak rate 2, l/min	Burst pressure or final psi
SGH001	UC4 (600 HTMA Sensitized for 48 hrs @ 600C	OD axial, 68.8 hrs	3000 psi gas bubble: Yes!	OD, 95%/5.08mm, 90%/11.43mm	Room temp.	No leak	No leak for 5 min.	one drop/6sec @3620	3970	0.19 @4600	0.25 @4600	-
SGH002	"	67.8 hrs.	"	Axial indication(3)(10.5mm)	Room temp.	No leak	No leak for 5 min.	one drop @2500	4110	0.24 at 4300 psi	4.28 at 5000 psi	5200
SGH005	YG5,6 (600 HTMA Sensitized for 48 hrs @ 600C)	68.8 hrs	"	Before pressure: 90.63 volts/38°, 100%/12.7mm After pressure: 30.04 volts/54°, 90%/15.24mm	282C	No leak	No leak	2700	Leak @2700, but disappeared after 15 min, No more leak @ 2760			
SGH005 (1)	"	"	"	After Heat tinting(Before RT pressure test):	Room temp.	Spray @ 60 psi	-	-	0.23 l/min @1560 psi	0.32 @2500	0.34 @2750	0.43 @3000
SGH005 (2)	"	"	"	"	Room temp.	Pressurization at 1000 psi/sec, Leak rate of 44.28 l/min at peak pressure of 5000 psi.						
SGH006	"	68.8 hrs.	"	Axial indication(3)(12.3mm)	Room temp.	No leak	No leak for 5 min.	one drop @3400	3400	0.27 at 4000 psi	3.87 at 5000 psi	5700
SGH009	"	404 hrs.	"	Axial OD, 80%TW/10.16 mm	Room temp.	No leak	No leak for 5 min.	one drop @3000	3160	0.019 @3500	0.023 @3500	0.023 @3500(6)
SGH010	"	607 hrs.	"	Axial OD, 85%TW/5.08mm	Room temp.	No leak	No leak for 5 min	one drop @3300	3600(7)	0.13 @4300	0.16 @4300	0.16 @4300(6)
SGH012	"	264 hrs.	"	Axial ID, 100%TW/24mm	282 C	No leak	No leak for 135 min					
SGH012 (2)	"	"	"	"	Room Temp.				atomized spray@1200 psi	0.020 @1200	6.99 @4100	

Crack length of inside tube is shorter than that of the outside surface on each tube. It is considered to come from two things; one of them is the inside pressurization procedure and the other is outside diameter crack development procedure. An effective crack length, which is related with leak rate, is considered as inside length of the crack. The relationship between leak rate and crack length is shown in Fig. 3.

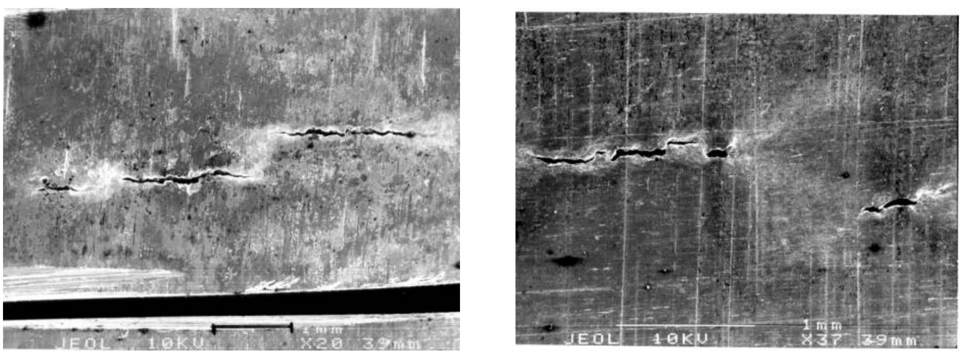


FIG. 2. Flaws on the SGH001 after the pressure/leak test at room temperature

Room temperature pressure test on the tube SGH005 was carried out after the high temperature pressure test, which is described later in this section. The tube demonstrating a non-measurable leak at the previous test, showed a jet spray at 0.4 MPa, leak rate of 0.23 l/min at 10.8 MPa and 0.43 l/min at 20.7 MPa. After the high temperature pressure test, where

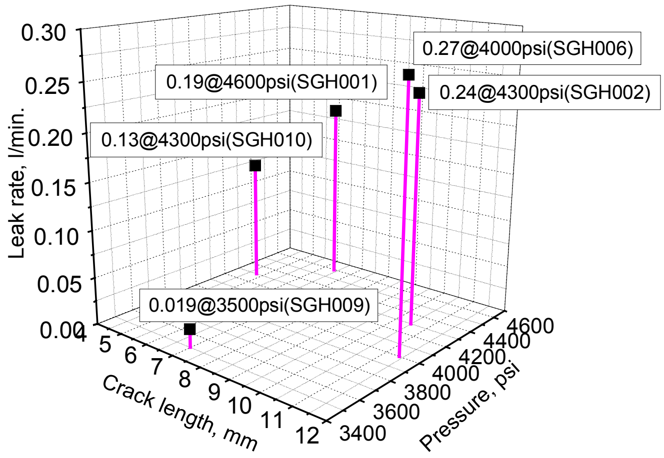
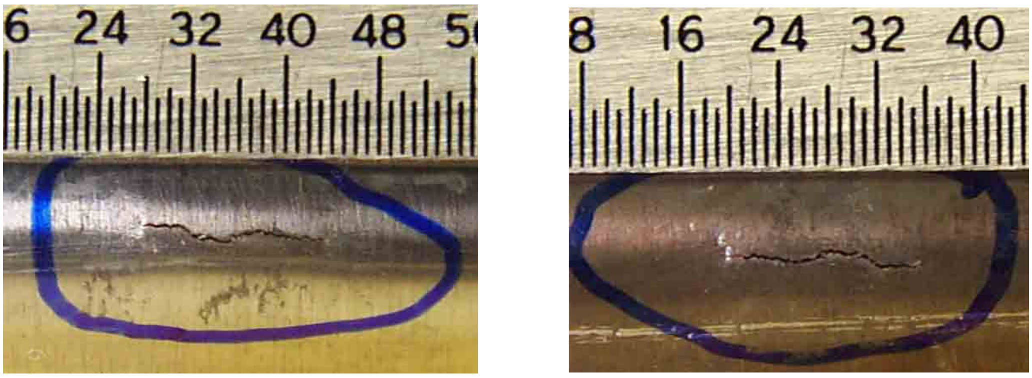


FIG. 3. Relationship among crack length, pressure and leak rate (1000 psi =6.9 Mpa).

a non-measurable leak was detected, the specimen was heat tinted to perform the metallography easily. The crack seems to open during the heat tinting, to show the jet spray at the low pressure and to show a large leak rate at 10.8 MPa. Fig. 4-(b) shows a feature after the room temperature pressure test, the length of the flaw increased a little, but an apparent feature such as crack opening was not different from that in the Fig. 4-(a), which was obtained after the high temperature pressure test. In order to know the unstable burst pressure of the flaw, SGH005 underwent the test with the pressurization rate of 6.9 MPa/second. Leak rate was 44.28 l/min at the peak pressure of 34.5 MPa. There was no big change in the flaw length even after the unstable burst, whereas the crack opening increased.

3.2. High temperature measurement

The tube SGH005, which had been confirmed as 100 % through wall during crack development, was subjected to the high temperature pressure test first. The tube did not show a leak until the pressure was 18.6 MPa. At the internal pressure of 18.6 MPa, a little steam coming out from the inside of the tube was detected at the end of muffler of the test facility. The leak rate was considered as low as 0.1 liter/min. The steam, however, disappeared after 55 minutes; no further steam flew out even after the pressure increased to the maximum value of 19.0 MPa. After the maximum pressure was applied for about 10 minutes, the test stopped and the specimen was cooled down for further testing by using the room temperature test facility.



(a) (b)

FIG. 4. Flaws on the SGH005 after (a) High temperature pressure test, (b) Room temperature pressure test.

This behavior may be interpreted as a crack closure inside the tube while opening up the outer crack by the internal pressure. ECT (Eddy current test) indicated that the effective length of the crack after pressure test decreased to 12.7 mm from 15.2 mm, whereas, the EC voltage increased to 99 volts after the pressure test from 30 volts before the pressure test as shown in Fig. 5. These ECT results allow us an assumption of crack closure during the pressurizing. However, the ECT results and the assumption could be fortuitous.

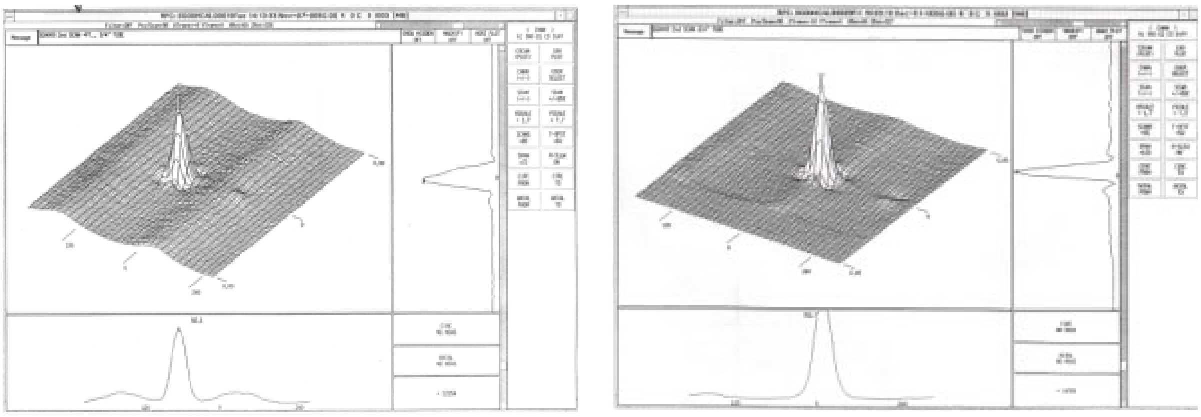


FIG. 5. Eddy current examination on the SGH005.

(a) before HT pressure test *(b) after HT pressure test*

SGH 012, which had inside cracks of 23 mm long, showed similar behavior at 282 ^{o}C. It did not show a leakage at 17.2 MPa for 135 minutes. But it showed a water spray at 8.3 MPa at room temperature leak test.

3.3. Leak rate prediction

This work analyzed three tubes, SGH001, SGH009, SGH010, on which pressurization was applied until a stable leak rate was obtained before the unstable burst.

Four different pressures were applied to obtain leak rate on the tube SGH001. The first leak from the tube was recorded at 25.0 MPa in the form of a single water droplet. Leak rate of that kind of drop is considered as less than 0.01 l/min, calculated crack length based on the leak rate model described in the section 2.3 is about 0.6 mm. A measurable leak rate was obtained at 31.7 MPa, it changed a little with time at the same pressure. According to the leak rate model, the calculated crack length is 2.10 mm for the final leak rate of 0.25 l/min as shown in Fig. 6. Crack lengths of the main crack of this tube were 3.85 mm and 1.55 mm outside and inside respectively. The calculated crack length of 2.1 mm is between the measured inside diameter crack length and outside diameter. From the difference of crack lengths between measured and calculated one, it is considered that some of the final leak rate came from the minor cracks. When the pressure reached at 31.7 MPa, the calculated leak rate for 1.55 mm crack is 0.12 l/min. The measured value of 0.25 l/min is close to the calculated value for 2.10 mm crack.

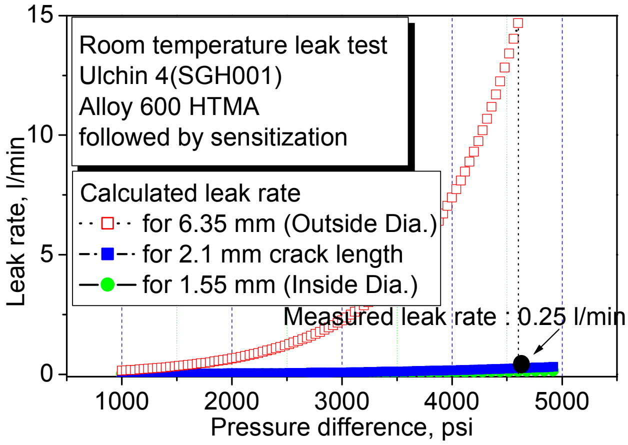


FIG. 6. Comparison of the leak rate calculated and measured on the tube SGH001 (UC4)(1000 psi=6.9 MPa).

In the case of SGH009, the final leak rate of 0.023 l/min was obtained at 24.1 MPa. A calculated crack length is 1.0 mm, which is longer than inside crack length and much shorter than the OD crack length. The measured leak rate of 0.023 l/min is on the leak rate line for 1.0 mm crack length as shown in Fig. 7. This means that the crack length related leak of SGH009 is in between the measured inside crack length and outside the diameter crack length also.

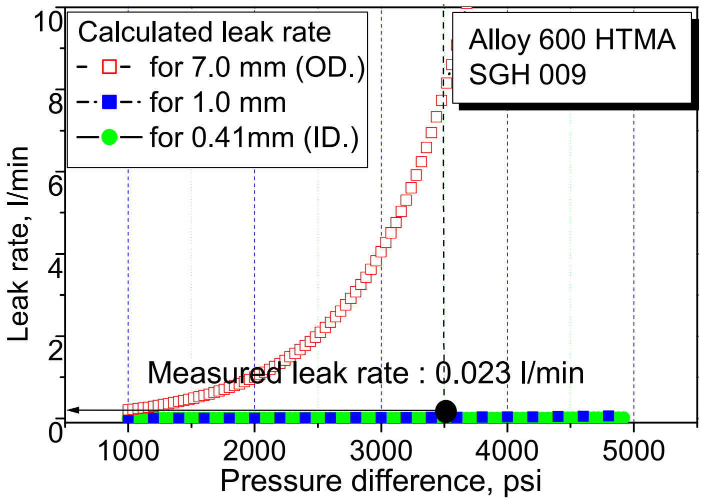


FIG. 7. Comparison of the leak rate calculated and measured on the tube SGH009 (YG5,6) (1000 psi=6.9 MPa).

Fig. 8 shows the relationship between leak rate and effective crack length of the tube SGH010. Like the tube SGH001, this tube showed a time dependent increase of leak rate. Final leak rate was 0.16 l/min at 29.6 MPa, and this is correspondent to the calculated crack length of 1.9 mm. The calculated crack length is longer than the measured inside diameter crack length of 1.07 mm.

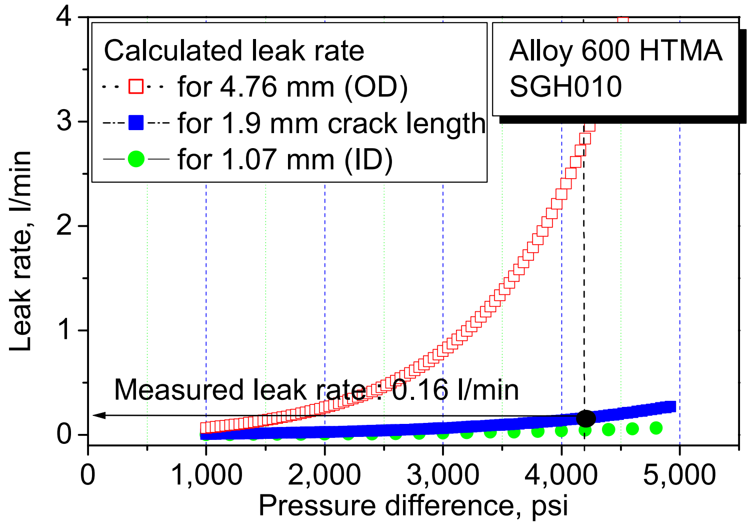


FIG. 8 Comparison of the leak rate calculated and measured on the tube SGH010 (YG5,6)

The calculated leak rate for the crack length of 1.9 mm fits the measured leak rate well. The twice or four times larger leak rates than calculated in the tubes SGH009, SGH010, seem to be related to a different crack opening of the two tubes. Unlike tube SGH001, the two tubes had crack opening areas two times larger than calculated as indicated. From this fact, we know that heat YG5,6 opened easily rather than increased in crack length, thus consequently showed a large leak than expected from the leak rate model.

4. CONCLUSIONS

Pressure and leak tests were conducted at room temperature and 282°C using archive steam generator tubes, which have outside diameter stress corrosion flaws grown at the laboratory.

- All the tubes on which cracks were developed at 20.7 MPa and having 100 % through wall penetration didn't leak at the plant operating pressure.
- In this test, outside crack length was at maximum 17 times longer than inside crack in a tube.
- The first leak in the form of water droplet was detected between 17.2 MPa and 24.8 MPa depending flaws.
- Burst pressure of through wall crack of 10 mm long was 35.9 to 39.3 MPa.
- Calculated leak rate for a crack length relatively close to inside crack length fits the measured leak rate well.

ACKNOWLEDGEMENT

This work was carried out as a part of the assessment of corrosion characteristics and development of remedial technologies in nuclear materials project under the nuclear R&D program sponsored by MOST in Korea.

REFERENCES

[1] J.S.Kim, S.S.Hwang, et al., 'Destructive analysis on pulled tubes from Ulchin unit 1' Korea Atomic Energy Research Institute, 1999.
[2] D.R.Diercks, S. Bakhtiari, et al., 'Steam generator tube integrity program-annual report', NUREG/CR-6511, AN Vol.7, ANL-0014, Argonne National Laboratory, 2000.
[3] D.R.Diercks, S. Bakhtiari, et al., 'Steam generator tube integrity program-annual report', NUREG/CR-6511, Vol.2, Argonne National Laboratory, 1998.
[4] S. Majumdar, Ken Kasza, et al., 'Pressure and leak rate tests and Models for failure of flawed steam generator tubes', NUREG/CR-6664, Argonne National Laboratory, 2000.

Session 4
TECHNOLOGICAL AND OPERATIONAL ASPECTS OF PLIM

Sub-Session 4.4

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NUCLEAR ENERGY PLANT OPTIMIZATION (NEPO) PROGRAM CABLE SYSTEM AGEING MANAGEMENT PROJECTS

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Abstract

Cable polymer aging and condition monitoring is being studied in detail under the NEPO Program, which is co-sponsored by the U.S. Department of Energy and EPRI. Significant advances in modeling of polymer aging and condition monitoring have occurred and continue to be developed. The activities include:

– Analysis of the linearity of the Arrhenius model to room temperature
– Determination of the aging fragility point for composite EPR/CSPE insulation with respect to LOCA function
– Development of visual/tactile training aids for cable assessment
– Development of a totally new nuclear magnetic resonance condition monitoring technique
– Assessment of existing techniques with regard to repeatability, accuracy and ease of use.

1. INTRODUCTION

Under the NEPO Program, a number of cable polymer aging assessment projects were begun in 2000. This paper provides an overview of a few of these projects.

The NEPO Programis for existing U.S. nuclear power plants and has the goals of performing research that leads to increased plant output, improved overall fleet capacity factor, and removal of technical barriers to successful operation through the end of the license renewal periods. Aging of cable in nuclear power plants can affect safety and operations if not managed. Accordingly, the NEPO Program includes a significant amount of research to assure that aging of existing cable insulations and jackets is well understood, that useful techniques for assessing the condition of cables exist, and that data for accurate projection of remaining life are available. The work is being performed both at Sandia National Laboratories and EPRI.

Sandia is improving cable polymer aging models through use of very accurate oxygen uptake experiments, developing a "wear-out" assessment technique that allows estimation of remaining life for insulations, and developing and evaluating a number of polymer condition monitoring techniques. In the process of these efforts, Sandia is compiling condition-monitoring data for a number of different methods performed on the specimens subjected to numerous different accelerated aging regimens. The data are providing correlations among many different condition-monitoring techniques for commonly used cable insulations and jackets.

EPRI has performed fragility assessment of ethylene propylene rubber (EPR)/chlorosulfonated polyethylene (CSPE) composite insulation, developed visual/tactile cable aging assessment training aids, and compiled a polymer aging database for cables from U.S. and international sources.

2. AGING MODEL IMPROVEMENTS

The Arrhenius thermal aging model is commonly used to allow long real-time cable lives to be simulated with relatively short, high-temperature exposures. The model extrapolates high temperature conditions (120 to 160°C) to lower operating temperatures (40 to 70°C). Potential problems exist with the accuracy of the extrapolation. The activation energy used in the calculation is determined using data from high temperatures. Some materials have different chemical degradation reactions at higher temperatures than they do at lower temperatures. For crystalline polymers, the melting point may be between the lower operating temperature and the accelerated aging temperature. Therefore, data from above the melting point of the crystals may be not at all related to aging at operating temperatures. Experiments to obtain low temperature data to confirm the accuracy of the extrapolations require experiments to be performed in real time and take too long to complete. Figure 1 provides a well-known example of an aging curve where low temperature behavior was totally different than the extrapolation from high temperature.

Through use of highly precise oxygen consumption experiments, Sandia National Laboratories is verifying the linearity of the Arrhenius model to room temperature. In these experiments, small quantities of cable polymer are placed in vials with known amounts of oxygen and aged at much lower temperatures than is possible with standard accelerated aging techniques. Most of the aging reactions for cable polymers are oxygen based. Therefore, measuring oxygen consumption from thermal aging will provide a correlation with the degree of aging of the polymer. Oxygen consumption can be measured to room temperature. The technique is being applied to commonly used insulation and jacket polymers.

The first effort is to perform high temperature experiments to confirm that the slope of the oxygen consumption line is parallel to the line for a standard monitoring technique that tracks degradation. The oxygen consumption experiments are then continued at lower and lower temperatures to determine actual aging rates at low temperature. Some results for a neoprene cable jacket are plotted on an Arrhenius plot in Figure 2. Conventional elongation results and oxygen consumption results at higher temperatures indicate an Arrhenius activation energy of ~23 kcal/mol. Lower temperature oxygen consumption results (down to 23°C!) indicate a small drop in activation energy at temperatures in the traditional extrapolation region.

The oxygen consumption experiments are conducted by sealing a small section of polymer in a sample holder with a known quantity of oxygen. The oxygen pressure is adjusted such that standard pressure occurs at the aging temperature. Subsequent to aging, the oxygen that has been consumed is determined with a gas chromatograph. Careful experimentation allows the technique to measure consumption at temperatures corresponding to polymer lifetimes greater than 100 years. Figure 3 shows typical sample holders used for the tests and Figure 4 shows the gas handling station. Common cable materials (neoprene, Hypalon, crosslinked polyethylene, and ethylene propylene rubber) from a number of different cable manufacturers are being evaluated for oxygen consumption.

3. AGING FRAGILITY OF EPR/CSPE COMPOSITE INSULATIONS

Some cable insulation systems have EPR insulation covered by a CSPE (Hypalon) layer for fire retardance. The two layers of this insulation system are bonded together. For common wire sizes used in instrument and control (12 and 14 AWG (2.05 mm and 1.63 mm), 30 mils (0.76 mm) of EPR is covered by 15 mils (0.38 mm) of CSPE for a total insulation thickness of 45 mils (1.14 mm). The CSPE layer that is bonded to the EPR must not be confused with the overall jacket that covers multiple cabled conductors. The overall jacket is not bonded to the individual conductors and is not of concern in this discussion.

EPR ages more slowly than CSPE and was chosen as the conductor insulation to be capable of withstanding ohmic heating and temperature rise from short circuit currents. In some nuclear power plant uses, the ambient temperature is high during normal operation. In such cases, the CSPE layer may age relatively rapidly and eventually lose its properties as a rubber and become hardened. When this occurs, the CSPE layer can control failure under manipulation or if a pressurized steam condition, such as a loss-of-coolant accident, occurs. If the CSPE hardens to the point where only a few percent elongations at break remains, bending the insulated conductor slightly will cause the CSPE layer to crack. Continued bending can cause the crack to propagate through the EPR. Even if the insulation system has not been cracked from manipulation, exposure to pressurized steam can cause the insulation to crack along its length or circumferentially due to swelling of the materials. The cracks propagate to the conductor if the CSPE has been severely aged [1].

To determine the aging fragility point after which failure in a LOCA exposure is likely, EPRI performed a research program that subjected two different manufacturer's ERP/CSPE insulation systems to 36 different degrees of aging from new through five degrees of thermal aging and new through five degrees of radiation aging. The specimens were then subjected to 150 Mrad (1.5 MGy) of CO_{60} gamma irradiation and saturated steam testing. A second set of specimens was subjected to superheated conditions with the same thermal profile. Because the failure mechanism is mechanical in nature, short specimens (6-in (15.2 cm)) were used and the specimens were not energized. Figure 5 shows a few of the test specimens in the holding rack.

While approximate accelerated thermal aging times were determined through use of the Arrhenius model, the specimens were removed from the ovens when they reached a specific Indenter modulus (a form of hardness measurement). For each aging increment sufficient specimens were available to perform Indenter modulus and elongation-at-break test. These specimens were not subjected to accident irradiation. Figure 6 shows the elongation-at-break and Indenter modulus data for the Okonite specimens. Figure 7 shows lifetime equivalencies with assumed activation energy of 1.1 eV. A long life can be expected for the EPR/CSPE materials if moderate plant temperatures exist during normal operations. For plant areas with elevated temperatures, early replacement may be necessary and condition monitoring is recommended to allow replacement before over aging occurs. The complete program is described in Reference [2].

4. VISUAL/TACTILE ASSESSMENT

While sophisticated condition monitoring techniques can be applied to cable systems, visual/tactile assessment may be used to perform initial screening assessments. Most currently installed cable types have at least some aging attributes that can be assessed either visually or by manipulation. To support visual/tactile assessment, EPRI developed kits containing both single insulated conductors and multi-conductor cables in new and aged states. The specimens allow the unaged state of the insulation and jacketing systems to be understood with respect to configuration, coloring, hardness and flexibility. The specimens with four degrees of aging allow the trainee to identify the differences between the unaged and aged states of the materials. The specimens with the higher degrees of aging are much stiffer than those of the unaged state and some are aged sufficiently that they will crack when bent even slightly. Use of the specimens allows the trainees to identify both over-aged cables and cables that have begun to age significantly such that in-plant inspections in areas with high normal temperatures can be performed. If a large number of cables are found to be aged, more sophisticated condition monitoring techniques can be applied to more accurately determine the degree of aging to allow planned replacement programs to be implemented.

Figure 8 shows the single conductor specimens. Three different manufacturer's insulations are included. A common descriptive tag is provided for each type of cable insulation. Sub-tags describing the degree of aging are attached to the individual specimens.

A report [3] has been prepared that describes the aging regimens used to prepare the kits and what can be learned from each set of specimens.

5. CONDITION MONITORING TECHNIQUES

If large numbers of cables have aged perceptibly, then more sophisticated condition assessment techniques will be needed to discriminate between degrees of aging to allow planning and scheduling of replacements. The NEPO cable program is evaluating a number of existing techniques to determine their ease of use and accuracy, and is developing a totally new technique based on nuclear magnetic resonance (NMR) relaxation measurements. The NMR measurement is performed on a polymer sample that has been allowed to swell in a suitable solvent. Solvent swelling enhances the discrimination between unaged and aged polymers. The relaxation time of the hydrogen nuclei spins is closely related to the mobility of the polymer chain. As materials harden from cross-linking of long chain molecules, the relaxation time, T_2, of the nuclear spins decreases. Figure 9 shows the relaxation time as a function of aging time at various temperatures for a Hypalon material. Repeatable results have been achieved for specimens as small as 0.1 mg. Thus, the test can be performed without removing significant amounts of jacket or insulation material from a field cable.

6. CABLE POLYMER AGING DATABASE

One of the key components of the NEPO cable program is a consolidated database of research results from condition monitoring experiments for cable insulation and jacketing systems. Nine entities have provided data to date. These are:

- EPRI - University of Connecticut (Natural versus Artificial Aging Program)
- EPRI - Ogden (Indenter Data)
- International Atomic Energy Agency
- Nuclear Research Institute REZ
- Pacific Sierra Research Corporation
- Research Institute of Scientific Instruments
- SKi - Ingemanson Technologies
- US Department of Energy - Sandia National Laboratories
- US Department of Energy - Brookhaven National Laboratory

Each of these entities has provided data from accelerated aging tests that included assessment of cable insulation and jacket specimens through use of condition monitoring. The database is searchable by insulation/jacket manufacturer, insulation/jacket chemical name, condition monitoring test type, and supplier of the data. The data is linked to either summaries of the test program or the report for the program. The data is also linked to test regimen and test specimen descriptions. The data may be viewed and printed in tabular and graphical form. The database is expected to be available for use in early 2003.

ACKNOWLEDGMENT

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REFERENCES

[1] U.S. NUCLEAR REGULATORY COMMISSION, Assessment of Environmental Qualification Practices and Condition Monitoring Techniques for Low-Voltage Electric Cables, NUREG/CR-6704, Washington, DC

[2] TOMAN, G., L. DUNCAN, W. WEAVER, Investigation of Bonded Jacket Cable Insulation Failure Mechanisms, EPRI, Palo Alto, CA and U. S. Department of Energy, Washington, D.C.: 2002, 1001002

[3] TOMAN, G., L. DUNCAN, Training Aids for Visual/Tactile Inspection of Electrical Cables for Detection of Aging, EPRI, Palo Alto, CA and U. S. Department of Energy, Washington, D.C.: 2002, 1001391

[4] HARRIS, D.J., M. CELINA, R. A. ASSINK, Oxidative Degradation Analysis of HTPB/IPDI Polyurethane Using ^{17}O and ^{13}C NMR, Sandia National Laboratories, Albuquerque, NM. Polymer Preprints, 2001

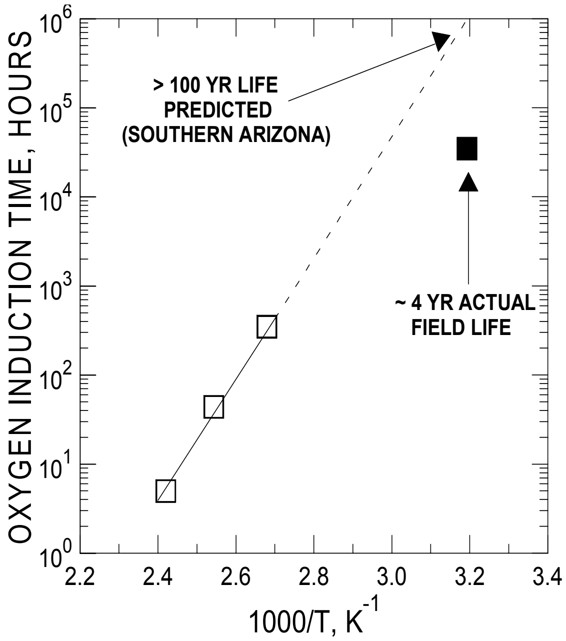

FIG. 1. Predicted versus Actual Life of a Polyethylene Material.

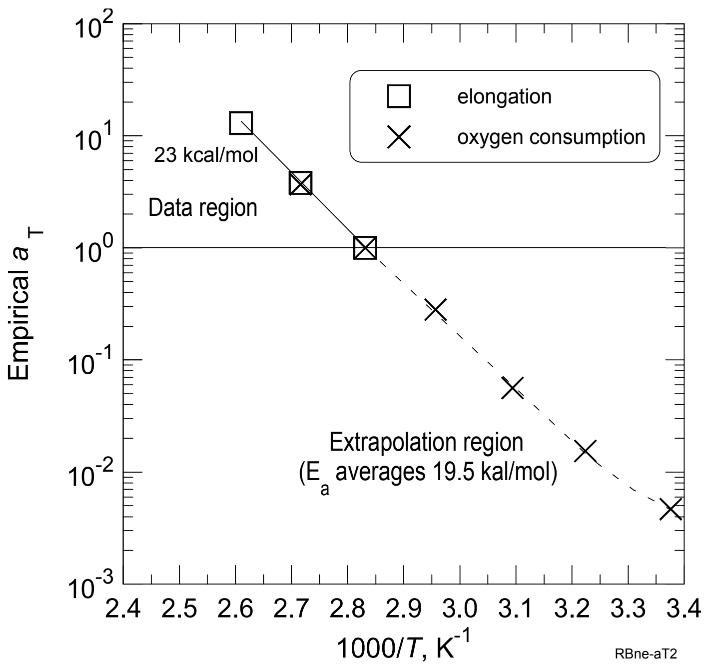

FIG. 2. Oxygen Consumption and Elongation at Break Results for a Neoprene. (23 kcal/mol = 1 eV; 19.5 kcal/mol = 0.85 eV)

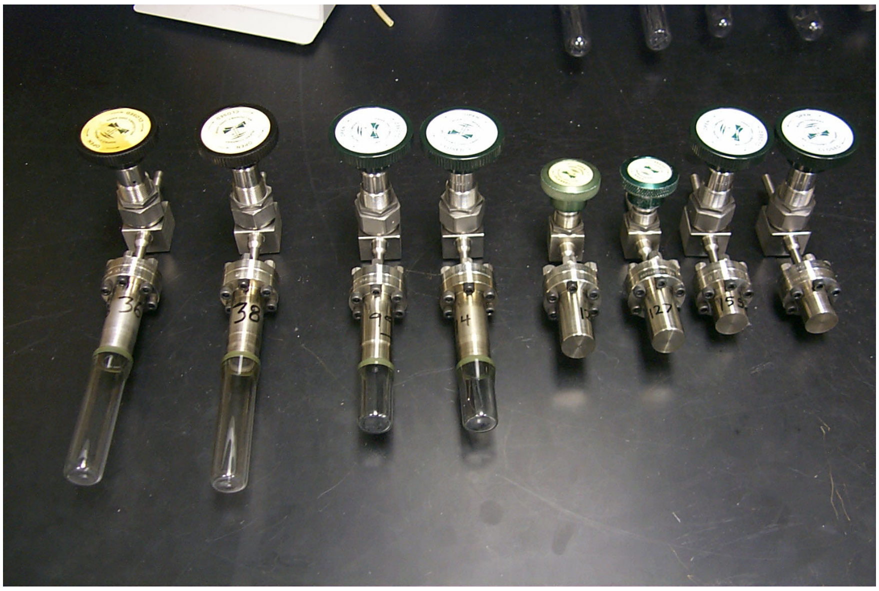

FIG. 3. Oxygen Consumption Sample Holders.

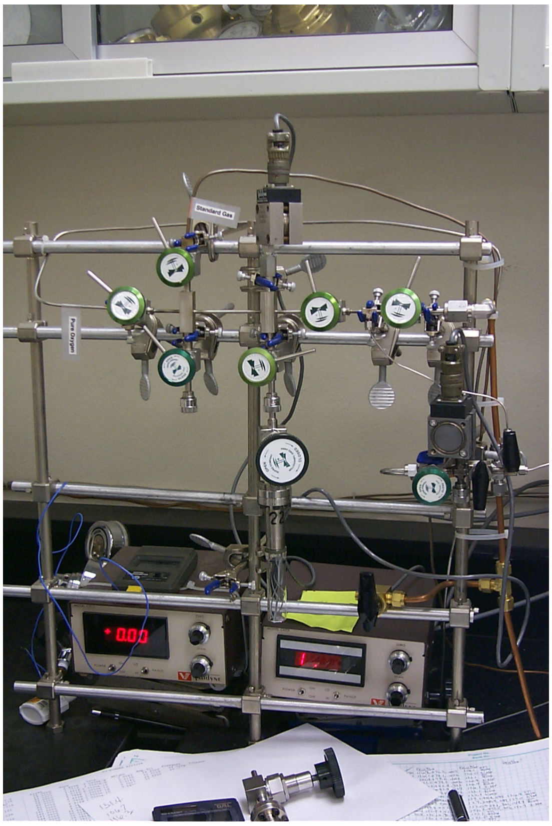

FIG. 4. Gas Handling Station.

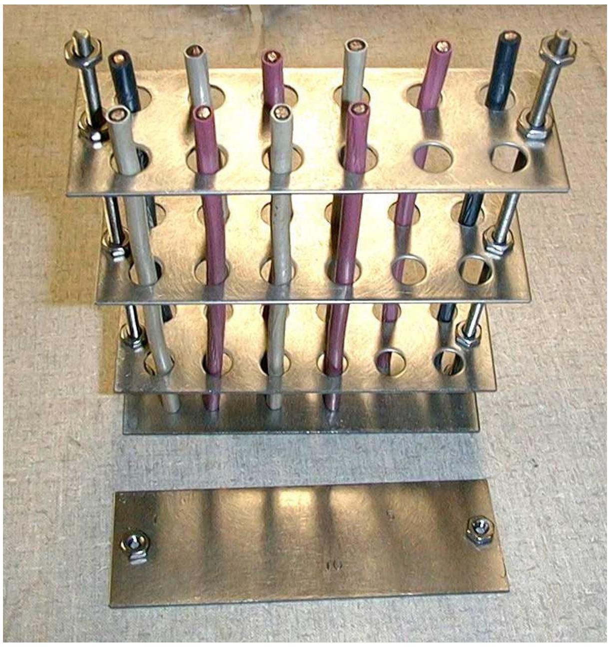

Figure 5. EPR/CSPE Fragility Test Specimens and Holding Rack.

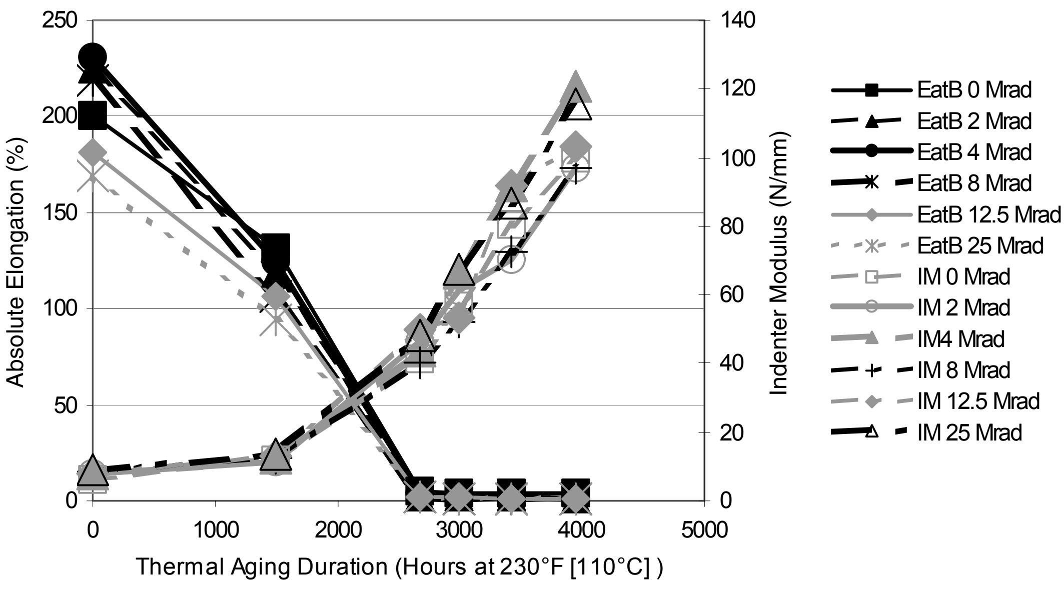

FIG. 6. Indenter Modulus and Elongation at Break for Okonite EPR/CSPE Specimens.

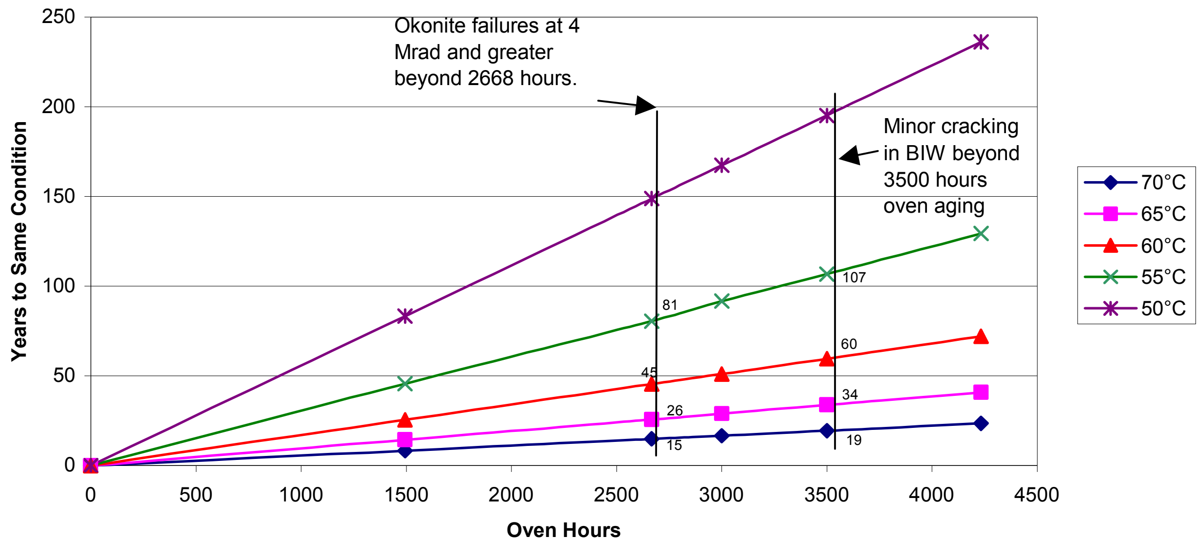

FIG. 7. *Fragility Curve for BIW and Okonite EPR/CSPE Insulations.*

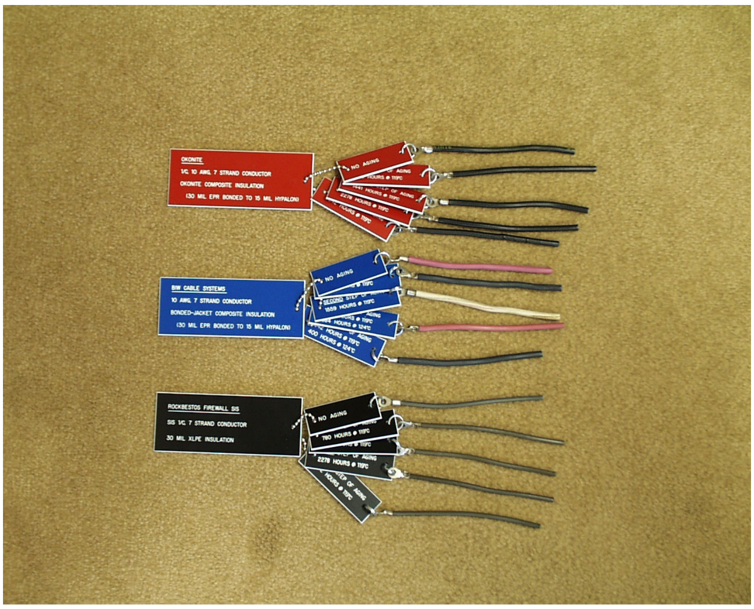

FIG. 8. Insulated Conductor Training Specimens (Okonite EPR/CSPE, BIW EPR/CSPE, and Rockbestos XLPE).

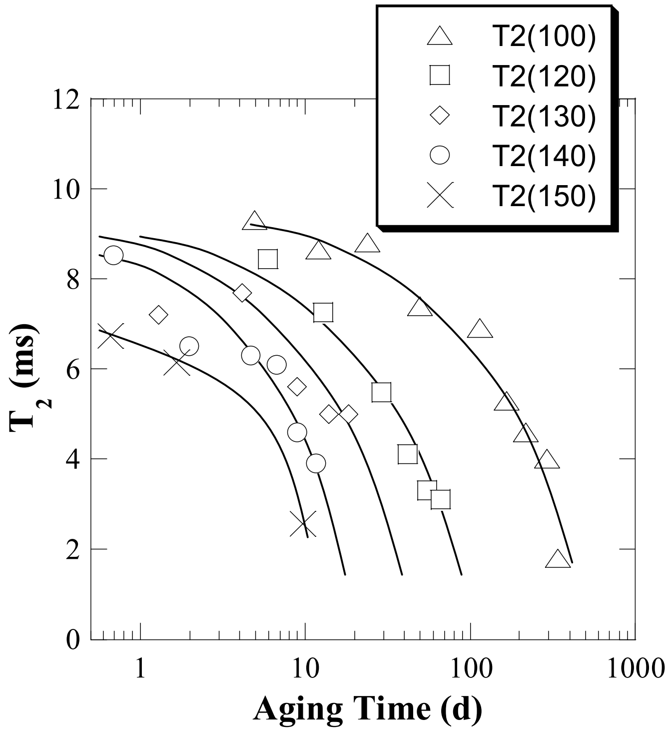

FIG. 9. NMR Results for an Anaconda CSPE Jacket Material.

THE ADVANTAGES OF RELIABILITY CENTERED MAINTENANCE FOR STANDBY SAFETY SYSTEMS

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Abstract

On standby safety systems, nuclear plants have to balance the requirements of demonstrating the reliability of each system, while maintaining the system and plant availability. With the goal of demonstrating statistical reliability, these systems have extensive testing programs, which often makes the system unavailable and this can impact plant capacity. The inputs to the process are often safety and regulatory related, resulting in programs that provide a high level of scrutiny on the systems being considered. In such cases, the value of the application of a maintenance optimization strategy, such as Reliability Centered Maintenance (RCM), is questioned.

Part of the question stems from the use of the word “Reliability” in RCM. When RCM is applied to a system maintenance program driven by reliability requirements, there appears to be redundancy. A deeper look at the RCM process, however, shows that the RCM goal is to ensure that the system operates “reliably” through the application of an integrated maintenance strategy. This is a subtle, but important distinction. Although the system reliability requirements are an important part of the strategy evaluation, RCM provides a broader context where testing is only one part of an overall strategy focused on ensuring that component function is maintained through a combination of monitoring technologies (including testing), predictive techniques, and intrusive maintenance strategies. Each strategy is targeted to identify known component degradation mechanisms.

The conclusion is that a maintenance program driven by reliability requirements will tend to have testing defined at a frequency intended to support the needed statistics. The testing demonstrates that the desired function is available today. Maintenance driven by functional requirements and known failure causes, as developed through a maintenance optimization assessment, will have frequencies tied to industry experience with components and rely on a higher degree of predictive monitoring. The testing in this strategy is part of an effort to ensure that the desired function is not only available today, but will be available tomorrow as well.

This paper considers the application of a streamlined form of RCM to the Emergency Core Cooling (ECC) and Standby Diesel Generator (SDG) Systems of a CANDU® plant.

Recently completed studies provide useful insight into the important value added of the systematic assessment approach (using RCM techniques) for these standby safety systems.

In the case of RCM analysis performed on the Emergency Core Cooling (ECC) System of Point Lepreau Nuclear Power Generating Station (PLGS), it was found that 60% of the current maintenance tasks are testing (functional, stroke, logic, and annunciation tests). Similarly, the SDGs have 50% of the maintenance tasks associated with testing. The paper considers how the results of the RCM analysis demonstrate that the analysis can be used to assist in the optimization of the testing program (as dictated by reliability) while also taking better advantage of the testing through condition monitoring and predictive maintenance techniques. Further, the results illustrate the importance of identifying and linking the different plant activities within a well integrated plant culture.

1. INTRODUCTION

In the world of maintenance, redundancy and operating context are considerations that impact significantly on the maintenance strategy for a component. Given two parallel components, with one always performing a standby function in case of failure of the second, normally running, component, the standby component will be subject to a more intensive program to ensure it is available when called upon. The level of intensity would be further impacted by the criticality of the component to system and plant functional requirements.

The example of a standby component can serve as a basis of understanding for the approach to be taken on a complete system whose primary function is to be in standby mode until called upon. Applying the comparison leads to the conclusion that the entire system would be subject to a higher level of intensity in the maintenance program. This intensity will be increased if the system is a standby safety system.

Typically, the Emergency Core Cooling (ECC) and Standby Diesel Generators (SDG) are two systems in a CANDU NPP that fall into this category of standby safety systems. As expected, these systems are subject to a high level of attention from the maintenance program. In fact, looking at the testing program, the ECC and SDG systems make up 15% of the operating manual tests (OMTs) for the Point Lepreau Generating Station (PLGS). The systems themselves are subject to regular assessments of reliability and the designs include a high level of redundancy.

This line of thinking leads one to believe that everything appears appropriate in the strategy development for these systems. In this light, it appears questionable as to the value of applying a detailed maintenance optimization process to these systems. However, recent studies of the PLGS systems have helped to highlight the advantages of the applying maintenance optimization processes to such systems.

2. EVALUATING THE EXISTING STRATEGY

As part of the maintenance optimization assessment process, the maintenance and performance history of the systems are reviewed. Two notable items were identified. The first item was that there were incidents of valve seat wear out on the ECC system, which can be directly attributed to exercising the valves. In a standby system, testing is the primary source of exercising these components. Consequently, the testing program intensity has been high enough to require valve seat replacement. The second item noted was that the SDGs have had several incidents of failure that could be attributed to failure causes detectable with the appropriate maintenance strategy. These two examples are notable because they identify two key areas of concern regarding standby safety system maintenance strategies. The first is that testing frequencies sufficient to cause valve wear out give rise to uncertainty as to the suitability of these frequencies. Second, the occurrence of potentially preventable failures leaves in question whether all the attention these systems receive is appropriately focused.

Considering the frequency question further, Table V shows the reconciliation of the current preventive maintenance program and maintenance optimization study recommendations. There were a large number of tasks (40%) for which it is recommended to modify to more relaxed frequencies. Most of the changes are focused on a decreased frequency of testing of the ECC system (monthly to quarterly) and diesel generator system (bi-weekly to monthly) and the vast majority of these modified maintenance tasks are part of the current operating manual tests (OMTs). The frequency modifications have been recommended based on standards, codes, operating experiences, maintenance history, and templates. Examples are the ASME-OM Code, US NRC Regulatory Guides, IEEE standards, IAEA publications, and EPRI/ERIN recommendations. It is noted that the reliability models are not directly considered in this case. The conclusion is that the statistical models are resulting in an increased level of testing than what might be determined from consideration of the component degradation itself.

To consider these observations from a broader perspective, it is useful to consider extreme maintenance strategies. In theory, a standby system could be maintained 100% of the time. Would this result in increased reliability? The answer is no, because the introduction of infant mortality problems and the increased wear rates alone would dictate a decreasing rate of return for the effort. This is compounded by the zero availability of the system. The alternative extreme is to ask what happens if no maintenance is performed. This leaves the system subject to random failure mechanisms, which results in reduced system reliability. However, the system would now have the potential for perfect availability. The conclusion is that there is a point between the extremes that optimizes both reliability and availability. We have also shown, that in order to properly assess maintenance strategies, there is a need to understand the degradation mechanisms of interest, and hence, the effective strategies to deal with the identified degradation mechanisms.

The extreme cases discussed above assume that the system maintenance frequencies are in fact working together as a single coherent strategy. When considering the frequency of maintenance on these standby systems, it is important to note that there is a significant difference between the testing and planned maintenance. The dominant driver behind testing intervals in these systems is reliability analysis. Here statistical estimation of the system reliability is made. The reliability models dictate a required testing frequency to demonstrate system function and prove reliability. The planned maintenance program is built in principal to meet the system's maintenance needs.

The difficulty associated with separate influences for testing and maintenance as a whole, is that this can be a source of discontinuity in the overall management of the system degradation. The reality is that the testing, preventive maintenance, and the surveillance activities all combine to provide the overall maintenance strategy. The conclusion is that in addition to finding the optimal maintenance effort to maximize both reliability and availability, the complete program must be well integrated, representing a comprehensive and coherent strategy. This strategy must then be carried out in an equally well-integrated plant culture to ensure the program is maintained in an optimal state and is executed in an effective manner.

Returning to the assessments performed for PLGS it was noted that the recommended tests are supplemented with monitoring and trending tasks to ensure functionality of the system and to provide a basis for taking corrective actions needed to maintain high in-service reliability of the systems. In Table V, more than 65% of the total added tasks involve system health monitoring and surveillance tasks such as predictive, trending operational and process parameters, walk downs, inspections, functional checks and verification. The remaining 35% are tests triggered when the result of condition monitoring tasks indicate eminent component/subcomponent failure. This illustrates the high level of interrelationship that exists in the system maintenance program. Many of these added tasks are not necessarily new to the plant activities, but they are now being identified as an integral part of the overall maintenance strategy.

This discussion has identified the need to be able to assess system degradation and provide meaningful strategy development to maintain the system. Second, the need to provide an integrated strategy has been seen. These are both areas where an RCM based maintenance assessment will provide the necessary structure to address these needs. The examples provided demonstrate how the studies carried out for PLGS have provided the necessary results to evaluate these standby systems.

3. UNDERSTANDING FAILURE MODES

3.1 General

It has been mentioned previously that in order to define the optimal maintenance strategy it is necessary to understand the degradation mechanisms or failure modes. The examples given above were the motorized valves of the ECC system and the standby diesel engines. Failure modes and their underlying failure mechanisms are discussed for the ECC MOVs and the SDGs to identify the potential for a more optimal frequency selection than dictated by reliability considerations alone.

When looking at failure modes, the interest is to define random failure modes and time based failure modes. Random failure modes have the characteristic of having an equal probability of occurrence at any time in the life of the component. The time based failure modes are further broken down into true time based failure modes and demand based failure modes. The difference can be subtle, but it is important. Consider a bearing as an example that may see higher wear under start-up conditions due to poor lubrication and changes in geometry due to non-standard temperatures. However, the same bearing sees a sustained, but lower level of wear during extended operation. The failure mechanism may be either demand or time based, depending on the dominant degradation source.

3.2 ECC Motorized Valves

With these explanations in mind, the failure modes of the ECC motorized valves can be considered. Typical failure modes of the motorized valve are, internal leakage (isolation impairment), external leakage, fail to open and close, and erratic operation (open/close spuriously). Table I shows the typical failure modes and the contributing mechanisms of motorized valves. The table identifies whether the various mechanisms are random, time based, or demand based. To explain the type of failure mechanism, each failure mode is considered in turn.

The first failure mode is associated either with the wear of the valve seat or the control mechanism failure. Wear or damage to the valve seat can be caused by testing the valve too much or by foreign material ingress. This mode is primarily a demand-based mechanism since the wear can only occur when the valve is either opened or closed. For the ECC, the valves are exercised for testing purposes. The case of debris in the system adds a random element to the failure potential. However, in this case, if the valve is not exercised, there is no seat wear and the function would be maintained.

The seat material could also be subject to degradation. This is primarily a function of seat material and the local environment. This is a time based failure cause impacting potentially on one or two failure modes. There are a number of parameters that can increase or decrease the effect, and depending on the specific component, these could be monitored as well.

Damage or wear to the valve seat would be exclusively demand based for isolation (normally closed) valve, because the only source of friction and wear at the valve seat would be when the valve changed position. For normally open valves there is an element of time dependence because of erosion by the fluid flow over the valve seat. However, since the fluid in the ECC systems is clean water, the erosive element is probably very small, and the friction and wear is dominated by the demands placed upon the valve to change the position.

Seals or packing failure usually leads to external leakage. Wear of the seal and packing may be caused by frequent exercise of the valve. Improper installation may distort the seal or packing, which may damage and eventually leads to leakage. There is also an effect of the environment leading to a true time based element to the failure mode.

Failure to open and close may be caused by a variety of failure mechanisms. The control mechanism may fail to energize the actuator when required, or may drive the valve in the wrong direction. Similarly, corrosion, loss of lubrication, stem binding, sticking and dirt accumulation may also contribute to this failure mode. Reduced exercise may also restrict the movement of the valve. It is worth noting that many of the mechanical failure mechanisms can be attributed to both exercising (ie demand based) and not exercising (ie time based) the valves.

The actuator failure is regarded as a demand based failure mechanism, because during normal operation the actuator is not under stress unless the valve is triggered to change position. When actuated, the actuator is subjected to both mechanical stress to overcome the friction in the valve and electrical stress to compensate for high motor starting current. This includes failures of the motor, torque switches and the mechanical components. The actuator may also fail from lubricant contamination. The contamination could be either from external sources that build up until it causes a problem or could be from deterioration of the oil or grease due to the exposure of heat and radiation within the actuator. In both these cases the failure is time based and independent of demands placed on the valve.

The control mechanism failures are likely due to the failure of limit and torque switches, which can be both random and demand based failures. In this case, the real consequence of the switch failure needs to be understood and accounted for in assessing the impact on reliability.

The last failure mode is associated with the control mechanism failure that energized the actuator when not required. Mechanical defects may also lead to a change in the opening and closing position of the valve. A failure of a valves internal mechanism that leads to a spurious opening or closing of the valve is likely to be random event occurring in time with low probability.

Testing may contribute to the wear of the control components (switches, relays, electrical contacts etc.) and valve internals (fracture, stem/disc separation, packing hardening, and weld failure). Solid-state contacts are subject to degradation in some environments, which would be strictly time based. Any electronic components can be subject to random failures as well.

From the review of the work history at PLGS for the motorized valves until the year 2000, it is evident that more than 35% of the failure/maintenance tasks were attributed to the torque/limit switches calibration and replacement, 20% addressed to actuator and 20% to other electrical components, 15% to packing/seal, and 10% to others. Likewise, AECB and EPRI reports [2] and [3] summarize experience with failure of motorized valves in Canada and United States, showing a similar distribution. The AECB report is based predominately on Ontario Power Generations plants. It covers failures of all types, but lack in providing specific information on the failure mechanisms. The EPRI report concludes that the major portions of faults are caused by mechanical and electromechanical failure. Table II compares the percentage of failures taken from PLGS maintenance history and the EPRI report.

The failure modes, as described above, indicate a mix of demand, time based, and random failures. A reliability study conducted for CANDU safety systems [7], found that about 80% of the total observed failures of the motorized valves in ECC system are attributed to the demand failures, as shown in Table III. This leads to the conclusion, that a reduction in testing, being the primary source of demand, should be able to reduce the overall failures and maintain a net higher reliability.

3.2 Standby diesel Generator

For the standby emergency diesel generator systems, the most dominant failure modes are fail-to-start and fail-to-run. Engine, cooling, and fuel oil subsystem failures are the most likely to result in fail-to-run. In a recent study [4], it is reported that the largest set of complete failures (62%) occurred in the fail-to-start group. The most likely root cause is design, manufacture or construction inadequacy. In another reliability study, the new plants are reported to have statistically higher failures per year than that for the older plants [5]. Table IV gives some typical failure data for CANDU stations reflecting similar results.

Most of the faults are focused in the instrumentation and control system of the diesel generator system, which contributes a significant portion to the fail-to-start mode. It is noted that initial design faults and modifications made subsequent to the original installation introduced additional failures. Due to the complexity and function of the instrumentation and control subsystem, the common failure modes are especially susceptible to the human factor. Procedures, maintenance, operation, and testing all contribute to this root cause.

It is interesting to note that testing has been the primary way to detect common cause failures. Due to a lack of operational data and critical parameters for trending, the newer plant has to undergo a series of regular testing. There are a number of parameters that need to be monitored and trended to determine the status of a diesel generator. A recommended list of the monitoring and trending parameters is presented in Table C.1 of IEEE STD 387-1995 [6].

Typically, the diesel engines themselves see a greater degree of degradation during startup associated with lubrication and temperature changes. The diesels are designed to run for long periods of time, but the mode most common for the plant application is relatively short running periods under less than ideal conditions. Even the control equipment is under increased stress as they are called upon more frequently under startup conditions than catered for in the design.

Considering the failure modes of the standby diesels, there is a different set of trends, pointing to the system being subject to infant mortality. Further, considering the system as a whole, the diesels are likely to suffer from a testing frequency that is excessive. It is clear that from a careful consideration of the failure modes and the underlying mechanisms is an important contributor to defining the optimal strategy for these systems as well.

4. AN INTEGRATED RELIABILITY PROGRAM

As noted previously, integrating the maintenance program into a comprehensive strategy is the desired result of a maintenance review process. It has been noted that the purpose of the periodic testing is not solely to perform the immediate "health check" of the component. A test that demonstrates that the component is fit today does not guarantee that the component will not be unfit tomorrow or in the near future. In the case of a slowly developing (impending) aging/degradation mechanism, a simple test for function may miss the reality that the component condition is deteriorating and will be unable to perform its function before the next test. However, by simply monitoring and trending of testing parameters the long-term degradation mechanisms can be managed more effectively. The regular review and use of these recommended parameters increases the system reliability and reduces aging concern by detecting the potential for substandard performance early before actual failure occurs.

The appropriate determination of which parameters should be trended requires knowledge of the degradation mechanisms. Further, an understanding of the progression of degradation will further enhance the effectiveness of the process to the point where well founded predictive models can be developed. Many mechanisms are non-linear in nature, which may be more difficult to capture with simple trending practices. Further, there are numerous parameters that can improve or accelerate the degradation, and if these are not understood nor monitored, the accuracy of the model is limited. Finally, the margin available in the system and component performance can impact on the level of effort required to detect problems and mitigate consequences with acceptable risk. These factors contribute to the selection of an appropriate strategy that identifies what parameters to monitor, how they are combined to trigger condition based maintenance tasks, and the frequency of data collection required.

Typically, standby safety systems are subject to reliability assessments and many of the testing frequencies derive from regulatory requirements. The difficulty, as noted earlier, is that the need to demonstrate reliability can result in the testing frequencies that potentially contribute to enhanced degradation of system function, and therefore reduce system reliability. There is a need to reconcile the maintenance needs of the system derived from knowledge of degradation mechanisms, and the need to demonstrate system reliability.

One step taken at PLGS to bring this reconciliation closer to reality is the unification of efforts required to assess reliability, and determine the requirements for maintenance and system health monitoring programs. It was recognized that both Functional Failure Analysis (FFA) and Failure Modes and Effects Analysis (FMEA) are common starting points for all three programs. Although some of the details change to address specific needs in each area, essentially a common procedure can be developed. When applied, there is a common system assessment relating system functions through to component failure modes that can be input into any of these programs. Further, the same procedures can be applied to any new design at the station.

The derivation of a model that identifies failure modes for use in both reliability models and maintenance strategy development is a good starting point to integrate the programs. The next step is to integrate the results of the reliability assessment and the maintenance assessment. The reliability study will identify a number of tests and their frequencies needed to demonstrate reliability. The tests, based upon failure modes also identified for the maintenance strategy development, should be easily integrated into the maintenance strategy as functional tests. The maintenance perspective can be used to augment the test procedures to ensure that parameters identified from the more detailed failure cause assessment are captured. That is, the test is more comprehensive in that it captures the information for both needs. Further, the functional test can then become part of the overall strategy that also identifies the necessary level of trending or predictive assessment required to relate the test to prediction of potential failure.

These first two steps ensure that the reliability program requirements and the maintenance program strategy are well integrated. However, this does little to affect the frequency requirements derived from the reliability assessment. Through the process of integration, the “ideal” frequency from a maintenance perspective can be identified and compared against that derived from reliability considerations. This is the approach taken in the maintenance optimization studies for PLGS. Earlier it was noted that there were numerous recommendations to relax frequencies. Ultimately, the reliability perspective, with its regulatory ties, must define many of the frequencies. However, the comparison identifies areas where the reliability model should be assessed for potential relaxation. This, in itself becomes a means of feedback and allows for the opportunity to continually improve in a controlled manner.

To make a more direct impact on frequencies, the reliability model requires new information to justify the frequency relaxation. There is a concept that provides just that kind of change. Currently, failure probabilities do not account for the capabilities of the maintenance program explicitly. It is proposed that if the effectiveness of a maintenance strategy could be quantified in terms of probability of capturing failure before it happens, then this mitigating factor could be introduced into the reliability model. This idea requires quantifying the level of aging degradation coverage and the capability of the strategy to capture the failure mechanism of interest.

This concept of assessing “coverage” is already part of the one industry preventative maintenance tool. This tool compares the selected maintenance strategy for how well it covers the known degradation mechanisms identified in the tool. Therefore, the information is already appearing in a tangible form. It is suggested that as maintenance strategies continue to be enhanced by experience, research, and detection capabilities, there will be a demonstrated ability to create defendable probabilities with acceptable uncertainties to evaluate different maintenance strategies. These probabilities can then be incorporated into reliability assessments, resulting in relaxed testing frequency requirements. This would truly allow for reconciliation of the statistical model requirements with those derived from maintenance considerations.

5. CONCLUSION

Standby safety systems are correctly subject to a high level of scrutiny and attention from both the reliability perspective and from the maintenance perspective. In answering the question as to the value of maintenance optimization assessments for such systems, it is noted that specific failures lead one to question whether testing frequencies are appropriate for enhancing reliability and whether maintenance strategies are effectively capturing all the failure mechanisms.

Two considerations that derive from maintenance optimisation assessments give rise to further uncertainty of the ability of reliability based test frequencies to result in an enhancement to reliable system operation. First, based purely on maintenance considerations, there was a general trend to recommend relaxed testing frequencies. Second, from consideration of failure modes and their underlying mechanisms, it is evident that there are a number of potential failures attributed to demand based degradation. Some assessments have indicated that this is a significant contributor to observed failures.

Finally, it is suggested that steps can be taken to resolve some of this uncertainty. Unifying the FFA and FMEA, as has been done at PLGS is good step to bring reliability concerns and maintenance strategies together. This step needs to be followed by an integration of the results into a single comprehensive strategy. This strategy must be executed in plant culture that facilitates the flow of information and continued improvement of the strategies developed. Finally, with the advances in the maintenance world, it is now possible to ascribe probabilities of success to maintenance strategies. This then becomes the means to ultimately bring the reliability requirements maintenance strategies developed for reliable operation in line with each other.

REFERENCES

[1] Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plant. NUMARC 93-01, Revision 2, April 1996.

[2] Aging and Malfunction of Valve in Special Safety System, AECB, Project 2.115.1, AECB-INFO 0303,1989.

[3] Foster; Miller Inc. Valve Motor Operator Improvement", EPRI Report NP-5696, July 1989.

[4] ICDE Project Report on Collection and Analysis of Common Cause Failures of Emergency Diesel Generator, NEA/CSNI/R(2000)20, Feb 2001.

[5] UNRC,GL 84-15, July 1984.

[6] IEEE Standard Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations, IEEE-Std 387-1995.

[7] Smith J.E et. Al., "A Bayesian Reliability Study on Motorized Valves for the Emergency Core Cooling, Heat Transport Isolation and Shutdown Cooling Systems at Gentilly-2 Nuclear Generating Station, 17th Annual CNS Conference, Fredericton N.B., 1996.

Table I. Failure Modes And Mechanisms of Motorized Valves

Failure Mode	**Failure Mechanism**	**Mechanism Type**		
		Random	**Time Based**	**Demand Based**
Internal Leakage	Control failure-not fully closed	Yes	Yes	**Yes**[a]
	Valve seat worn or damaged	Yes	Yes	**Yes**
External leakage	Seal/Packing worn, distorted or damaged	No	Yes	**Yes**
	Valve body cracked	No	**Yes**	Yes
Open or close spuriously	Control spuriously energizes actuator	**Yes**	Yes	Yes
	Internal failure	No	Yes	**Yes**
Fails to open or close when required	Controls fail to energize actuator	**Yes**	Yes	Yes
	Valve mechanism jammed-corrosion etc.	No	**Yes**	Yes
	Valve mechanism jammed-foreign matter	Yes	**Yes**	Yes
	Actuator internal failure	No	Yes	**Yes**
	Actuator contaminated	**Yes**	Yes	Yes

a: Bolded entries indicate estimated dominant mechanism type

Table II. Failure Cause Breakdown of Motorized Valves

	PLGS	EPRI
Mechanical • fail to operate • actuator • packing/seal • bent stems • damage to valve seats • gear binding and damage	35%	22%
Electromechanical • torque switch failure • torque switch adjustment • limit switch adjustment	35%	32%
Electrical • motor • contacts • MCC and others	20%	27%
All others • vibration • wear • lubrication • unknown	10%	19%

Table III. Number Of Failures Of Motorized Valves In ECC System Of CANDU Reactors [7]

Plants	***No. of Failures***	***Time Based***	***Demand Based***	***Operating Time (h)***	***No. of Demands***	***Time Based Failure (f/1E6h)***	***Probability of Failure per Demand***
Point Lepreau	58	12	46	6216096	8850	1,93	0,0052
Gentilly-2	36	8	28	5040864	6600	1,59	0,0042
OPG	216	44	172	31325760	34680	1,40	0,005
Total	310	64	246	42582720	50130	1,50	0,0049

Table IV. Typical Standby Diesel Generators Failure Analysis Data For CANDU Systems

		Starting		***Running***		***Unavailability (hours)***	
Plants	***Units***	***Attempts***	***Failures***	***Hours***	***Failures***	***Forced***	***Maintenance***
Point Lepreau	2	1386	74	9798.7	50	993.09	3029.57
Pickering A	6	2083	59	5576.23	40	33045.98	48208.16
Pickering B	6	1876	7	6276.66	11	9290.56	30995.01
Gentilly 2	4	5032	54	12340.1	-	3154.41	2967.31
Bruce A	4	1413	21	3194	51	14823.4	66873.3

Notes:

- The starting failures include both test response and emergency response failures.
- Point Lepreau data collected over first 12 years of service (ie 1982- 1993)
- Pickering A & B data is for the last 12 years (1990-2001)
- Gentilly 2 service lifetime data. Running failure data not available.
- Bruce A based upon eight years of data (i.e. 1990-1997)

Table V. Reconciliation of Current Maintenance Practice With Streamlined RCM Analysis

RCM Analysis Recommendation	***ECC System Number of Tasks***	***Emergency Diesel Generator Number of Tasks***
MODIFY	241	215
ADD	378	177
RETAIN	236	119
DELETE	9	32
TOTAL TASKS	864	543

EXPERIENCE WITH SAFETY I&C MODERNIZATION AT PAKS NUCLEAR POWER PLANT

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Abstract

Modernization of the reactor protection system represents a technical and safety technology backfit of decisive significance within the safety enhancement measures of Paks Nuclear Power Plant. Installation and commissioning of the new reactor protection system was successfully performed in Units 1, 2, 3 and 4 during the annual outages in 1999, 2000, 2001 and 2002, respectively.

A very large contract for extensive refurbishment affecting all four reactor units and the associated full-scope simulator was awarded to Siemens AG, now Framatome ANP GmbH, in a two-step tendering process. This project has several features that are significant for TELEPERM XS projects:

- The four units were to be equipped with a digital reactor protection system (RPS) of the same design based on the process system requirements specified by Paks Rt.
- Every year the RPS of one plant unit was to be replaced during a normal extended outage.
- The units were to be started up immediately after implementation of the new system without on-line open-loop testing.

This paper outlines the reasons for the refurbishment, the refurbishment scope, the structure of the reactor protection system, the verification & validation methods used, the planning and implementation of installation and commissioning, as well as special features of the TELEPERM XS system platform.

INTRODUCTION

Paks Nuclear Power Plant, the only nuclear power plant in Hungary, is situated on the Danube River in the middle of the country, 110 km to the south of Budapest. The power plant consists of four VVER-440 Model 213-type pressurized water reactor (PWR) units built according to the same concept. Construction started in 1976 with the first unit being connected to the national grid in 1982 followed by another unit each in 1984, 1986 and 1987.

The units are not completely identical at the equipment and device level. In addition to a few main electrical components, the unit computer and a large number of sensors and transmitters for Units 3 and 4 were manufactured in Hungary.

The process control systems of the units were designed in the 1970s, but the requirements for the safety systems changed significantly in the period preceding construction. The related I&C design was adapted to the new requirements and the I&C drawings were completed during the construction of Units 1 and 2.

Since 1987 several safety enhancement measures have been implemented at the Paks plant. Preparing for the I&C upgrade has been a continuous engineering activity at Paks since the end of the 1980s. Activities currently underway include equipment replacement, some system modifications as well as development of new systems. The upgrading strategy has concentrated on the most obsolete equipment, which is to be upgraded due to its importance within the control system.

In 1992, after 10 years of operation, refurbishment of the safety I&C systems – namely of the Reactor Shutdown System, the Emergency Core Cooling System, the Ex-Core Neutron Monitoring System and the Containment Isolation System – became a prevalent issue.

THE ORIGINAL REACTOR PROTECTION SYSTEM

The original Reactor Protection System (RPS) represented that of the VVER Model 213 design, with minor modifications, and complied with the Soviet OPV 82 Safety Standard. The various safety functions and tasks were implemented by the following autonomous systems in the structure shown in Figure 1.

⇨ Reactor Technological Protection System (RTPS)
⇨ Neutron Monitoring System (NMS)
⇨ Emergency Core Cooling System (ECCS)
⇨ Reactor Protection Central Cabinets (RPCC)
⇨ Diesel Load Sequencer (DLS)
⇨ Reactor Power Limiting System (RPLS)
⇨ Steam Generator Protection System (SGPS).

The RTPS and the NMS had two trains, whereas the ECCS had three completely identical and independent trains of 100% capacity each. The two trains of the RPS and the three trains of the ECCS were of triplicate redundancy on the input side, which meant 15 times sensing and initiation criterion generation for identical parameters. The input signals were voted 2-out-of-3 after set point comparison. The safety systems were based on relays and on standard measuring instrumentation for process parameters. The NMS used analog and digital integrated circuit boards dating back to the 1970s.

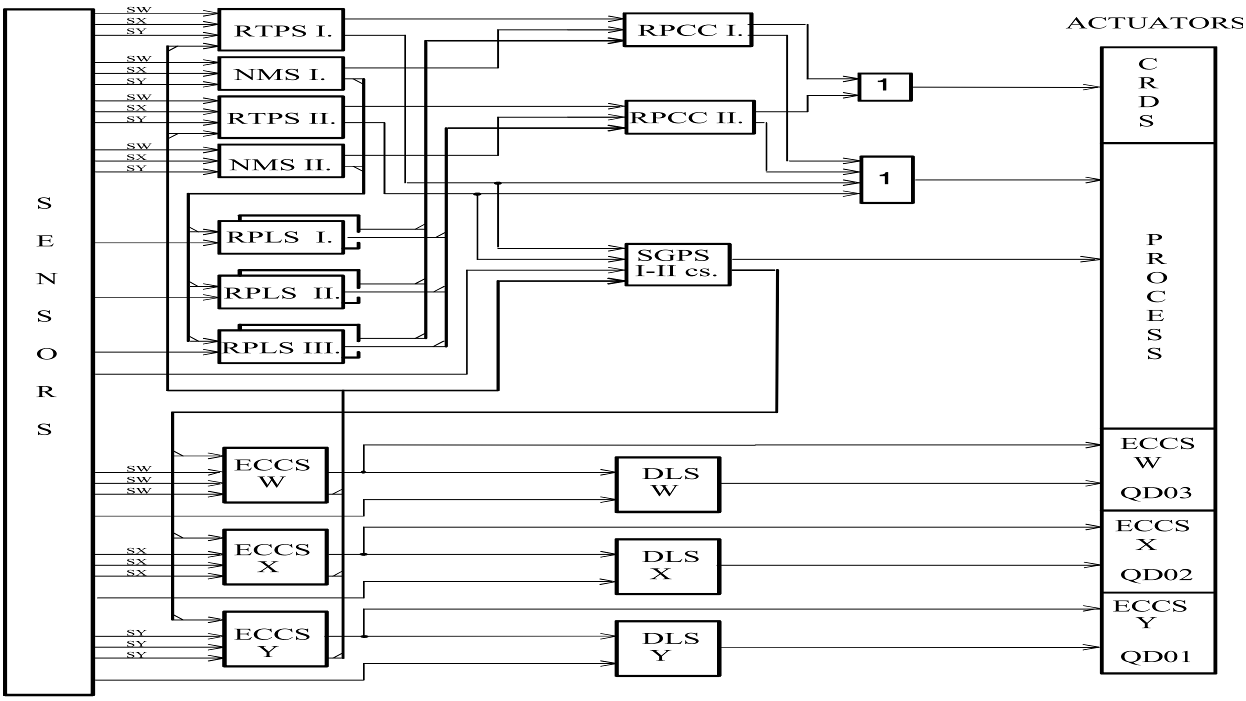

FIG. 1. Original safety I&C structure.

REASONS FOR REFURBISHMENT

At the time when the decision was taken in favor of refurbishment, it was recognized that the nuclear plant would soon have to be adapted to increasingly stringent safety and licensing requirements. The authorities' requirements for the new reactor protection system were not precisely defined and the licensing procedure had also become more demanding. It was essential that the new protection system should satisfy all of the authorities' requirements, as well as all applicable national and international codes and standards.

A number of studies and safety analyses had recently been performed for the first and second generations of VVER units in order to define their safety features and deficiencies.

In 1994 the AGNES (Advanced and General Evaluation of Safety) Study was performed specifically for the Paks plant and was to constitute the basis for evaluation. This study identified the deficiencies and weak points of the existing I&C systems. Within nuclear technology it is the field of I&C in which risk reduction and compliance with the latest safety requirements can be achieved in the most cost-effective way.

Equipment obsolescence and growing maintenance costs

Some obsolete equipment and a shortage of spare parts were causing great problems. The volume of maintenance work required for ensuring a level of reliability in line with expectations had reached a rather high level.

A significant volume of preventive maintenance was required for the electromechanical comparators (recorders equipped with microswitches operating on the principle of compensation) and a few relay circuits. In 1990 the annual replacement of comparators in the safety systems was introduced as a preventive maintenance measure. This entailed removing a sufficient number of comparators from a certain train, subjecting them to thorough repair, maintenance and laboratory functional testing and then re-installing them in the same train or another train during the next outage.

Life extension and environmental qualification

The physical and moral lifetime of I&C equipment is generally much shorter than the design service life of process components. Aging management in the field of I&C basically comprises equipment upgrading, replacement or refurbishment at the system level.

According to nuclear safety regulations, the operating licenses of the individual units have to be renewed every 12 years. One of the main conditions for renewing the operating license is that it must have been demonstrated that the equipment concerned can withstand the environmental conditions defined for a design basis accident in the latest nuclear codes and licensing regulations. Other major environmental criteria are the seismic resistance of safety-related equipment and systems as well as protection against electromagnetic interference.

Within the scope of refurbishment, the certification of all affected equipment and components is a major requirement.

Functional considerations

In addition to the refurbishment of component technology, protection functionality has been significantly upgraded. The functional modifications were needed for the following reasons:

- The designer of the original system was constrained by the functionality provided by the components available at the time. Naturally, the I&C tools of our times facilitate more exact and more efficient event detection and actuation.
- The significant amount of operating experience accumulated by the plant operator, the evaluation of which resulted in a number of functional changes.
- In the early 1990s, Hungarian scientific institutes prepared a probabilistic safety analysis (PSA) model of the nuclear plant and evaluated plant safety. The PSA report points out the weak points in the functionality as well.
- International literature has been extensively discussing the safety systems of the VVER reactors and has put forward several recommendations. Evaluation of these international recommendations also induced functional modifications.
- In 1992, the GRS report evaluating the safety of Unit 5 at Greifswald was issued which discussed the safety features of the VVER-440 Model V213 reactor in detail.

The most significant functional modifications are listed below:

- Cancelled EP 2
- Introduction of functional diversity
- Consistent sequencing of reactor shutdown and ECCS
- Semi-automatic neutron flux setpoint adjustment
- Primary-to-secondary leakage management
- Primary overpressure protection function design
- Implementation of the 30-minute rule.

Analysis of the individual modifications and evaluation of the overall functional modification package were performed by KFKI, the research institute of the Hungarian Academy of Sciences.

SCOPE OF REFURBISHMENT

Paks Nuclear Power Plant was originally equipped with I&C systems designed in the 1970s for the VVER-440 Model V213 reactors. A characteristic feature of these systems is the implementation of safety I&C functions using several autonomous subsystems. A simplified schematic diagram of these subsystems is shown in Figure 1. The figure illustrates how the system structure was split up to a very high degree. This resulted in a disproportionately high degree of redundancy on the sensor side. The figure also shows the physical scope for refurbishment.

Upgrading of the other major I&C systems to the extent required, which depends on their importance in terms of safety and economics, is a task for the coming years.

STRUCTURE OF THE SYSTEM

To train the personnel it was necessary to integrate the reactor protection system (RPS) model into the plant simulator. Whenever a hardware upgrade was performed, a model of the refurbished RPS function was integrated into the simulator so that it could be operated with the old RPS model as well as with the new one for training purposes.
The structure of the refurbished system differs fundamentally from the structure of the system based on the original Soviet design. The old system was characterized by the use of several autonomous systems in various redundancy structures. The new RPS is an integrated system consistently divided into three trains, which ensures that functional diversity is pursued within each train. The main requirements for the design of the architecture were as follows:

- Adjustment to the I&C task to be implemented
- Fulfillment of all deterministic safety requirements
- Fulfillment of desired probability requirements
- Fulfillment of environmental requirements, also considering the specific conditions of the plant environment
- Use of I&C hardware and software tools qualified for state-of-the-art safety applications.

A schematic of the architecture illustrating the underlying design concept is shown in Figure 2. As indicated in the figure, a specific parameter is measured by one sensor per train, i.e. by altogether three. After preliminary sensor signal processing, each train transmits the analog values to the neighboring trains via point-to-point communications links. Thus three analog values are available for each train.

The 'a' and 'b' computers implementing diverse functions have no logical interconnection. Accordingly, side "a" and side "b" are not connected to each other, except for the signal OR gating on the outputs for the purpose of reactor trip.

Cases do, however, exist in which a parameter is to be measured for both side "a" and side "b" due to a need for functional diversity. In such cases the sensors were duplicated, meaning that a total of six sensors measure the given parameter.

In accordance with the three physically separate Emergency Core Cooling Systems of 100% capacity each, the I&C system also includes three voting computers. Each of these computers receives data from the data acquisition and processing computers of the three trains and, after 2-out-of-3 voting, issues an actuation command to its dedicated ECCS.

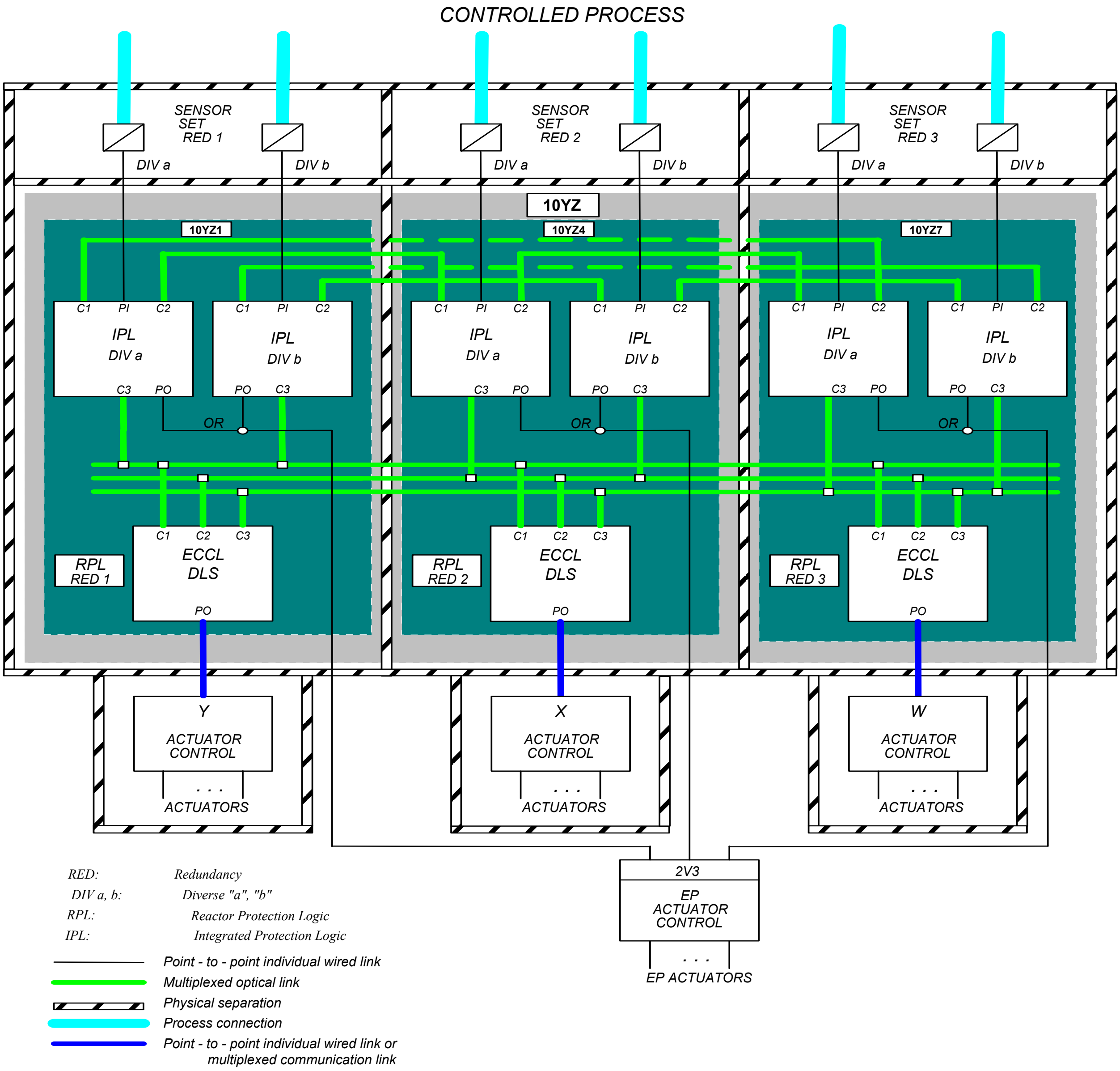

FIG. 2. New safety I&C structure.

PLANNING AND IMPLEMENATION OF INSTALLATION AND COMMISSIONING

A basic requirement from the nuclear plant operator was that the RPS refurbishment be performed during the scheduled annual outages; i.e. no additional outage was allowed.

Accordingly, the installation activities were carried out during three scheduled outages. In the first two shorter outages (approx. 30 days each) we performed all preparatory work that did not affect unit operation but required reactor shutdown. In the period immediately preceding the third outage, the cabling work and installation of the new components were performed in the unit while it was still in operation, wherever this was possible.

The third outage comprised an outage performed once every four years which involved complete discharge of the reactor core and lasted approximately 60 days. The completely unloaded reactor facilitated simultaneous work on all three safety trains.
Prior to removal of the old RPS, we ensured provisional manual operability of the actuators needed for an undisturbed outage regardless of the condition of the RPS.

Installation of the new RPS took around 30 days. The remaining outage time was used for system commissioning.

First the central portions were commissioned autonomously. During this part of the commissioning process Framatome ANP demonstrated that the equipment had not become damaged during shipment and installation.

During loop-level commissioning of individual components, each and every measuring circuit was tested by simulating live input signals and thus initiating real output signals for actuation. At the outputs, the operational tests of the individual components were performed by eliminating the provisional state of the actuators. These tests required more personnel and careful organization as well as adaptation to the specific conditions of the plant unit. As experience showed, the inherent features of the TELEPERM XS system greatly contributed towards making these activities easy to perform. The operational tests are tests that are performed each year during unit startup but with certain additional items the first time they are performed.

Approximately six months prior to installation, the theoretical training and simulator exercises for the personnel were started. This training proved to be good preparation for the commissioning activities during which each operator had an opportunity to gain hands-on experience in the operation of the system.

Thanks to the thorough preparations and carefully executed training program, installation and commissioning of the RPS did not cause any unexpected loss of operation in any of the plant units.

V&V METHODS EMPLOYED

The design process was defined by the project's Development Plan. The Development Plan detailed the project design activities by supplementing and refining the life cycle model of IEC 880. The verification and validation (V&V) activities assigned to the design activities were defined by the project's V&V Plan. The specified control tasks were perfectly consistent with the principles set forth in the international standards and had the following project-specific features:

- Apart from its own in-house verification process, Framatome ANP's documents were verified by the experts from the Paks plant as well as by their contractual partner, the Institute for Computer Technology and Automation of the Hungarian Academy of Sciences.
- The documents supplied by the customer (functional and technical requirements, conventional I&C specification) were verified by the experts and contractual partners of the plant as well as by Framatome ANP as the main supplier.

- The Paks experts prepared the software model of the new RPS in the full-scope simulator. This model was developed in a diverse way, separate from software design for the reactor units. The open-loop and closed-loop testing of the model demonstrated the adequacy of the functional specification.
- The standard features of the TELEPERM XS system were type-tested by the relevant German institutes (GRS ISTEC, TÜV NORD) as part of the qualification process. The customer only checked that the relevant certificates existed.

Framatome ANP's supply contract for the four reactor units also included delivery of a representative configuration which was installed next to the full-scope simulator in Paks. The resulting testing environment enabled us to demonstrate system adequacy prior to implementation via open-loop and closed-loop testing of the software for the real reactor units running in the TELEPERM XS hardware environment.

LIFE MANAGEMENT FOR TELEPERM XS PLATFORM

I&C systems generally have a limited lifetime. As an example, Figures 3 and 4 show the life cycle of safety I&C systems in Germany. Figure 4 shows that even digital I&C systems are now "cancelled products"; i.e. production has ceased and modules are only available nowadays for spare parts.

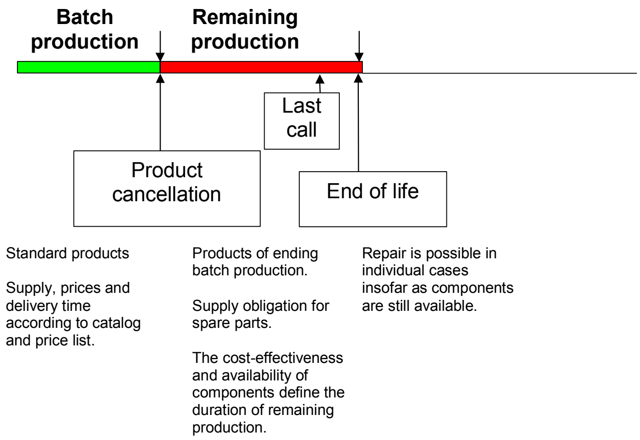

FIG. 3. Life cycle of I&C systems.

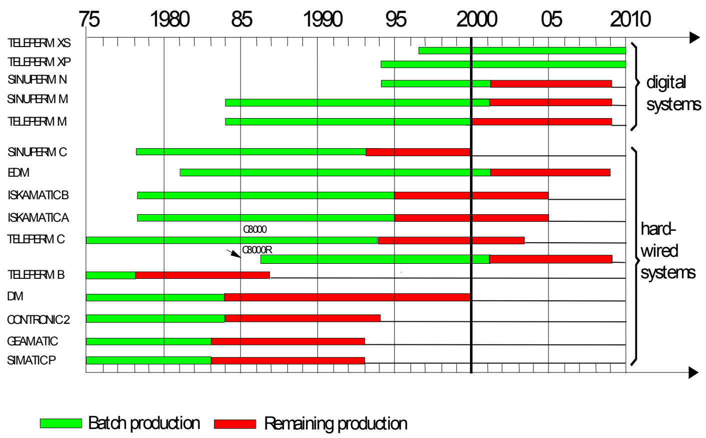

FIG. 4. Overview of safety I&C system platforms in Germany.

Another aspect which we can identify from the figure is that a system usually has a life span of only a little more than 20 years. For digital systems especially, the life span of components is far shorter. Intel CPU chips, for example, change every three years. By taking special measures that are shown in the next figures, Framatome assures a long lifetime for the TELEPERM XS platform.

Figure 5 shows the various modules of the TELEPERM XS platform. This division into modules has the advantage that only part of the system has to be qualified to KTA 3501 or IEC: namely, the hardware and the TELEPERM XS system software.

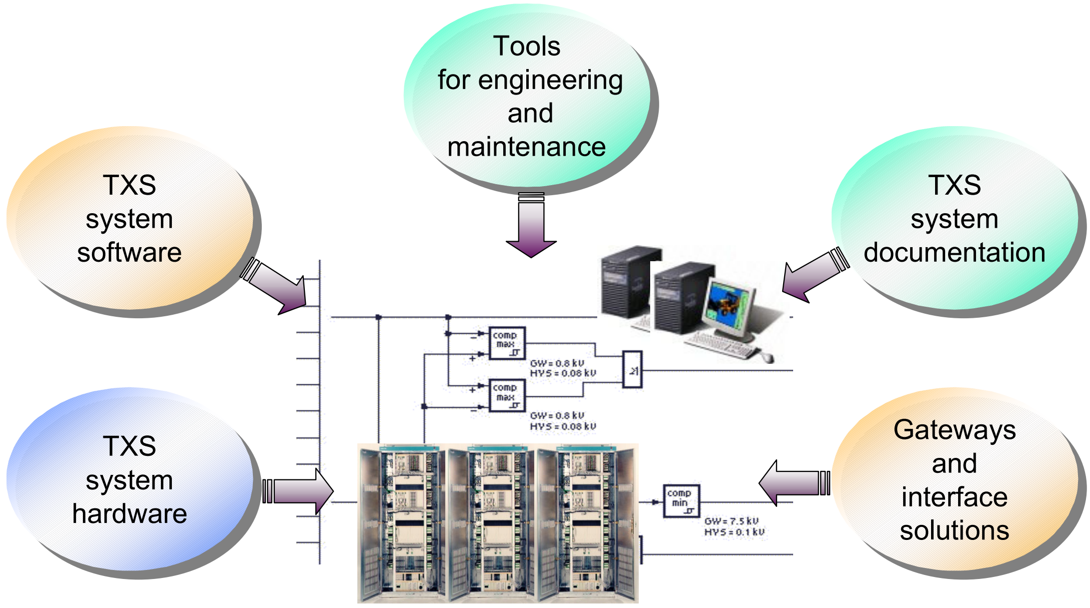

FIG. 5. Modules of the TELEPERM XS platform.

Figure 6 shows the objectives pursued by Framatome ANP to ensure that "old" applications and systems can continue to be operated over the entire period of time during which the TELEPERM XS platform exists. New developments in hardware or software must remain compatible with former equipment.

> Integrate **future versions of hardware components** into the system platform (e.g. for manufacturing reasons)
- ensure spare parts supply
- realize benefits from platform innovation

> Maintain the **qualified status of TXS system platform** software and hardware

> Keep **compatibility** with any existing project-specific application software

> Provide **innovative solutions** to implement interfaces between TXS, peripheral equipment and the plant process (bus systems, distributed peripheral devices)

> Use **improved tool functionality** for engineering and maintenance of TELEPERM XS applications

FIG. 6. Objectives for further lifetime of TELEPERM XS platform.

One example of how further developments are handled is the successor for the CPU module SVE1. The properties are listed in Figure 7.

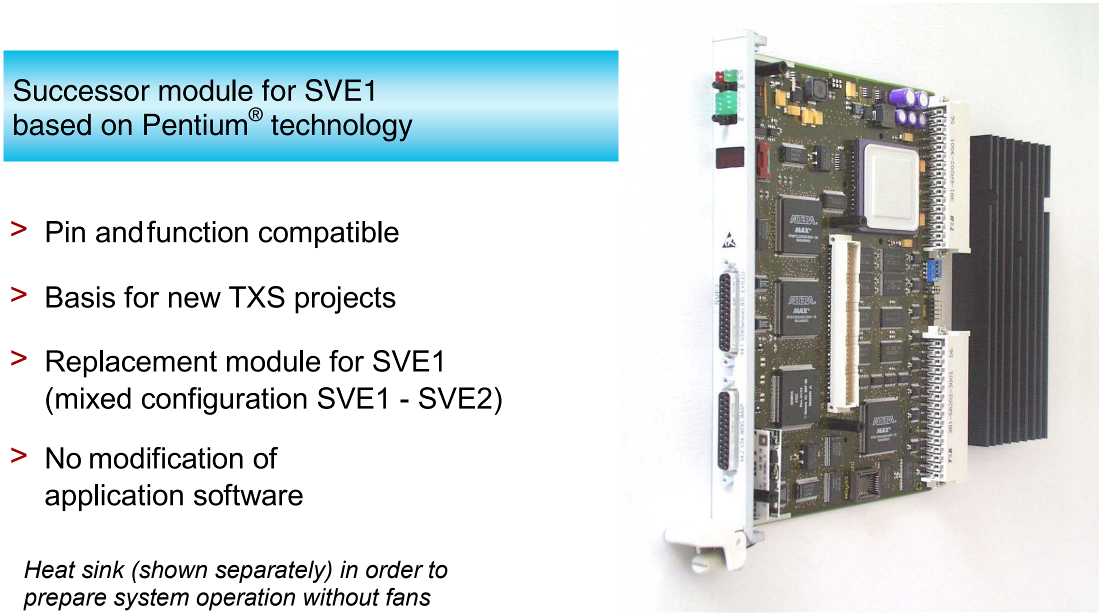

FIG. 7. The CPU module SVE2.

Paks Rt. decided to use the TELEPERM XS platform with its SPACE development software due to the supplier's long-term commitment towards the system, as described above, and the special features of TELEPERM XS for reactor protection systems.

SPECIAL FEATURES OF TELEPERM XS for reactor protection

The most important requirements to be met by the safety I&C equipment of a nuclear power plant concern the automation equipment for reactor protection. The necessary fault tolerance characteristics require a distributed multi-computer system. The automation devices used for the RPS permit the implementation and supervision of widely distributed topological structures such as that shown in Figure2.

The combination of these requirements with the requirement for unsafe plant states to be prevented (reliably thanks to the above-mentioned characteristic) by means of certain rapidly initiated protective actions results in very high performance requirements. TELEPERM XS fulfils these requirements and also has system characteristics that support the implementation and operation of highly redundant, spatially distributed architectures. For example, highly reliable and highly available I&C systems consisting of two, three or four redundant trains can be implemented with any required topology within the trains. Electrical isolation of the automation units is mainly achieved by using fiberoptics for communication. To avoid a reliability bottleneck in the interface between the redundant I&C trains and the individual control level, a highly reliable and highly available architecture has been applied for the associated voting processor, or relay voting is used for EP1 (SCRAM) actuations.

Robustness requires design measures that affect both the hardware and the software. The hardware is designed to meet the environmental conditions of KTA 3501 or the appropriate IEC requirements: as regards the temperature operating range, humidity and acceleration. Appropriate shielding measures for the cabinets ensure that the stringent requirements with respect to electromagnetic compatibility are met. In addition to design measures in the hardware, special design rules are also applied in the software. One important rule demands efficient decoupling of the plant process from the behavior of the I&C system because it must be guaranteed that a disturbance or accident in the plant process cannot under any circumstances have an impact on safety I&C. For this reason the TELEPERM XS computers process all tasks cyclically and do not use event-controlled programs. This means that measurement signals are read in, limit signals are formed and control commands are output in a never varying sequence regardless of what happens in the plant or in other computers.

For using the signals supplied from the TELEPERM XS system, it is connected to the plant process computer (VERONA) and the core analysis computer (SCADA) via unidirectional gateways.

QUALIFICATION OF TELEPERM XS PLATFORM

Qualification of the TELEPERM XS platform was divided into a generic part comprising type testing which was done once for the platform and after that only for additions or modifications, and a part based on qualification for a specific application.

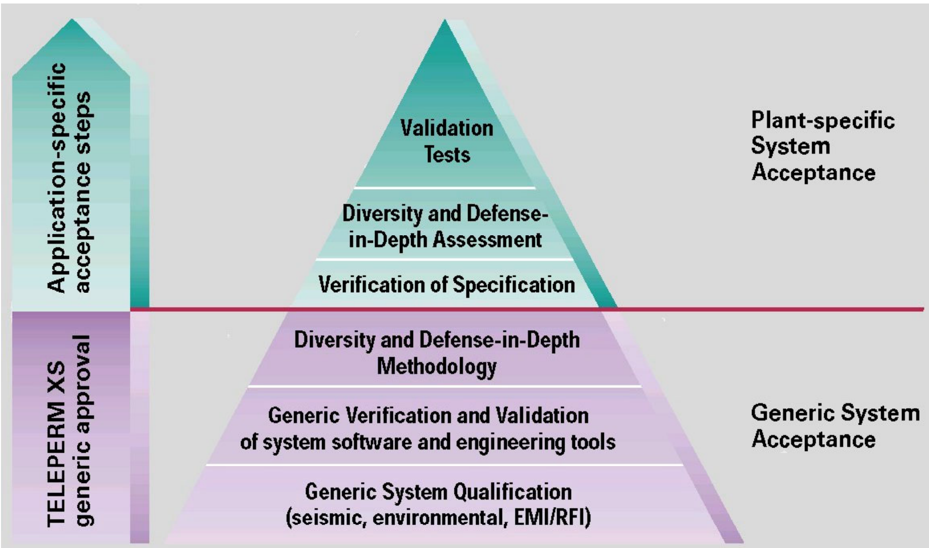

FIG. 9 Qualification of TELEPERM XS platform

Type testing was performed to demonstrate compliance of the modules with the data specified in the corresponding data sheet. The idea of type testing is to evaluate all safety-related features of a component as well as the manufacturing process independently of a specific application such that the component can later be used as a black box (for which the features evaluated in the type tests no longer have to be verified). It includes a theoretical and a practical part. Essential issues in the theoretical assessment are failure modes & effects analysis, critical load analysis and a program for practical tests which comprise functional tests and robustness tests (internal and external factors of influence and/or events). An authorized expert (independent institute) is responsible for reviewing the documents prepared by the manufacturer, evaluating the practical test program, participating in the practical tests (to the extent considered necessary) and assessing the test results.

One new aspect was that all re-usable software was also subjected to a type test analogous to the stipulations of KTA 3503. This method is of special advantage to the TELEPERM XS concept because the entire application software is made up of re-usable software modules combined by generators; i.e. software modules are used in the same way as hardware modules in hardwired systems. A software module has a well-defined functionality and a well-defined interface that can both be checked against the specification. While in the hardware type test greater emphasis is placed on practical testing, the software type test equally emphasizes theoretical and practical testing.

The **integration and system test** was introduced for the first time in the generic qualification phase as an additional step. The reason for this was that some important safety-related system features cannot be demonstrated at the module level.

The main safety-related system features verified by these tests were:

- Specified system performance; e.g. strict cyclical processing of the application code and proving the correctness of the processor and bus loads as well as of the response time for the specified application code calculated by the engineering tool SPACE
- Process for generating the application code
- System behavior independent of the application software and its allocated input data trajectories.

The generic qualification of the TELEPERM XS system was finished in 1997. Experience suggests that the qualification effort for software and hardware components is comparable if the software is developed according to the relevant standards and has a clear and simple architecture.

ENGINEERING

The development of plant-specific application software is generally considered the most error-prone activity during the implementation of safety-related I&C systems. For this reason, for TELEPERM XS, the production of application software has been strongly formalized. In this way problems related to staff members' individual working methods can be effectively avoided. Not only the production process but also the verification process can be automated to a remarkably high degree. In addition to the function described, the tools required for V&V were also integrated into SPACE, the engineering system of TELEPERM XS, so that its functional scope is far greater than that of any previously known engineering tool.

Paks Rt. provided the process requirement specifications in the form of "synoptical functional diagrams" and an I/O database. Framatome ANP used the SPACE editor of the engineering system SPACE to specify the I&C functions in the form of detailed functional diagrams. These functional diagrams with their well-defined presentation conventions can be read and understood by the parties who originated the task specifications for verification purposes and also serve as documentation for the customers.

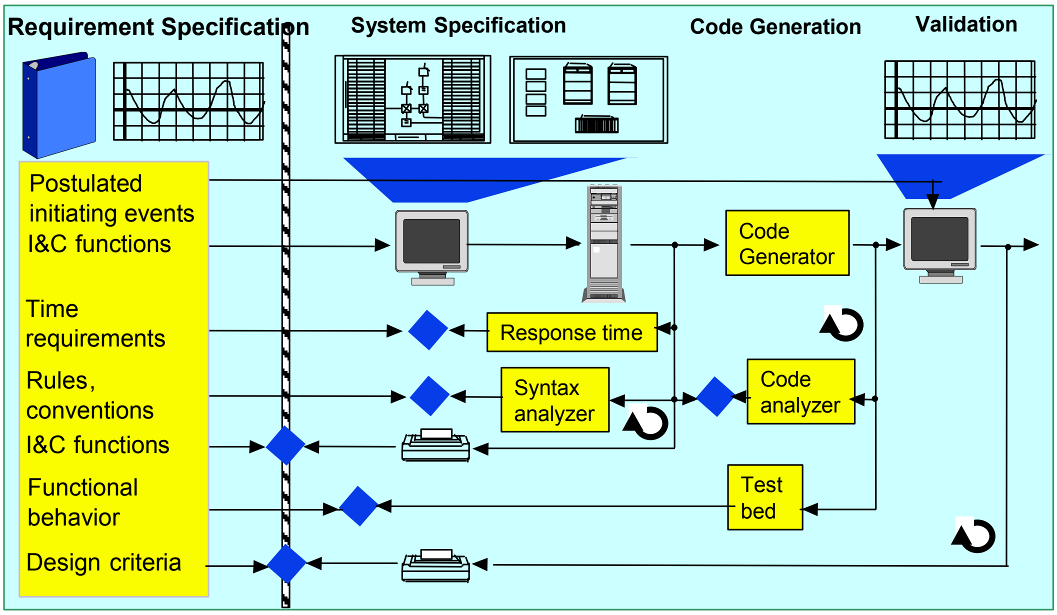

FIG. 8. System design process.

Once the specification is complete, it can be checked automatically according to various criteria using analysis tools. This analysis includes such important characteristics as completeness and the absence of ambiguity. This has only been made possible by the use of rigorously formal methods. The formally correct specification is then passed on for verification by the originators of the requirement specification who are also responsible for its release. For the quality of the application software it is of decisive importance that the application software be automatically derived from the formal specification by qualified generators. Because the software was generated by an automatic tool, it is possible to verify the consistency of the generated code with the formal specification using a second automatic tool. One consequence of this method is that quality verification of the code generator can be kept economically viable.

After this verification step, the code is ready for testing.

SPECIFIC DESIGN OPTIONS

The digital I&C platform offers special design options for handling signal inputs and for defining the behavior of the system in the event of false or missing input signals, thus helping improve system availability without reducing reliability or safety.

TRIPLE INPUT FAILURE HANDLING

In the TELEPERM XS system each analog and binary signal can be marked with an ERROR status, depending on signal values or failure events in the system. The ERROR status of a signal influences further processing of the signal in the TELEPERM XS function blocks (FBs). The ERROR status of a signal is processed according to simple rules:

1. If a function block is provided for majority voting (2-out-of-3, 2nd MAX, 2nd MIN), the ERROR status of the FB output signal is "NO ERROR" as long as at least one FB input

signal has the status "NO ERROR". Only if all FB input signals have the status "ERROR" does the output signal also have the status "ERROR".
2. All function blocks with no majority voting function (all others) give an "ERROR" status to the FB output signal if one of the input signals has an "ERROR" status.
3. If a signal is output via analog and binary output modules (S470 for analog, S451 for binary signals), the ERROR status of the signal results in 0V at the hardware channel output.

The ERROR status of signals can be set either, in the case of bus failures, by the error detection features of the TELEPERM XS system or, at the functional diagram level, by specially designed functions that detect special failure situations.
Bus failures are considered as single failures that are covered by the system design.

MONITORING OF ANALOG SIGNALS

Analog signals are acquired by the TELEPERM XS reactor protection system as 4 to 20 mA signals. The first software function block after signal acquisition (A-MBU) provides two monitoring setpoints (upper limit and lower limit) used to monitor whether the value of the analog signal is within the specified range or not. In the Paks RPS both setpoints are normally used to set the ERROR status for the initiation signal. The signal is then sent to the monitoring subsystems of the RPS and is used for the safety logic in all three trains. There a 2nd MAX or 2nd MIN function block is applied.

MONITORING OF BINARY SIGNALS

Binary signals are acquired by the TELEPERM XS reactor protection system as changeover (Morse) contacts for each binary signal that is specified in the I/O database. The two signals are checked for non-coincidence. If both signals have the same value due to electrical failures in the signal circuits, the ERROR status is set for the initiation signal concerned. The signal is then sent to the monitoring subsystems of the RPS and is used for the safety logic in all three trains. There a ≥ 2-out-of-3 function block is applied.

MEASURING RANGE VIOLATION

In the course of operation of a VVER-440 power plant (startup, power operation, shutdown, refueling and other outages) situations can arise in which especially analog signals may have values that exceed the specified measuring ranges. In this case the measuring range monitoring setpoints (as described above) are tripped, causing an ERROR status to be assigned to each of the signals involved. However, as all three redundant input signals will be affected, this causes an ERROR status after the majority voting FBs in all trains. This, in turn, will give an ERROR status to all RPS output signals. If this is not advisable in view of the current condition of the plant, then measures have to be taken to avoid such situations.
Hence investigations were performed to determine for which input signals the measuring ranges can be exceeded in special plant situations or, in other words, which output value should be defined in these special cases of triple input failures. This was specified by the operator, Paks Rt., on the basis of an accident analysis that considered the location of the sensors and the environmental impacts to which they are subjected.

The response by the RPS has been designed such that plant safety is not jeopardized. In addition, all responses by the RPS system can be identified through signaling of these actions to the operator via the Safety Monitoring System (SMS).

In summary, Framatome ANP sees the advantages of TELEPERM XS reactor protection systems as follows, although Paks Rt. was not able to realize these advantages in each and every instance due to plant-specific requirements or restrictions imposed by the authorities:

- No need for periodic testing of installed application and system software
- Reduced frequency for accuracy testing of analog input modules
- Use of Framatome ANP's pre-approved Diversity and Defense-in-Depth methodology.
- Pre-approved Equipment Qualification (e.g. seismic, environmental, electromagnetic interference and radiofrequency interference)
- Reduced scope of hardware modules.

INSTRUMENTATION AND CONTROL AGEING MANGAGEMENT: A FOCUS ON ELECTRONIC PARTS AND BOARDS

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Abstract

Aging management is a growing concern for nuclear operators. Instrumentation and control circuit boards require cost effective preventive maintenance. This paper presents field data on the aging of circuit boards in I&C systems. Some predictive techniques that can be used to monitor this aging are then exposed.

1. INTRODUCTION

Many nuclear plants have been operating for over twenty years. In the context of life extension, aging management is becoming a growing concern for the nuclear power industry.

An EPRI report on nuclear power plant common aging terminology [1] defines aging management as "engineering, operations and maintenance actions to control within acceptable limits aging degradation and wearout of System, Structure and Components (SSCs)." It explains that the term "is not limited to specific types of actions such as maintenance actions, but instead includes any activities which help assure that an SSC is able to perform its intended function as it ages. The definition recognizes that a certain degree of aging degradation and failures may be expected and acceptable, so long as they are controlled or managed. It is important to note that aging management includes the control of both normal aging degradation and error-induced aging degradation."

Even though aging management usually first refers to mechanical components like steam generators or reactor pressure vessels, or electro-mechanical parts like cables, nuclear plant operators are also facing increasing aging issues with original equipment installed for instrumentation, control, and safety system applications. Failures of instrumentation and control (I&C) systems due to aging may have an immediate impact on plant reliability and availability, but aging also affects long term plant safety and performance. Aging management of I&C systems therefore interacts with major issues of nuclear plant management, e.g., the optimization of maintenance programs or the life extension.

One specific area that needs attention is the aging of electronic boards and components used in I&C systems in nuclear power plants. IAEA published in 2000 a technical report on the management of aging of I&C equipment in nuclear power plant [2]. This report provides a list of I&C equipment vulnerable to aging degradation. According to this report, research work was still needed in 2000 for resistors, diodes, transistors, integrated circuits, etc.

In this context, EPRI launched in 2001 a project to better understand, characterize, predict, manage or mitigate effects of electronic board and component aging. The goals of this project are 1) to better understand the degradation mechanisms of I&C electronic systems, 2) to develop new methods to monitor prevalent mechanisms and forecast generic aging cases, and 3) to give guidelines for the aging management of I&C electronic systems. The ultimate objectives are 1) to reduce the likelihood of losing plant availability due to a board or component failure and 2) to reduce the global cost of maintenance by optimizing preventive maintenance tasks and the schedule for refurbishment or renewal.

2. DEGRADATIONS OF ELECTRONIC BOARDS IN NUCLEAR PLANTS

Aging mechanisms of electronic parts depend on technologies, manufacturing quality, and conditions of environment and operation (see reference [3]). A distinction has to be made between normal aging degradation and error-induced aging degradation. Aging of electronic parts is well documented in harsh environments (e.g., aerospace, military, or automotive industries data). However, only a small amount of data on electronic part failures in nuclear power plants is available because the conditions are benign and the failure rates are supposed to be low. It is thus difficult to identify what will predominantly cause the aging of electronic systems after twenty or thirty years of operation. Usual causes are temperature, corrosion, temperature cycles, or extrinsic aggressions (e.g., electrical over stress (EOS), electro static discharges (ESD)).

In the first phase of the project, members of the EPRI I&C steering committee were requested to fill out a table and indicate their interest levels (from 0 = None to 6 = Highest) for each family of electronic parts (relays, integrated circuits, capacitors, resistors, etc.). Two criteria were used:

a) The probability of wearout, given the field data and what is known on the technologies
b) The damage which would be caused by a generic wearout, given the number of systems or boards which would be affected and their functions

After the survey, different types of information were collected from several utilities, stemming from:

a) Failure analysis reports

   As explained in Amerasekera and Najm's book on failure mechanisms in semiconductor devices [4], failure analysis refers to "the task performed subsequent to a failure, in order to identify the failure mode, failure mechanism, and root cause." A failure analysis is generally conducted when a failure may be generic to many parts or boards. After the analysis, corrective actions may have to be undertaken (e.g., preventive replacement of parts, specific inspections or measurements, etc.).

b) Physical analysis reports

   These analyses are conducted on functioning parts which may have aged. The analyses are performed to find the effects of aging on used parts. They may be used in preventive maintenance programs.

c) Visual inspection data

   Visual inspections are achieved on site, to visually check the physical status of the boards. The main observations are related to solder joints, corrosion, the pollution[1] of the boards, and mechanical shock effects on the boards or part packages. These inspections are generally a part of preventive maintenance programs.

d) Procedures related to the aging management of electronic boards

   Utilities use some specific procedures related to electronic part and board aging. We have collected some of them.

Table 1 presents the main results of this collection of field data.

On the one hand, we found that aluminum electrolytic capacitors, relays, potentiometers, and optocouplers were identified as the most "normal aging" sensitive devices. On the other hand, error-induced degradations are frequent for relays, edge board connectors, pin through hole (PTH) solder joints, switches and printed circuits, but their severities are high for capacitors (all types) and integrated circuits.

Figure 1 and figure 2 show examples of degradations on a capacitor and an integrated circuit found in operating I&C system cabinets.

[1] Pollution is defined as the presence of chemical contaminants stemming from the air, from human activities (e.g., grease or sweat on contact mates) or from the degradation of materials on the board itself. Pollution may affect the board reliability and provoke corrosion.

Table I. Main normal and error-induced stressors from field data

Family	Main normal stressors	Severity of normal degradations[a]	Main error-induced stressors	Frequency of error-induced degradations	Severity of error-induced degradations
Capacitors, Al electrolytic	Temperature & current	**High**	Manufacturing defect, misapplication	Seldom	**High**
Capacitors, others	Temperature & current	Low	Manufacturing defect	Seldom	**High**
Relays	Mechanical stress	**High**	Pollution, corrosion	**Frequent**	Medium
Potentiometers	Mechanical stress	**High**	Manufacturing defect	Seldom	Medium
Edge board connectors	Mechanical stress	Medium	Bad handling (mechanical) & pollution, corrosion	**Frequent**	Medium
Power diodes and transistors	Temperature	Medium	Manufacturing defect	Normal	Medium
Transformers and inductive devices	Temperature	Low	Manufacturing defect	Seldom	Medium
Thyristors	Thermomechanical stress	Medium	Manufacturing defect	Normal	Medium
Integrated circuits	Diverse, usually temperature driven	Low	Electrical overstress	Normal	**High**
Printed circuits	Temperature	Low	Bad handling	**Frequent**	Low
Signal diodes and transistors	Temperature	Low	Manufacturing defect, usually attachment	Normal	Low
Regulators and other analog circuits	Temperature	Medium	Manufacturing defect	Seldom	Medium
Optocouplers	Thermomechanical stress & current	**High**	Manufacturing defect	Seldom	Low
On board connectors	Mechanical stress	Low	Bad handling	Seldom	Medium
Fixed resistors	Temperature and current	Low	Bad rating	Normal	Medium
Light Emitting Diodes	Current	Low	Manufacturing defect	Seldom	Low
PTH solder joints	Thermomechanical stress	Low	Manufacturing defect	**Frequent**	Low
DC/DC converters	Thermomechanical stress	Medium	Bad rating & manufacturing defect	Seldom	Medium
Surface Mounted Technology solder joints	Thermomechanical stress	Low	Manufacturing defect	Seldom	Medium
Liquid Cristal Displays	No data	-	No data	-	-
Switches and keyboards	Mechanical stress	Medium	Handling (excessive mechanical stress)	**Frequent**	Medium
Quartz	Mechanical stress	Low	Handling	Seldom	Low

[a]After about 20 years of operation.

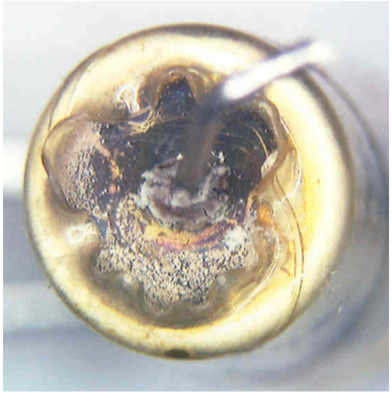

FIG. 1. Aluminum electrolytic capacitor degradation due to corrosion (from EDF)

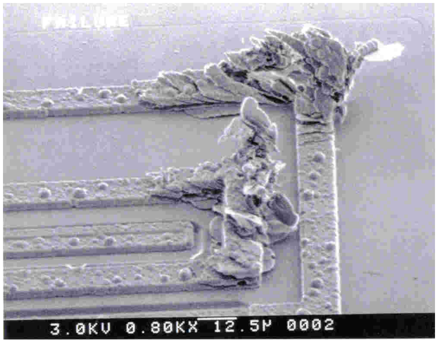

FIG. 2. Effect of electromigration on a poorly manufactured integrated circuit, after 20 years of operation (from San Onofre Nuclear Power Plant)

3. NEW METHODS FOR THE MONITORING OF ELECTRONIC PARTS

Aging management can partly be based on the monitoring of electronic system aging. The monitoring task aims at forecasting the aging, either normal or error-induced. It can rely upon two types of actions: aging tests on aged parts, at a given time, to assess their residual lives or continuous monitoring of key "end-of-life indicators".

Work is still under progress, but it seems that continuous monitoring may be effective:

a) For capacitors and relays, using impedance versus frequency measurements,
b) For thyristors, using threshold currents measurements,
c) For optocouplers, using current transfer ratios (CTR) measurements.

As an example of what can be done, figure 3 presents a versatile tool which was developed in the frame of this work to measure CTR of optocouplers. This tool can be used on site to preselect optocouplers that should be preventively replaced.

Aging tests on aged parts could apply for integrated circuits in general, and for UVEPROM memories more specifically for which the issue of data retention has to be solved[2].

FIG. 3. Portable tool for on site measurement of optocouplers Current Transfer Ratio (CTR)

4. CONCLUSIONS

Even if the physical environment of I&C electronic systems is benign, some actions must be undertaken to control their aging mechanisms. Between the two extreme strategies of “doing nothing” or “preventively replacing everything”, the sensible approach of monitoring specific parts seems to be a good compromise. It must be focused on “aging sensitive” parts, and can consist of both aging tests and continuous monitoring of end-of-life indicators. By monitoring the aging, preventive maintenance should be optimized.

REFERENCES

[1] ELECTRICAL POWER RESEARCH INSTITUTE, Nuclear Power Plant Common Aging Terminology, EPRI TR-100844, November 1992.

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Management of Ageing of I&C Equipment in Nuclear Power Plants, IAEA-TECDOC-1147, June 2000.

[3] ELECTRICAL POWER RESEARCH INSTITUTE, A Review of Equipment Aging, Theory and Technology, EPRI NP-1558, September 1980.

[4] E. AJITH AMERASEKERA, FARID N. NAJM, Failure Mechanisms in Semiconductor Devices, John Wiley & Sons, Second Edition, 1997.

[2] The ability of programmed UVEPROM cells to retain their charges may be critical to long term reliability. The electrons trapped in the floating gate structure can be removed by leakage through the oxides surrounding the polysilicon or by exposure to ultraviolet light.

Session 4

TECHNOLOGICAL AND OPERATIONAL ASPECTS OF PLIM

Sub-Session 4.5

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VESSEL HEAD PENETRATIONS: FRENCH APPROACH FOR MAINTENANCE IN THE PLIM PROGRAM

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Abstract

Base nickel alloys like Inconel 600 or 182 are particularly sensitive to stress corrosion cracking. This fact is well known since Corriou'works at the beginning of the sixties and its applications to the steam generator tubes in the seventies. For the RP vessel heads, the major fact of the nineties was the leak that occurred on one penetration in 1991 in the French NPP unit of Bugey. Several important decisions were taken after discover of this leak. First of them was to understand why it appeared so quickly, then test repairs for the Bugey case, then decide to replace all vessel heads considering that the repair solutions was to high cost. In parallel many developments were launched to establish laws for PWSCC and develop non-destructive methods to inspect the head penetrations. The conclusions obtained show the decision was good and no new leak happened on the VH penetrations.

INTRODUCTION

On September 1991, an important event occurred in France at the Bugey unit 3 NPP; a leak in the vessel head was discovered during the hydro test at 207 bar. It appeared that a control rod drive mechanism (CRDM) penetration had failed. Few investigations showed that the leak was produced by longitudinal cracks coming from inside the penetrations. After a destructive examination, the damage mechanism, which produced the cracks, was clearly identified and attributed to primary water stress corrosion cracking (PWSCC). In fact, for what concerns alloy 600, this problem was not a new one. Since few years, lots of cracks had been discovered in the steam generators tubes mad with the same material and PWSCC in such material was known. Nevertheless, this cracking was not expected so early in the CRDM adapters as all studies showed an initiation after more than 10^5 hours operating.

Effects of stress, material quality, welding, and manufacturing process had been underestimated. Then Electricité de France (EDF) decided to launch an important program for better understanding the phenomena. In parallel, the nuclear division decided to replace all vessel heads with alloy 600 by new vessel head with alloy 690. The two aspects of the program are developed hereafter.

PHENOMENON

900 MW VHs are equipped with 65 CRDM penetrations and 1300 MW VHs with 77 or 78. Just after discovering the leak, many works were engaged to understand what had happened. First of them were based upon the expertise of the Bugey 4 T65 vessel head penetration and after this from the Bugey 3 T54 where the leak had been discovered. Morphology of the crack was fully analyzed and first explanations could be given [1]. Clearly, the leak was attributed to a PWSCC. After VH removal and dismantling the CRDM, several examinations were performed. Firstly, few non destructive examinations, eddy current testing, visual testing, leak tests, etc …, revealed longitudinal cracks in the lower part of the penetration near the weld.

Secondly, after a destructive examination of the T54 VH penetration, the crack was initiated from inside and had a through wall extent that explained the leak as shown on figure 1.

Consequently, at this time in 1992-1993, the analysis of the phenomenon concluded to effects of:

- temperature and the water chemistry,
- stress level particularly for the peripheral penetrations due to the angle between the penetration and the VH and the fact the weld is not symmetrical,
- material (sensitivity).

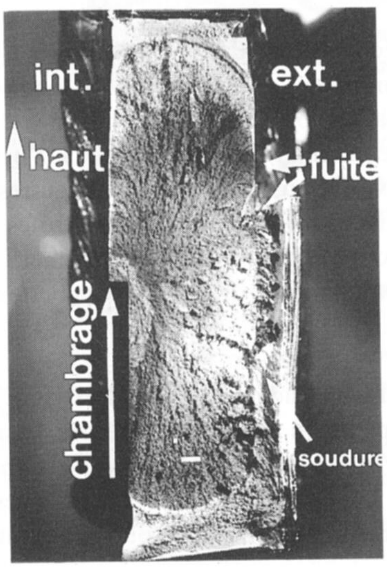

FIG. 1. TWE crack in T54 VH penetration.

Then it was decided to orientate research and developments in three main directions:

- develop methods to repair the VH penetrations and study the possibility to replace vessel heads,
- develop non-destructive methods for detection and characterization of the surface breaking cracks in order to know as soon as possible the crack initiation and measure the crack propagation,
- launch an important program of laboratory research to better understand PWSCC of both alloy 600 for base metal and alloy 182 for weld metal taking into account other areas of concern are studied like bottom mounted instrumentation (BMI) penetrations and core support blocks in the pressure vessel, Inconel welds of the division plate in the steam generators and all other areas where alloy 600 and 182 have been used during manufacturing.

INDUSTRIAL DEVELOPMENTS

Repairs

Proposals for repairs were made immediately but due to the shape of the VH and the number of penetrations, the high dosimetry, the difficulties were important and only few tests and repairs were performed during more than one year. In fact this solution was abandoned for the benefit of a replacement solution. For EDF it was obvious that with the first results of the non-destructive inspections all VH would be concerned and then to be repaired or changed.

Non-Destructive Methods

Few NDE methods were studied but the effectiveness of some of them was proved after many tests. For crack initiation, visual testing, dye penetrant testing and eddy current testing were studied. Due to the thermal sleeve inside the penetration, all three NDE methods and particularly dye penetrant testing were very difficult to implement. They needed to move it during the inspection or remove it. The solution came with the development of "eddy current blade probes" that could be used with the sleeve. It was easy to perform the examination from inside inserting this probe between the sleeve and the penetration. Many tests were performed in EDF and CETIC facilities [1].

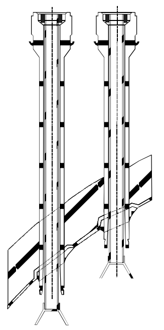

FIG. 2. Blade probe used for inspection.

[1] CETIC is a common EDF and FRAMATOME facility located in Châlon sur Saone in France.

For crack propagation the challenge was also very difficult as it was not an industrial way to use visual techniques because of the long duration of the examination. Then special "ultrasonic sword probes" were developed (maximum thickness 2 mm) and qualified using the "Time Of Flight Diffraction Technique" (TOFDT) that revealed to be the best for such flaw to size. The performance obtained showed a 3 mm surface-breaking crack was sized with a good accuracy on mock-ups with artificial flaws, realistic and real cracks. All results of the qualification were discussed at length [2], [3].

Laboratory Studies on Alloy 600 / Alloy 182

From the experience gained with the alloy 600 steam generator tubes, few resultants about PWSCC were acquired. The sensitivity of the material (for instance carbon content) was known and many expertises of pulled tubes helped for understanding this type of cracking. Nevertheless, the situation of VH penetration was different by the fact the level of stress can be very important due to welding process. Particularly peripheral penetrations are subject to constrains because of the dissymmetrical welds that introduce high level of stress. This is confirmed by the NDE results (the peripheral penetrations are more cracked).

Many studies were performed and the main results can be summarized as proposed.

Alloy 600 (Base Metal) Crack Initiation and Propagation

A model has been developed in EDF and many comparisons with the results obtained on SG tubes were done [4], [5]. It has to be underlined that for alloy 600, the most important conclusions are summarized as follow:

- there are three stages of cracking : incubation, slow propagation, fast propagation. The initiation is defined as that necessary to reach the fast propagation.
- crack velocity increases significantly in a range of temperature between 290°C and 360°C. Crack growth rate is a energy activated phenomenon and the mean value was evaluated at 130 kJ/mol.

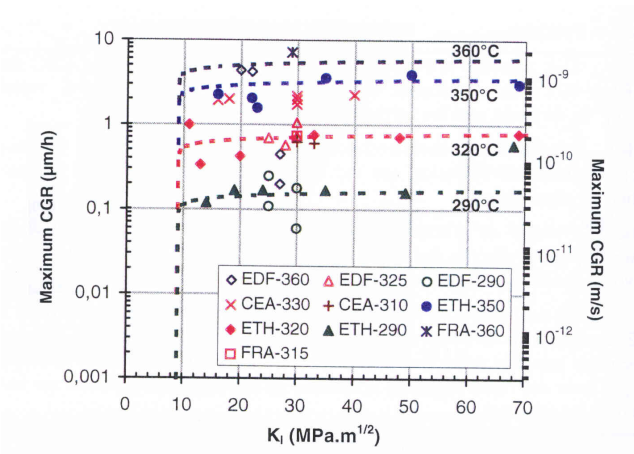

FIG. 3. CGR versus K for alloy 600.

The crack propagation rate is modeled by a law : $da/dt = \alpha \,.\, (K\text{-}9)^{0,1}$ with a plateau (see figure 3).

- the influence of the surface condition is important for the VH penetrations but seems to have less influence in the other Inconel zones if stresses remain moderate.
- 10% cold working increases the crack growth rate by a factor of 2.
- stress relief treatment decrease the crack growth rate by a factor of 2 compared with as welded conditions.
- microstructure can have a great influence on SCC. A high grain boundary coverage by the carbide precipitation generally correlates to good resistance to SCC.

Finally a crack growth rate law has been established for alloy 600 and can expressed as :

$$CGR_{600}\,(\mu m / s) = 17_{x}10^{9}\ _{x}\ (K\text{-}9)^{0,1}\ _{x}\ \exp[\,-Q/8{,}32\ _{x}\ (1/T)]\ _{x}\ \exp[a\ _{x}\ Re]\ _{x}\ \alpha$$

- Q: mean activation energy,
- T: temperature
- a: coefficient
- Re: yield strength
- α: depending of cold working.

Alloy 182 (Weld Metal) Crack Initiation and Propagation

A same study has been performed on alloys 182 and 82 used to weld CRDM penetrations but also BMI penetrations and core support blocks.

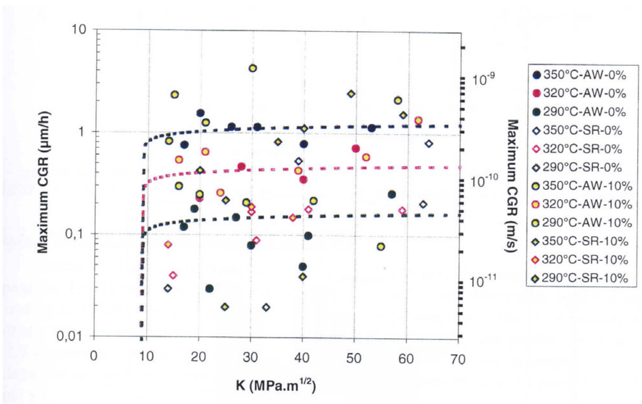

FIG. 4. CGR versus K for alloy 182.

The effects of the main parameters on the kinetics have also been studied and the most important conclusions are :

- crack growth rate is a energy activated phenomenon and the mean value was evaluated at 130 kJ/mol.
- stress intensity factor can be modeled by a law $(K\text{-}9)^{0,1}$ (see figure 4).
- 10% cold working increases the crack growth rate by a factor of 2.

- propagation in a direction parallel to the dendrites in weld metal is faster than that perpendicular to them (estimated factor between 2 and 5).
- cyclic loading (Westinghouse type) increase the propagation by a factor of 2.

As for alloy 600, a crack growth rate for alloy 182 was proposed as well :

$$CGR_{182}\ (\mu m / s) = 175_x 10^{9}\ _x\ (K\text{-}9)^{0,1}\ _x \exp[-Q/8{,}32\ _x\ (1/T)]\ _x\ \alpha\ _x\ \beta\ _x\ \delta\ _x\ \chi$$

- Q: mean activation energy,
- T: temperature
- α: depending of cold working
- β depending of stress relief heat treatment
- δ depending of the dendrites orientation
- χ depending of cyclic loading

It has to be underlined that the comparison of the laboratory results between alloy 600 and alloy 182 show similarities. For alloy 82, laboratory studies and results are poor because of a better resistance to SCC due to the higher Cr content (20%) compared with alloy 182.

EXPERIENCE FEEDBACK FROM THE FIELD

As mentioned previously, many non destructive examinations were performed on all 54 French vessel heads with alloy 600 CRDM penetrations since 1994-1995 when ET and UT inspection procedures were finalized and stabilized. From this important set of data, few conclusions are now established and show clearly evidences.

INSPECTION PROGRAM

The inspection program is based upon the maximum CGR measured on the penetrations of one VH and considered as the most sensitive. As example for 900 MW VH penetrations and 12 months cycles; the inspection policy is summarized as follow:

VH penetrations without crack	Interval between 2 inspections is 3 years.
VH penetrations with a crack : 3 mm < SBC< 5 mm	Interval between 2 inspections is 2 years
VH penetrations with a SBC > 5 mm	Interval between 2 inspections is 1 year.

SBC : Surface Breaking Crack

In all cases, VH is replaced as soon as the safety criterion is reached (4 mm remaining ligament).

ALLOY 600 - VH PENETRATIONS BASE METAL

Two important conclusions have to be mentioned using the experience feedback from the NDE field :

- firstly, it is evident that the level of stress has a major role for the initiation and propagation cracking. The main fact concerns the number of cracked VH penetrations and the circle where it belongs. Figure 5 shows that peripheral penetrations are mostly cracked compared with central ones.
- For this consequence, French regulators agreed in 2000 not to continue the examination of the central VH penetrations (vent hole and 4 circles for 900 MW and 5 circles for 1300 MW).
- secondly, the effect of the heat sensitivity is particularly obvious for those that are classified sensitive. All VH penetrations were ranked in four categories from the less sensitive A to the most one D taking into account the material properties of each penetration. Figure 6 shows the effect of this sensitivity. Type B and C heat number are more often cracked than type A heat number but VH penetrations with A heat are more numerous. So in percentage, the value is undervalued. The effect also exists for type D but is less obvious due to the small amount of type D VH penetrations.

ALLOY 182 - WELD METAL

Another risk is the weld metal cracking concern. In fact the Bugey 3 leak did not concern weld metal as the crack developed fully in the base metal. Nevertheless taking into account international informations and after the discover of a defect in the J groove weld on Ringhals VH, EDF decided to perform dye penetrant examinations of the J weld on several removed VH.
754 welds were inspected and none of them were found cracked.

VH REPLACEMENT

At he end of 2001, 41 VH had been replaced in France as shown on figure 7. The percentage of cracked penetrations remained low for all types of RPVs as the detection of crack initiation is done early.

It has to be underlined that the maximum value of the crack kinetics (~ 3 mm / year) appeared only one time. All the other cases are around 2 mm / year or less now (~ 1 mm/year).

CONCLUSIONS

In 1991, after discover of a leak on a vessel head penetration of Bugey NPP unit 3, numerous studies to understand the phenomenon were implemented. Several of them were in the field of expertise, others in the field of non destructive examinations in order to have all necessary tools to perform inspections The most important works were laboratory studies on both alloys 600 and 182. One of the major results concerned crack growth rates laws that can predict crack propagation.

In parallel, the maintenance policy was based upon the following key points:

- very early, replacement of all VH with alloy 600; up to now 41 were replaced.
- importance of the NDE inspections to measure real CGR and determine the best instant for replacement.

- importance of 2 parameters, penetration angle in the VH and sensitivity of the heat number.
- In conclusion, in the frame of Plant Life Management, decision to replace all vessel heads with alloy 600 base metal penetrations was a good decision particularly for future and life extension.

REFERENCES

[1] J. ECONOMOU et al, Contrôle et expertises métallurgiques de traversées de couvercle de cuve International symposium FONTEVRAUD III - September 12-16, 1994

[2] J. SAMMAN – J. SALIN, Keeping vigilance for RPV crack : improved NDT tools for SCC in bottom head penetrations, Nuclear Engineering Vol 41 n° 505 – August 1996

[3] J. SALIN, NDE advances due to 5 years of inconel zones, EPRI workshop – Daytona 1997

[4] C. AMZALLAG et al, 9th Environmental Degradation Conference, Newport - September 1999

[5] S. LE HONG et al, Modelling of the stress corrosion initiation time on Alloy 600 in primary water of PWR's, TMS - August 1999

[6] S. LE HONG et al, ICG-EAC 2000 - April 2000

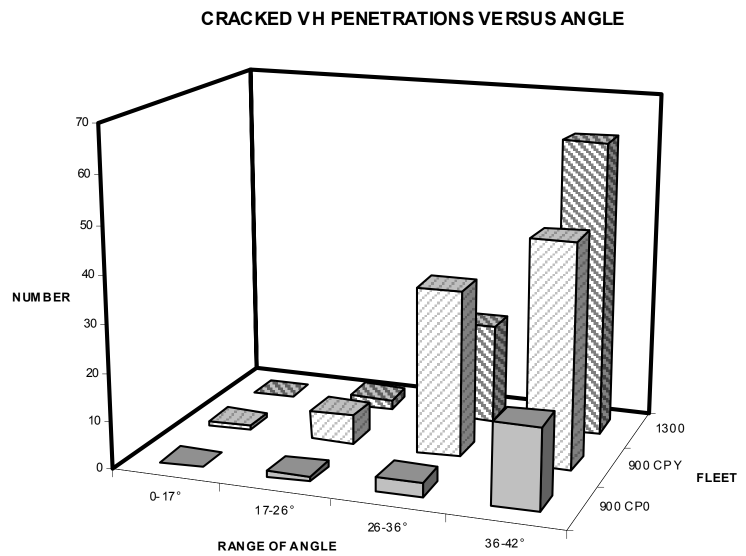

FIG. 5. Importance of the "angle effect" versus VH penetrations cracking.

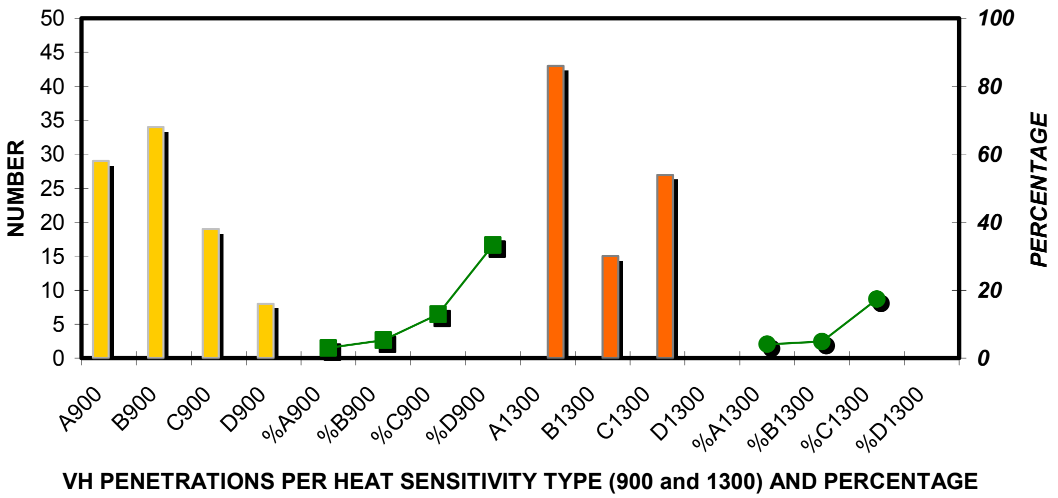

FIG. 6. Importance of the heat sensitivity versus VH penetrations cracking (percentage values are given versus the total number of penetrations in standardized plant series 900 or 1300).

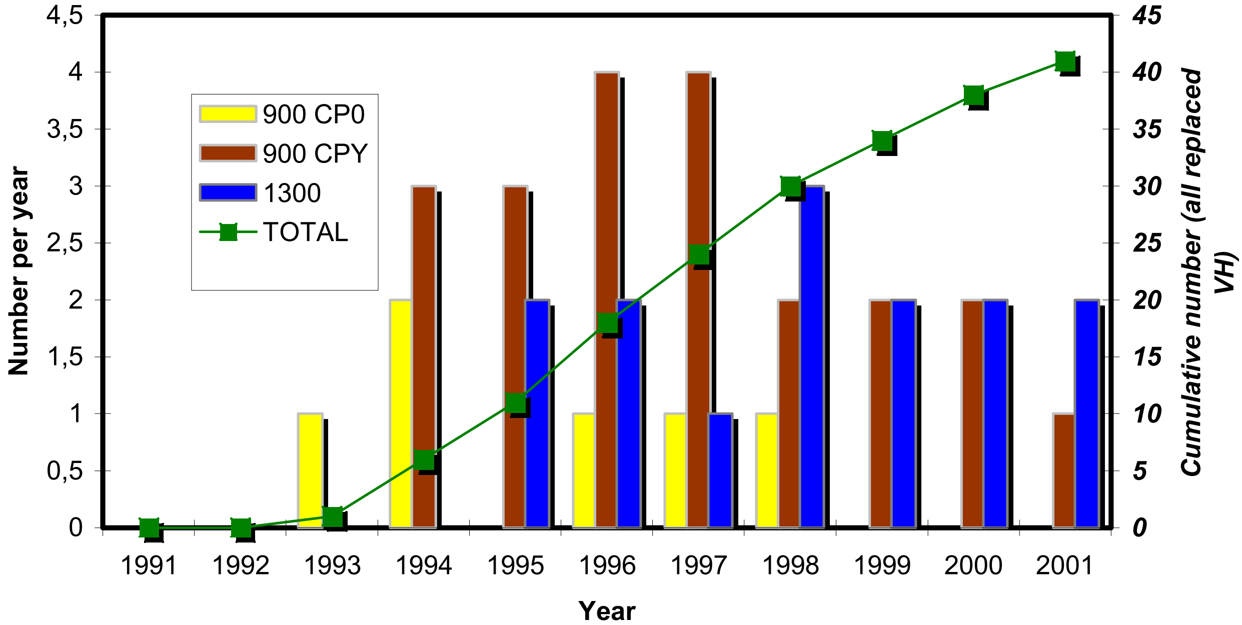

FIG. 7. VH replacements versus the 900 and 1300 MW fleet during last ten years.

DEVELOPMENT OF INSPECTION AND EVALUATION GUIDELINES FOR LIGHT WATER REACTOR INTERNALS

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Abstract

Should an aging degradation such as SCC be found at the reactor core internals, one of the most important and critical components in the study on Plant Life Management, a great amount of labor and time may be required for root cause investigation and analysis and subsequent repairs because it is very difficult to reach to the degraded portion due to structural, dimensional and environmental restrictions. Therefore, such situations may provide serious impact on plant operation and management. So, it is strongly recommended that a contingency plan for the reactor core internals should be prepared in advance. Thus, considering the significance of aging degradation, it is necessary to develop some standard rules for the inspection and evaluation of reactor core internals. Based on such understanding, the Thermal and Nuclear Power Engineering Society (TENPES) has organized a special committee to develop the guidelines necessary for the inspection, evaluation and repair of PWR and BWR reactor internals. The outline of the guidelines developed by the TENPES committee is presented herein.

1. INTRODUCTION

In Japan, the history of light water reactors (LWRs) began with the commencement of commercial operation of Tsuruga Unit 1 (BWR2 / 357MWe) on March 14, 1970. Since then, the number of LWRs in commercial operation has increased to 52. As of the end of August 2002, the current total electric generation capacity is about 46 billion-watt and, next to the United States and France, is ranked the third largest in the world.

Since 1983, the average capacity factor of Japanese LWRs has been maintained over 70% and was 80.5% in 2001. We believe this was mainly due to extensive preventive maintenance and the reflection of lessons learned from the worldwide trouble information as well as domestic trouble information.

In the mid-1990s, some LWRs were getting near to 30 years of operation and public attention toward aging degradation increased considerably.

Throughout the past decade, it was occasionally reported that cracks identified as intergranular stress corrosion cracking (IGSCC) or fatigue had been found in various parts of reactor core internals both in Japan and overseas. In 1999, many cracks were found in the shroud support of Tsuruga Unit 1.

Under the circumstances described above, a special committee was established in the Thermal and Nuclear Power Engineering Society (TENPES) to develop inspection and evaluation guidelines with technical rationale for LWR core internals in Japan. These guidelines were developed by assuming initial cracks and their growth at all weldments and to provide methods to determine the timing of initial inspections and re-inspections and also a method to evaluate the structural strength and safety function of cracked reactor core internals.

This paper presents a concept and outline of the developed guidelines for the reactor core internals with the potential to extend significant impact on plant life management for LWRs.

2. OUTLINE OF PLM STUDY IN JAPAN

In Japan, a ruling was made in 1996 that Japanese utilities carry out a “study on plant life management (PLM)” for each LWR before it reaches 30 years of operation.[1], [2] The outline of this study is shown in Fig.1. This activity consists of two parts. The Part 1 study was performed by the Japanese nuclear regulatory authority (METI) to formulate a basic concept and policy to cope with the aging degradation of aged plants. In this study, technical evaluations for major critical components were conducted.

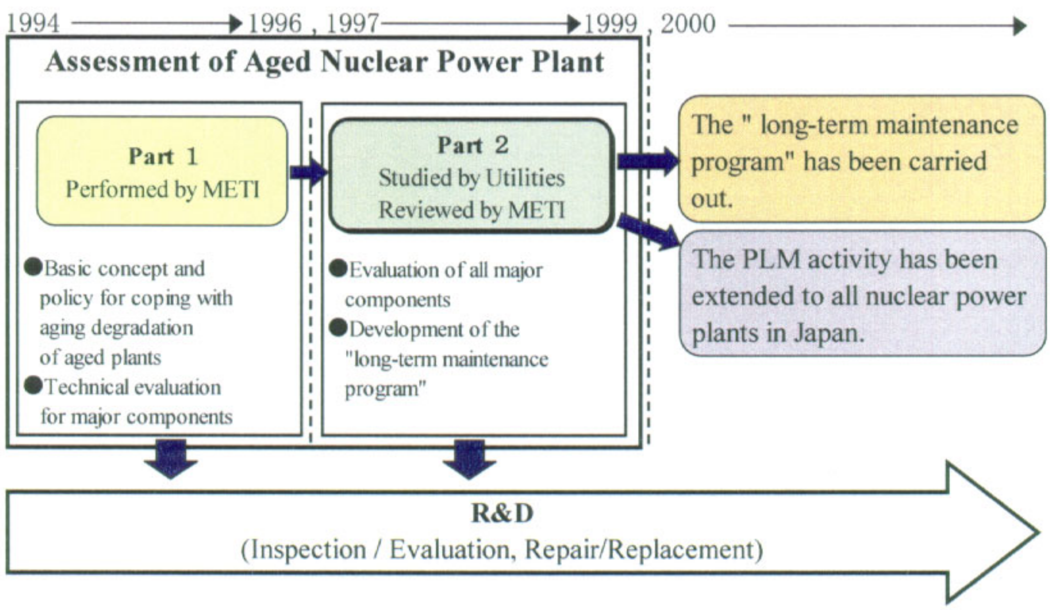

FIG.1. Activities on PLM study in Japan.

The Part 2 study was done by the Japanese utilities and reviewed by METI. Figure 2 shows the concept of the Part 2 study. In this study, from all systems, structures and components in an LWR plant, safety-related and power-supply-related components were selected and divided into 16 categories as follows : pumps, heat exchangers, motors, vessels, piping, valves, reactor core internals, cables, transmission equipment, turbines and auxiliaries, concrete structures, I&C, heating and ventilation systems, other important equipment (DG, CRD, etc), power supply equipment and other equipment. In the next step, aging degradation mechanisms that should be taken into account for evaluation were identified based on the latest knowledge from the worldwide O&M experiences and research etc. The utilities then selected a representative component from each category and evaluated its integrity from the viewpoint of identified aging degradations under the assumption of 60 years operation. Finally, based on the results of this evaluation, some additional maintenance tasks deemed necessary to cope with aging degradations were listed and prepared as the long-term maintenance program (LMP). These additional maintenance tasks were incorporated into the current maintenance program and should be carried out after 30 years of operation.

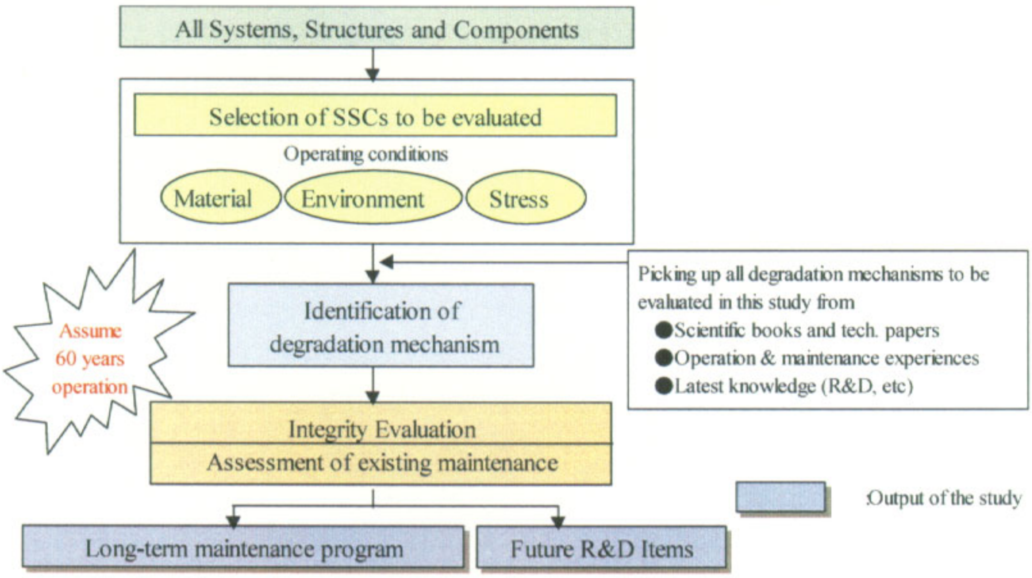

FIG.2. Evaluation steps in PLM study.

The study on PLM was incorporated into the current periodic safety review (PSR) activity, which is to be carried out every 10 years. In every PSR, the entire study will be re-evaluated and revised. (Fig.3)

3. NECESSITY OF PREPARING I&E GUIDELINES FOR REACTOR CORE INTERNALS

Should aging degradation such as IGSCC be found at the reactor core internals, one of the most important and critical components in the study on PLM, because it is very difficult to reach to the degradation portion under structural, dimensional and environmental restrictions, a great amount of labor and time may be required for root cause investigation and analysis and subsequent repairs. Therefore, degraded reactor core internals provide a great impact on plant operation and management. So, it is strongly recommended that a contingency plan for the reactor core internals should be prepared in advance. Thus, the activity to develop standard rules for the inspection and evaluation of reactor core internals was initiated. These rules to be developed should specify when, where and how to inspect. They should also specify the method to evaluate structural strength and safety function in the event degradation such as a crack is found. The repair methods, if required, should also be specified. To obtain public acceptance, the contents of these rules should be technically reasonable and clarify the technical reasons or rationale. This is essential not only for plant life management but also to fulfill the accountability for the standard rules.

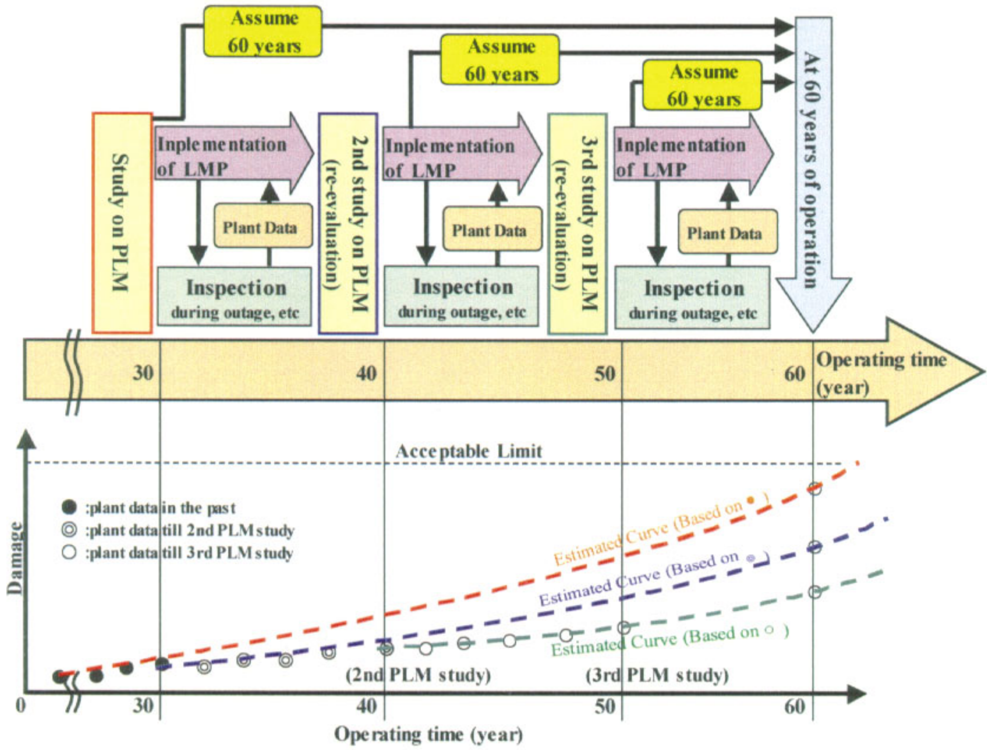

FIG.3. Concept of PLM study.

4. ORGANIZATION FOR DEVELOPING I&E GUIDELINES

Based on the understanding described above, the special Committee on inspection and evaluation guidelines for reactor core internals (Chairman: Dr. Asada, professor emeritus of the University of Tokyo) in the Thermal and Nuclear Power Engineering Society (TENPES) was organized in February 2000.

The purpose of this committee is to propose technically reasonable and rational I&E guidelines for LWR core internals. To perform comprehensive study with reference to the latest knowledge, the committee consists of scholars in key engineering fields and representatives from the utilities and plant vendors.

5. BASIC POLICIES FOR THE DEVELOPMENT OF I&E GUIDELINES

The basic policies of the committee for developing the guidelines are as follows.

1) Study on the major premise of ensuring reactor safety
2) Clarify the technical basis, reasons or rationale
3) Reflect the latest knowledge to the guidelines and revise them annually

6. DEVELOPMENT OF I&E GUIDELINES

Figure 4 shows the basic flow for the development of I&E guidelines. The basic steps for the development of I&E guidelines are as follows.

1) Select components
2) Identify degradation mechanisms for each selected component
3) Conduct structural analysis and evaluate safety function
4) Determine inspection methods
5) Conduct crack propagation analysis

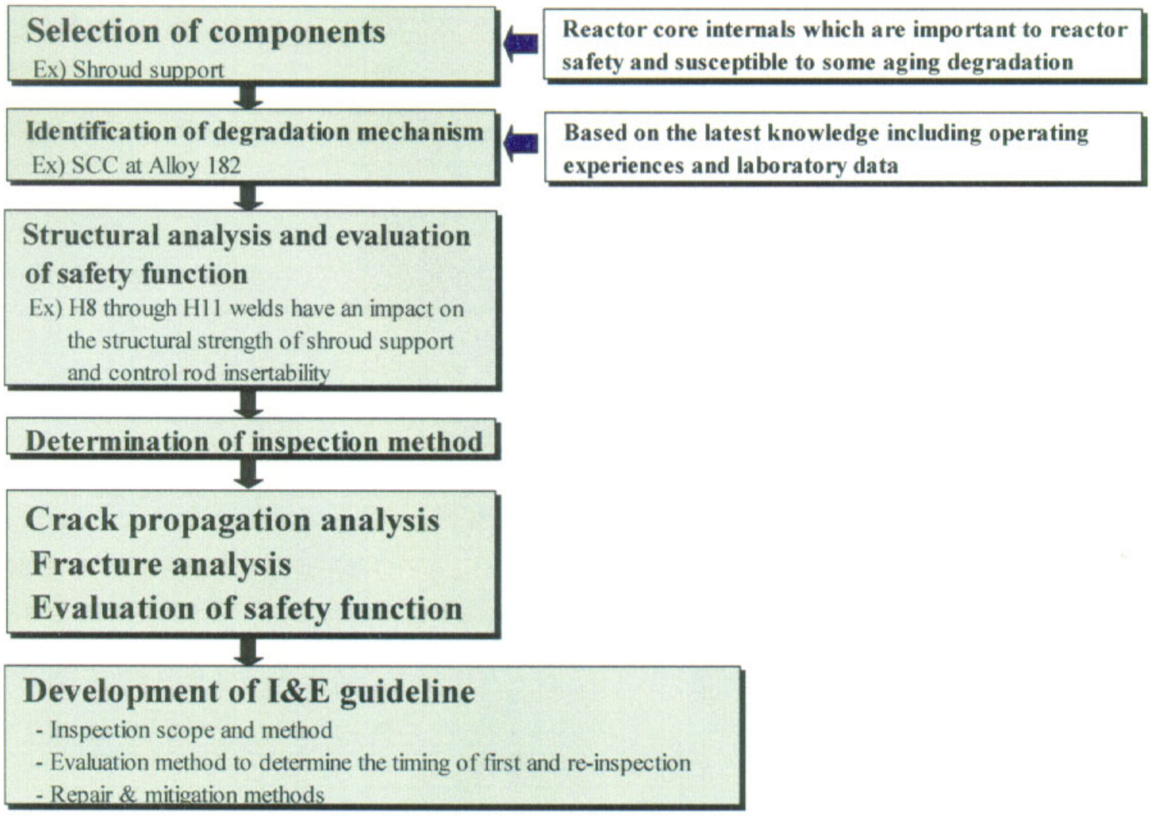

FIG.4. Evaluation steps for the development of I&E guideline.

Components for the development of I&E guidelines were selected from the following viewpoint.

1) Components deemed important to reactor safety or have safety function
2) Components deemed susceptible to some degradation or have the potential of degradation during 60 years of operation

The selected BWR components are the shroud support, core shroud, top guide, core plate, core spray sparger and piping, CRD housing, in-core monitor housing, jet pump and DP/LC piping (Fig.5). The selected PWR components are the baffle former bolts, core barrel, barrel formets, bottom mounted instrumentations, vessel head penetrations, and control rod cluster guide tube (Fig.6).

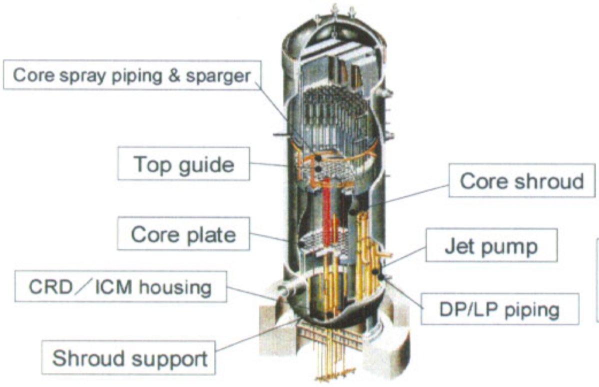

FIG.5. Selected BWR components.

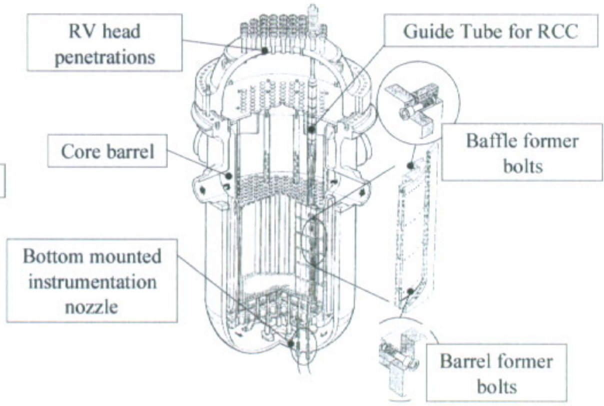

FIG.6. Selected PWR components.

7. I&E GUIDELINES FOR SHROUD SUPPORT

7.1 Shroud support designs

The shroud support is installed at the bottom of the RPV. It supports the reactor core to maintain the function of control rod insertion and to guide the primary coolant flow. It is formed from nickel base alloy plates (Alloy 600) and weld material (Alloy 82/182).

There are several shroud support designs in Japan. The most notable of these is a type of the shroud support plate with legs. (Fig.7). Using this design as an example, the contents of the I&E guideline are, therefore, described in this section.[3]

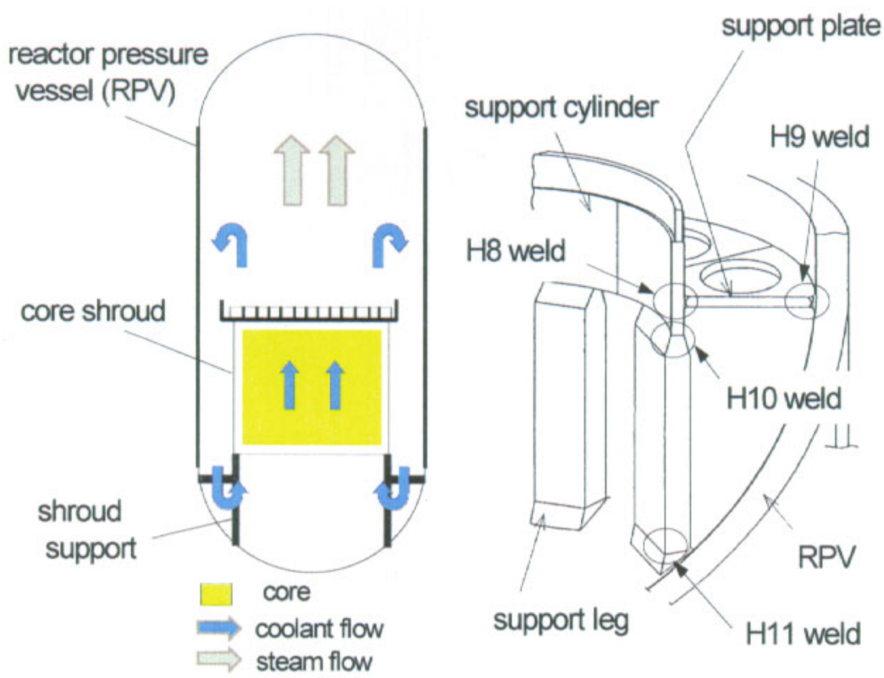

FIG.7. Shroud support design (leg-type).

7.2 Inspection scope

Based on the result of considering the design, fabrication process and operating conditions of the shroud support and also taking account of the latest knowledge including the current operating experiences and laboratory test data, the degradation mechanism was identified to be intergranular stress corrosion cracking (IGSCC) of Alloy 182. Therefore, in many cases structural analyses were conducted under the assumption of IGSCC along welds and the following was clarified.

- Radial and vertical cracks do not significantly affect the structural integrity and safety function of the shroud support.
- Cracks along the circumferential welds (H8 through H11) affect the structural integrity and safety function of the shroud support.

Thus, the circumferential welds with Alloy 182 were selected as the inspection scope of the shroud support structures. (Fig.8)

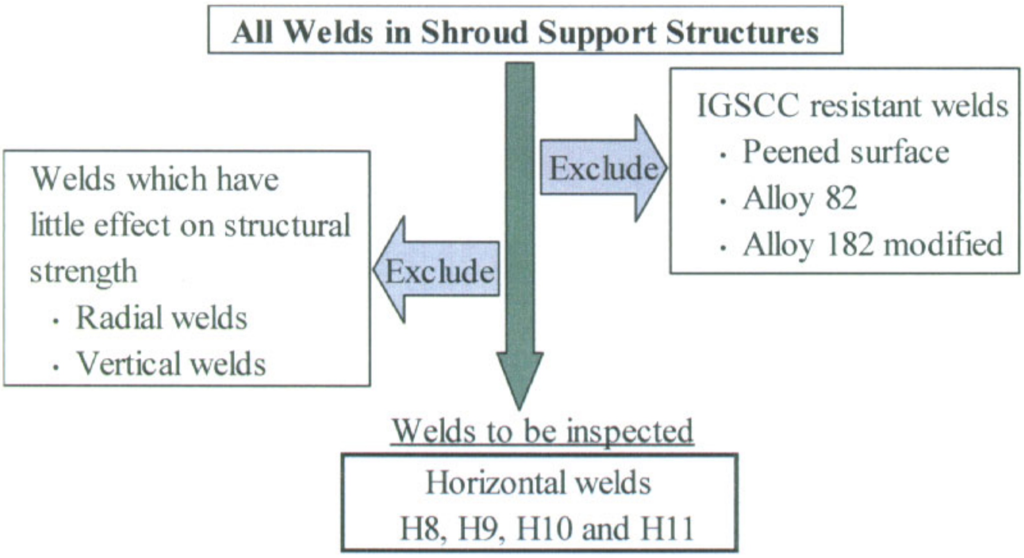

FIG.8. Inspection scope of shroud support structure (leg-type).

7.3 Structural analysis

To evaluate the minimum required cross section at each circumferential weld to maintain the shroud support structural integrity and safety function, structural analyses were conducted by the 3-dimensional finite element method. Collapse load was evaluated based on the double elastic slope method. (identical to the 2tan θ method defined in ASME Sec. III).[4].

The conditions used in the structural analyses are shown in Table 1.

The result of the structural analysis in the case of all welds with no crack is shown in Fig.9.

The collapse load is about 3.05 times as large as the design seismic load S_2. S_2 is the Japanese design earthquake, similar to the safe shutdown earthquake (SSE) of ASME.

Table 1. Conditions of structural analyses

Model Plant	Fukushima Daiichi NPS Unit No.5 (784MWe, BWR/4)
Shroud Support Model	180° Segment
Code	METI Code #501 Limit Load Analysis (2.3Sm Collapse Load)
Load	Normal Operation Loads (dead weight, DP) + S_2 Seismic Load
Acceptance Criteria	(2.3Sm Collapse Load) / (Calculated Load) ≥ 1.5

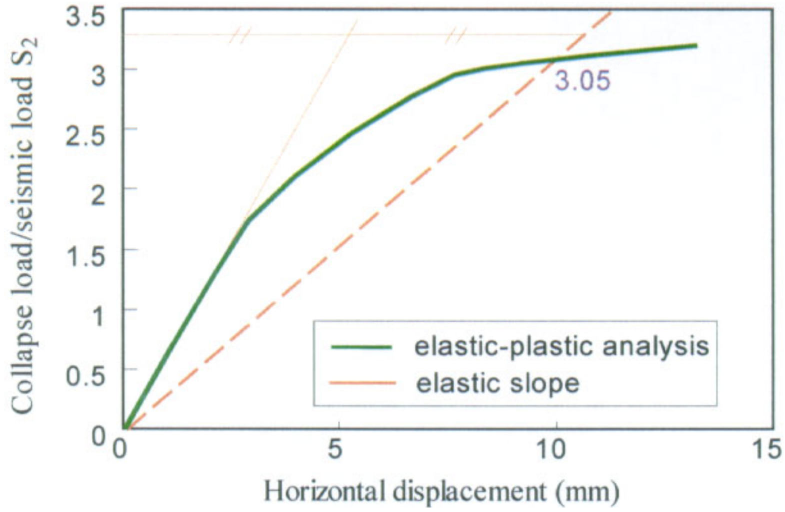

FIG.9. Horizontal displacement at support cylinder top level..

Figure 10 shows the relation between the shroud support collapse load and the H11 weld remaining cross section under the condition that 90% of the welds H8 and H9 is through-wall cracked. This figure also shows that the minimum required cross section for each H11 weld, which is the required cross section to maintain structural and functional integrity of the shroud support structure, is about 25% with a safety factor 1.5. Thus, if the flaw size of each H11 weld is less than 75% of the wall thickness, the shroud support structure could maintain its strength.

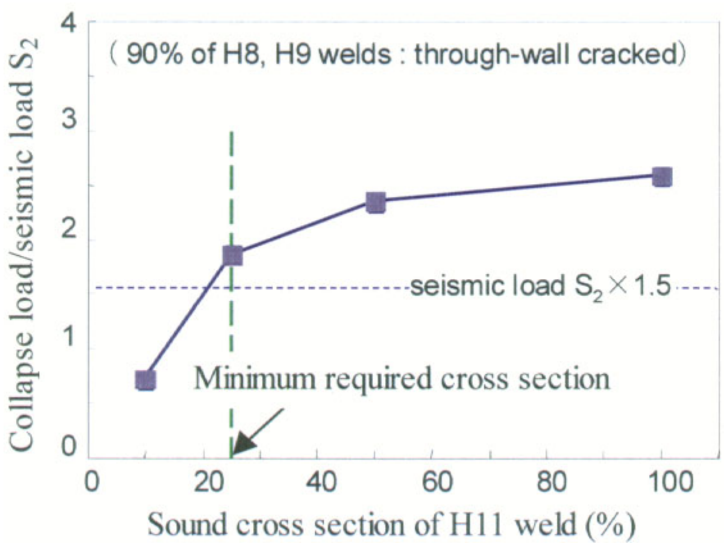

FIG.10. Minimum required cross section of H11 weld.

According to the structural analysis, the horizontal displacement at the core plate level under the condition that 90% of H8 and H9 is through-wall cracked and all H11 welds are 75% cracked is about 3.5 mm. This is smaller than the allowable horizontal displacement at the core plate level obtained by control rod insertion experiment and analysis. Therefore, if the flaw size of the shroud support is smaller than the minimum required cross section described above, it does not affect the function of control rod insertion.

7.4 Crack propagation analysis

Crack propagation analysis of each weld was conducted using the result of residual stress analysis and IGSCC propagation data in normal water chemistry condition. Figure 11 shows the residual stress distribution at H11 weld obtained by thermal, elastic-plastic residual stress analysis.

This residual stress changes from tension at the inner surface to compression at the middle of the wall thickness. This means that IGSCC does not grow to a through wall crack.

Figure 12 shows the IGSCC propagation data of Alloy 182 at BWR normal water chemistry condition. [5],[6],[7] Propagation analysis was conducted using the best-fit propagation curve shown in the same figure.

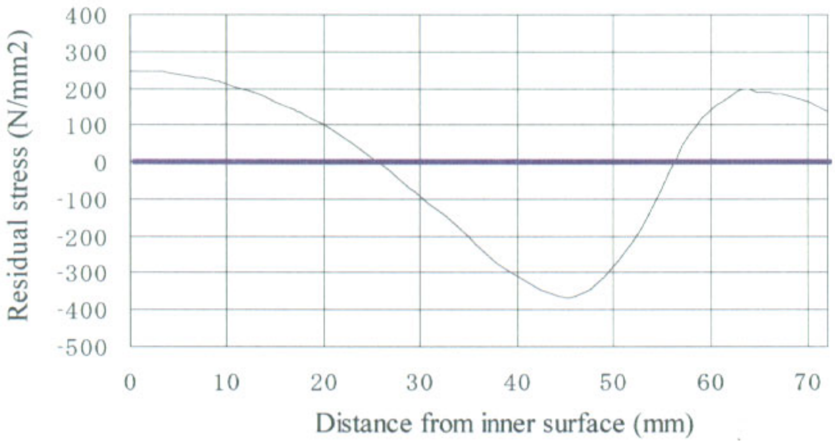

FIG.11. Residual stress distribution in H11 weld.

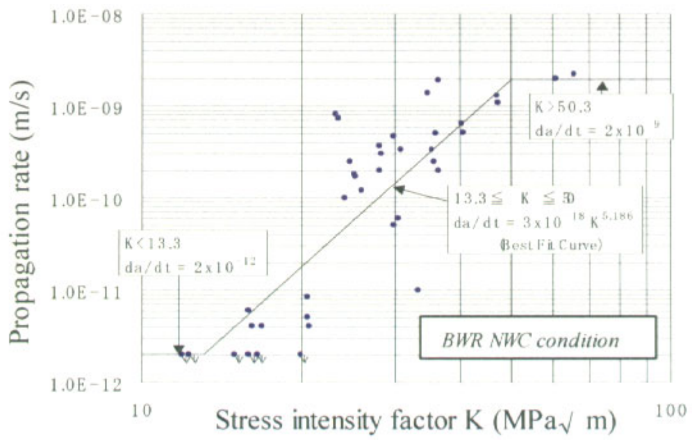

FIG.12. IGSCC propagation data of Alloy 182.

As a result, the IGSCC propagation at the H11 weld was found fastest among the circumferential welds H8 through H11, so the cracking at H11 weld was regarded as critical. The initial crack was conservatively assumed to be semi-elliptical and the size of 50 μm in depth and 500 μm in length (Fig.13). In the evaluation, the incubation time of IGSCC was ignored and the initial crack was assumed to propagate from the commencement of plant operation with the propagation rate associated with stress intensity factor K by Wang's function method. [8],[9] The result of the propagation analysis for H11 weld is shown in Fig.14.

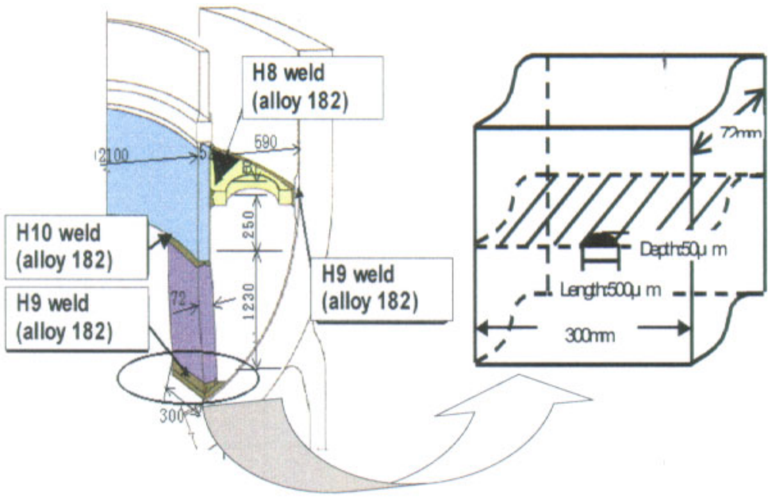

FIG.13. Initial crack size in crack propagation analysis.

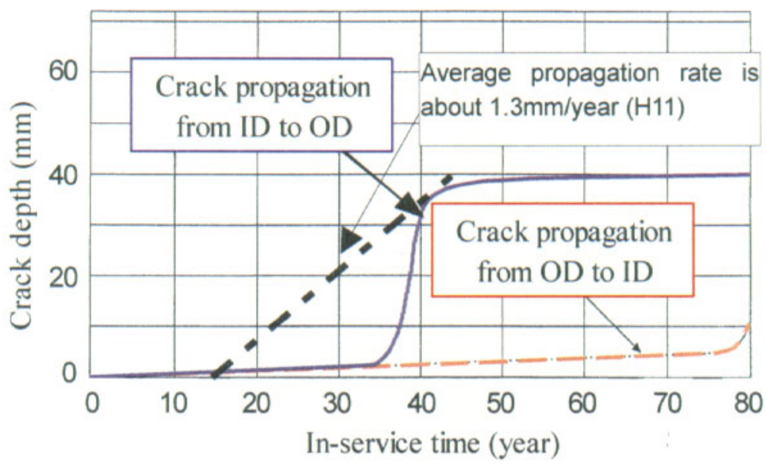

FIG.14. IGSCC propagation in H11 weld.

This shows that the propagation from the inner surface to the outer is faster than that from the outer surface to the inner. The crack depth increases quickly at about 35 years of operation, then decreases quickly and saturate to 40 mm in depth, which is about 55% of the wall thickness of the leg. The average propagation rate from 15 through 44 years of operation is about 1.3 mm/year.

7.5 Inspection program for shroud support structure

7.5.1 Inspection method

It was determined that basically, a visual inspection method capable of achieving a 1 mil (0.025 mm) wire resolution (MVT-1) could be used on a given weld in order to detect IGSCC. UT or ET could also be used if necessary.

7.5.2 Inspection interval and coverage

7.5.2.1 Initial inspection

According to the crack propagation analysis results, the depth of IGSCC at 15 years of operation is about 1 mm, which can be detected by MVT-1. The IGSCC propagation rate increases quickly at around 35 years of operation. Considering these results, it was determined that the initial inspection should be performed between 15 years and 25 years of operation.

It is not necessary to inspect all accessible welds because only 25% of the wall thickness of each leg is required to maintain the safety function of shroud support. However, as the initial inspection is to not only confirm the integrity of the shroud support but to also assure the validity of the I&E guideline, it is recommended that all accessible welds are inspected at the initial inspection.

7.5.2.2 Re-inspection

Inspection intervals and required inspection coverages are evaluated in order to guarantee that the cross section is always equal to or above that of the minimum required cross section. The inspection is required to cover the cross section that would decrease due to crack propagation until the next inspection, in addition to the minimum required cross section evaluated in the structural analyses. Required inspection coverage depends on both the inspection interval and the minimum required cross section. In other words, if the inspection interval is shortened, the required inspection coverage can be narrowed. If a flaw is found by inspection, structural integrity until the next inspection must be guaranteed by expanding the coverage of the inspection in order to satisfy the required inspection coverage.

Minimum required cross section
= cross section to maintain the strength of shroud support structure and safety function (%)

Required inspection coverage
= minimum required cross section +
cross section that would decrease due to crack propagation until the next inspection (%)

In the process to determine the inspection intervals, many conservative assumptions are taken into account. Examples of the conservative assumptions are listed below. (Fig.15)

- The cross section of the weld that can not be inspected due to restricted accessibility is ignored. This means that as it is impossible to confirm the soundness of welds which can not be inspected, is assumed that they are already through-wall cracked. (Fig.15 case 3)
- IGSCC incubation time is ignored so even if the soundness of a weld is confirmed by inspection, the cross section is assumed to decrease with the conservative rate 1.3 mm/year during operation, which is the average propagation rate obtained by the crack propagation analyses. (Fig.15 case 1)
- Should an inspection only on the one side of the weld be conducted and the result reveal no indication, 50% of the wall thickness is assumed to be cracked. Because of the residual stress distribution, IGSCC could not propagate beyond 50% of the weld thickness. (Fig.15 case 2)
- Should IGSCC be found by MVT-1, the depth of the detected crack is assumed to be 50% of the wall thickness. Because the depth of IGSCC can not be determined by visual inspection, it is assumed to be the maximum depth, i.e. 50% of the wall thickness. (Fig.15 case 4)
- The propagation rate of the detected crack along the surface is assumed to be 63 mm/year, which is the upper propagation rate data. (Fig.15 case 4)

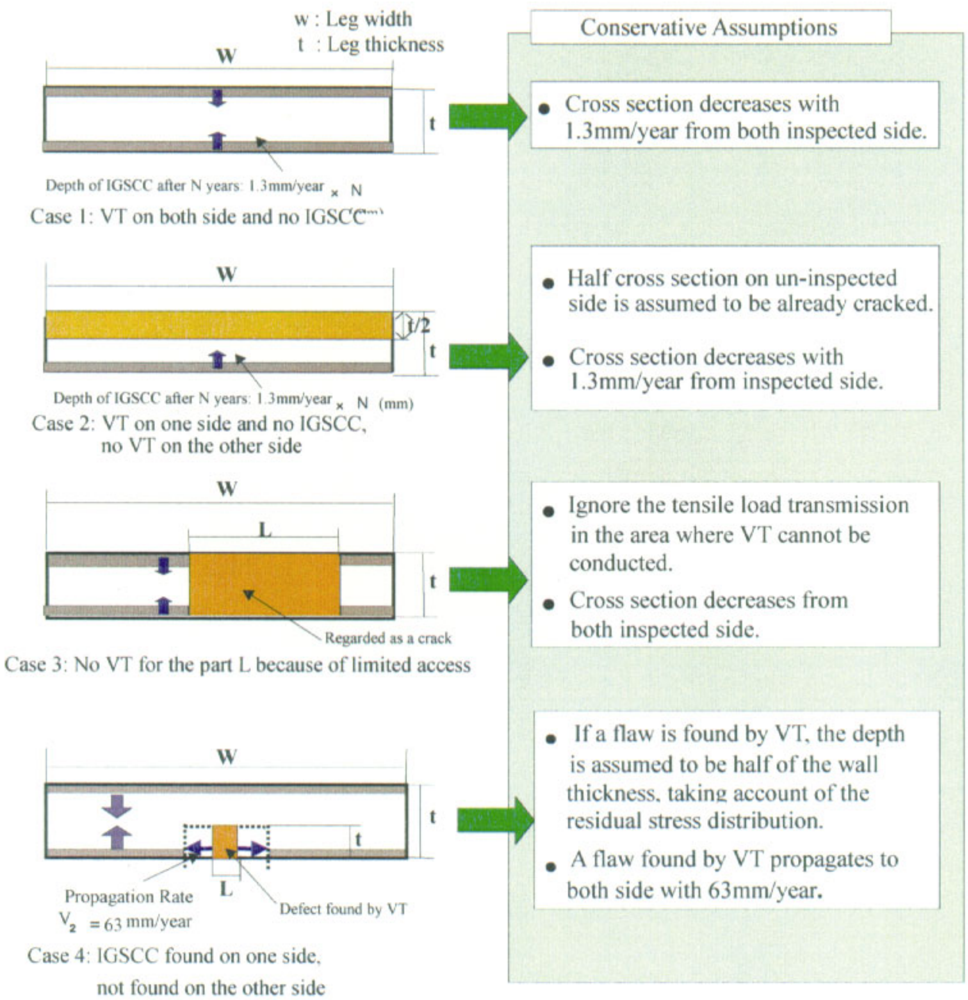

Fig.15. Conservative assumptions in crack propagation analysis.

7.6 Concept of inspection based on I&E guideline

Figure 16 is an overview of I&E guideline for shroud support.

The minimum required cross section of each weld is first evaluated. The interval until the next inspection and required inspection coverage are then determined in order to guarantee that the cross section is always equal to or greater than the minimum required cross section. The inspection is then conducted. Should the inspection coverage be larger than that required, it is possible to continue plant operation until next inspection. Should the inspection coverage be smaller than that required, an expansion of the inspection coverage is required. In this case, it is also possible to continue plant operation until the next inspection.

Wherein the soundness of the required inspection coverage is not confirmed because of flaws, even in the case of expanding the inspection coverage, it is not possible to continue plant operation. In this case, however, if the inspection interval is shortened, it may be possible to continue plant operation.

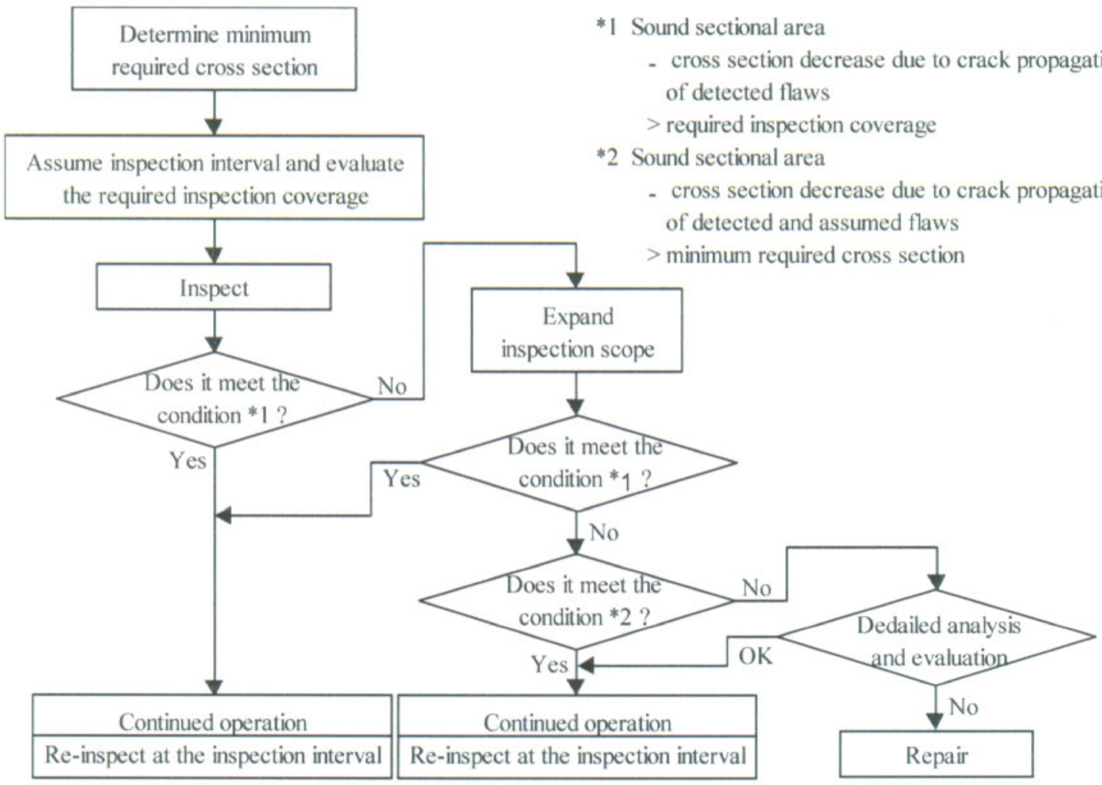

FIG.16. Inspection flow chart of I&E guideline for shroud support.

In the case where it is impossible to justify continued plant operation with the simple method described above, detailed inspection such as crack sizing by UT or detailed analysis and evaluation on the specific conditions of detected flaws is required. Consequently, whether to repair or not is determined.

Figure 17 shows the concept of inspection in the shroud support guideline. After initial inspection, IGSCC is assumed to propagate at a conservative rate during operation. The incubation time is ignored. The implementation of re-inspection is required before the intact cross section becomes less than the minimum required cross section. If no crack is found in the re-inspection, the cross section that decreases during the last operation cycle is reset. Re-inspection is the repeated to monitor or guarantee that the cross section is always equal to or greater than the minimum required cross section.

7.7 Evaluation for a model plant

The typical result of the evaluation for a model plant reveals that the minimum required cross section of each H11 weld is about 25%. In this case, if the inspection interval is set at about 10 years, the required inspection coverage is about 39%. This is because the cross section that would decrease due to crack propagation during 10 years is about 14% of the entire cross section of the H11 weld.

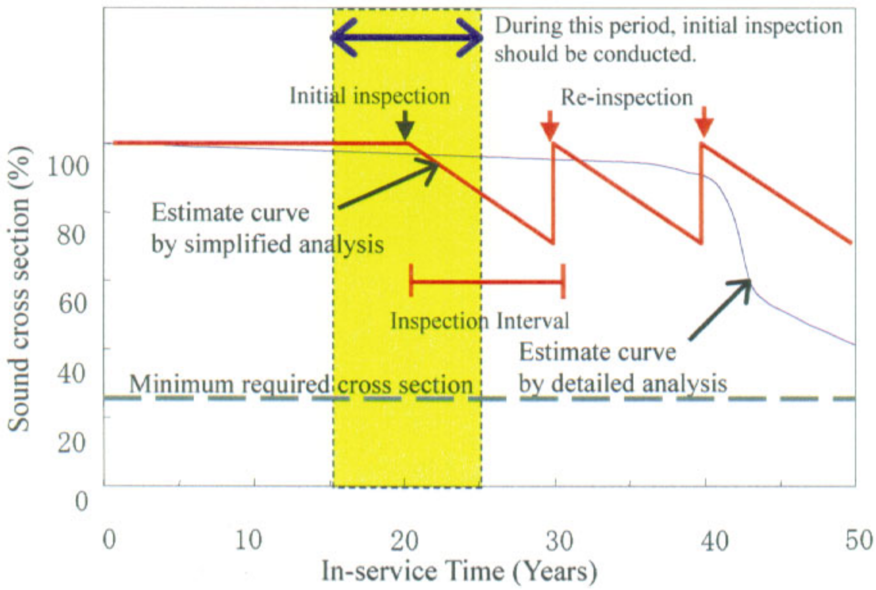

FIG.17. Concept of inspection based on the I&E guideline.

8. CONCLUSIONS

The occurrence of cracks at reactor core internals provides a great impact on plant operation and management. In crisis management it is, therefore, very important to prepare a contingency plan for such cracks in advance.

From this viewpoint, the special committee organized in TENPES determined to develop I&E guidelines for reactor core internals. The policy of this committee was that the guidelines should be technically reasonable and clarify technical reasons or rationale. Accordingly, the I&E guidelines were developed. They provide not only the rules for effective and reasonable inspections that guarantee the structural strength and safety function of reactor core internals, but also provide the technical basis and methodology to formulate an inspection program.

ACKNOWLEDGEMENT

The authors would like to express their sincere gratitude to the Chairman, Dr. Asada, for his proper guidance. They are also grateful to the members of the TENPES Committee on I&E Guidelines for LWR Core Internals.

REFERENCES

[1] Y.Tsujikura et al., "Plant Life Management Studies for Kansai Electric Nuclear Plants", PLIM + PLEX 97, December 1997
[2] A.Minematsu et al., "Plant Life Management Activities of LWR Plants in Japan", PLIM + PLEX 97, December 1997
[3] Thermal and Nuclear Power Engineering Society, BWR Inspection and Evaluation Guideline [Shroud Support] , March 2000
[4] Ministry of Economy, Trade and Industry, Notification #501, Reactor Core Support Structure
[5] M.Itow et al., "Crack Growth Rates of Alloy 182 in High Temperature Water", 7th International Symposium on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, August 1995
[6] M.Itow et al., "The effect of Corrosion Potential on Alloy 182 Crack Growth Rate in High Temperature Water", 8th International Symposium on Environment Degradation of Materials in Nuclear Power Systems – Water Reactors, August 1997
[7] U.Morin et al., "Crack Growth Rates for Ni-base Alloys with the Application to an Operating BWR", 6th International Symposium on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, August 1993
[8] G.S.Wang, "Weight Function Estimation of SIF for Model I Part – Elliptical Crack under Arbitary Load", Engineering Fracture Mechanics Vol.41, No.5, PP659-684 (1992)
[9] I.S.Raju and J.C.Newman Jr., NASA Technical Paper 1578 (1979)

THE HUMAN RESOURCE CONDITIONS OF LIFETIME EXTENSION

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Abstract

According to our present knowledge, the lifetime extension of the Hungarian NPP units will be feasible, in both the technological and economic aspects. It is far more difficult, however, to answer the question whether the human resource conditions of the further application of nuclear energy can be satisfied in Hungary. Many urgent tasks will have to be solved regarding the informing of the public and the nuclear engineering education. By analysing the age distribution of the employees of the Hungarian NPP one can easily foresee that 350 to 400 young engineers will have to recruit up till 2020 (i.e. 15 to 20 per year), while only 4 to 10 students graduate from the Hungarian universities who acquire some level of nuclear knowledge during their studies [2]. In a co-operation between BUTE and the Paks NPP we have been trying to work out a training programme and corresponding scholarship system, which is expected to direct quite a few engineering and physicist students to the nuclear profession. Another equally important programme is to tell secondary school pupils about nuclear topics and the prospective they could have in these fields. Therefore we have been implementing an educational programme, in the frame of which more than 3000 secondary school pupils visit the training reactor of BUTE. There are numerous tasks to be performed regarding informing the public: the lifetime extension of the NPP units will only be politically viable if the people can be convinced of its social profits and if the questions of nuclear safety and radioactive waste can be communicated properly to them. In Hungary, intense work is going on in this field too, into which we are trying to invite the young professionals as well as the undergraduate students. The society needs to be informed that there are young professionals who can see future prospective in nuclear energetics and are able to take over the tasks from the older generation and further operate the equipment in a safe manner.

According to our present knowledge, the lifetime extension of the Hungarian NPP units will be feasible, in both the technological and economic aspects. It is far more difficult, however, to answer the question whether the human resource conditions of the further application of nuclear energy can be satisfied in Hungary. Many urgent tasks will have to be solved regarding the informing of the public and the nuclear engineering education.

The training of nuclear experts is in crisis in many developed industrial countries. The university departments work with a staff mainly consisting of old and quite often near-retirement trainers. In order to show the trends, Hungary and Germany have been selected for this paper as examples. These trends are well reflected in Fig. 1 (for Hungary) and Fig. 2 (for Germany). Similar trends can be observed in different countries all over the world.

As one may conclude, the young generation is practically missing. A particularly grave problem is that in a number of countries hardly any student specialises and graduates in nuclear technology/engineering [1]. To be more specific, in Hungary a few, while in Germany zero (!) students per year have graduated in nuclear engineering between 1998 and 2001. Moreover, several nuclear training and research facilities have been shut down due to different reasons, such as ageing, political decisions, financial problems etc.

In Hungary the rising of the new generation of professionals may soon get into a crisis without immediate intervention. The training reactor of BUTE celebrated its 30th anniversary in 2001 and the technical conditions allow some further 20 or 25 years of operation. On the other hand, however, the age distribution of the operating staff can not be sustained even on a few-year term: the average age is 55 years, while 44% of them are retired (see Fig 1) [2]. Although, due to financing difficulties, the rejuvenation of the operating personnel has not been possible for years, it is definitely vital to maintain and develop the reactor and the ongoing educational work.

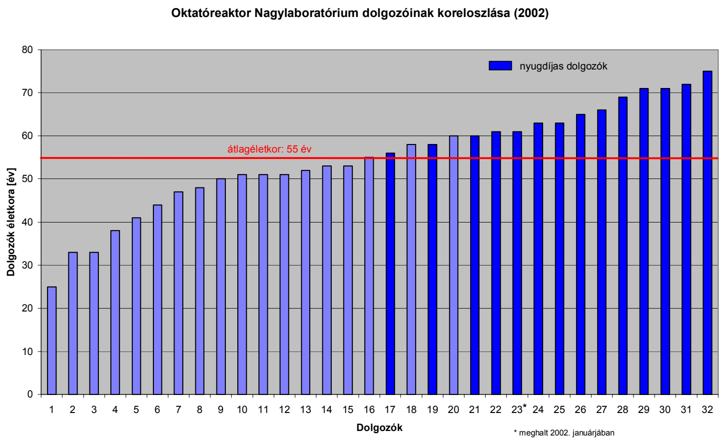

FIG. 1. Age distribution of the employees of the training reactor of BUTE.

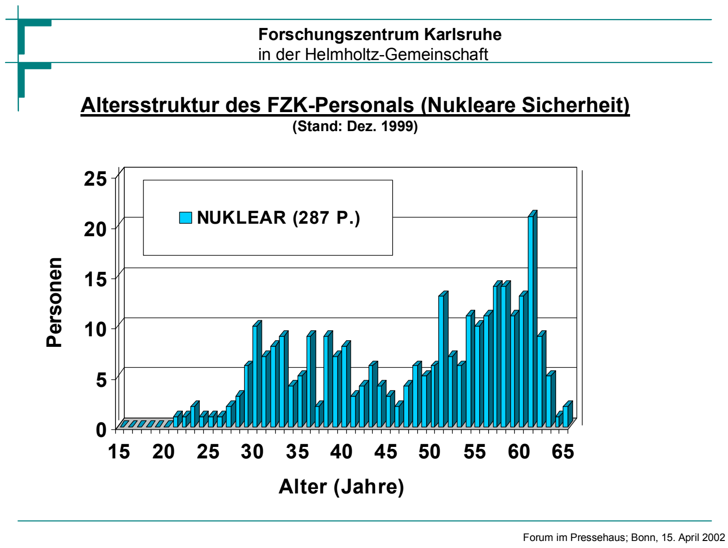

FIG. 2. Age distribution of personnel working at a large German research center

By analysing the age distribution of the employees of the Hungarian NPP one can easily foresee that 350 to 400 young engineers will have to recruite up till 2020 (i.e. 15 to 20 per year), while only 4 to 10 students graduate from the Hungarian universities who acquire some level of nuclear knowledge during their studies [2]. In a co-operation between BUTE and the

Paks NPP we have been trying to work out a training programme and corresponding scholarship system, which is expected to direct quite a few engineering and physicist students to the nuclear profession.

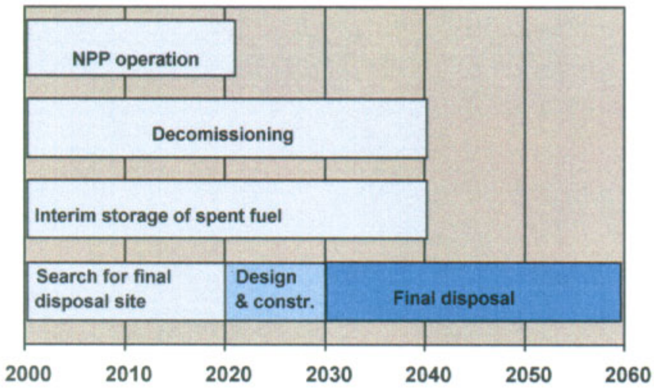

FIG. 3. The steps necessary to cut down the nuclear industry in Germany.

Another equally important programme is to tell secondary school pupils about nuclear topics and the prospective they could have in these fields. Therefore we have been implementing an educational programme, in the frame of which more than 3000 secondary school pupils visit the training reactor of BUTE.

The Western European trends give cause for serious concerns. We are afraid that the increasing need for professionals, e.g. in the German labour market, may draw away many young nuclear engineers from the countries which will join the EU in the coming years due to the higher wages and better reputation of the western workplaces. In this respect one has to consider that even in the case of Germany, where there is a political decision to shut down all NPPs by the year 2021, there will be a massive need for nuclear professionals in the next decades (Fig. 3). Accordingly, we also have to account for the fact that the university training will not only have to satisfy the domestic human resource requests, but we will have to train professionals for the Western European nuclear labour market (including research) as well.

There are numerous tasks to be performed regarding informing the public: the lifetime extension of the NPP units will only be politically viable if the people can be convinced of its social profits and if the questions of nuclear safety and radioactive waste can be communicated properly to them. In Hungary, intense work is going on in this field too, into which we are trying to invite the young professionals as well as the undergraduate students. The society needs to be informed that there are young professionals who can see future prospectives in nuclear energetics and are able to take over the tasks from the older generation and further operate the equipment in a safe manner.

REFERENCES

[1] Peter Fritz: Kompetenzerhalt in der Kerntechnik, Forum im Pressehaus; Bonn, April 15, 2002

[2] A. Aszódi: Expert needs in nuclear energetics and nuclear research, Hungarian Academy of Science, Scientific Assembly: “Nuclear science in education: summary and recommendations”, Budapest, April 26, 2002 (presentation in Hungarian)

Session 5

CURRENT NATIONAL APPROACHES TO PLIM

Sub-Session 5.1

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INTEGRATED NPP LIFE CYCLE MANAGEMENT – THE WAY TO PRESERVE GENERATING CAPACITIES

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Abstract

For the past couple of decades there has been a change of emphasis in the world nuclear power from that of building new Nuclear Power Plants (NPP) to that of taking measures to optimize the life cycle of operational plants. National approaches in many countries showed an increase of interest in Plant Life Management (PLIM), both in terms of plant service life assurance and in optimizing the service or operational life of NPP. A strong convergence of views is emerging from different National approaches, particularly in the area of the economic aspects of NPP operation and in the evolution in the scope of NPP PLIM. Recognizing the importance of this issue and in response to the requests of the Member States the IAEA Division of Nuclear Power implements the Sub-programme on ”Engineering and Management Support for Competitive Nuclear Power”. Four projects within this sub-programme deal with different aspects of the NPP life cycle management with the aim to increase the capabilities of interested Member States in implementing and maintenance of the competitive and sustainable nuclear power. Although all four projects contain certain issues of PLIM there is one specific project on guidance on engineering and management practices for optimization of NPP service life including decommissioning. This particular project deals with different specific issues of NPP life management including aspects of ageing phenomena and their monitoring, issues of control and instrumentation, maintenance and operation issues, economic evaluation of NPP life cycle management including guidance on its earlier shut down and decommissioning.

1. INTRODUCTION

The IAEA provides a global focus for nuclear co-operation in many areas. Presently, an area of major interest is the management of nuclear power plant (NPP) life cycle from concept development to decommissioning and disposal, with the primary objective of maximising the return on investment in nuclear facilities through efficient operation of NPPs.

In 2001, 438 NPPs, with a capacity of about 350Gwe, supplied 16% of the global electricity. Of these, about 300 NPPs have been in operation for 15 years or more and these older units with partially or fully amortized capital costs have proven to be the most profitable. Moreover, there are no significant safety or economic reasons not to continue the operation of well-managed NPPs over a longer period and consequently the issues of plant life management and license extension are receiving an increasing emphasis in many countries.

Forecasts of nuclear power growth over the next two decades range from 350Gwe in the worst case to 500Gwe in the best case. This will need additional personnel and expansion of the infrastructure in the developing countries, particularly as much of the new demand growth is forecast to take place outside the countries where most of the existing infrastructure resides.

All aspects of NPP life cycle management are addressed within the IAEA and are briefly described in this paper.

NPP Life Cycle

The Life Cycle consists of planning, design, financial arrangements, construction, commissioning, operation, maintenance and inspection, management of the workforce capability, economic assessments of license extension, decommissioning and waste management.

Such an integrated approach requires the consideration of a variety of activities to be undertaken during the plant life management programme (PLIM). *PLIM can therefore be defined as the total activities, including technical, financial, economic, administrative - managerial and socio - political aspects that are aimed at the achievement of a long term save and reliable operation within the optimised life cycle of the plant.*

Achieving a safe and economic extension of the plant operational life is equivalent to providing additional generating capacity. This involves an extension of the operational license to the maximum extent possible while maintaining its safety and competitiveness.

It should be remembered that the purpose of the NPP is to generate and sell electricity economically and its revenues have to support the financing of its entire life cycle.

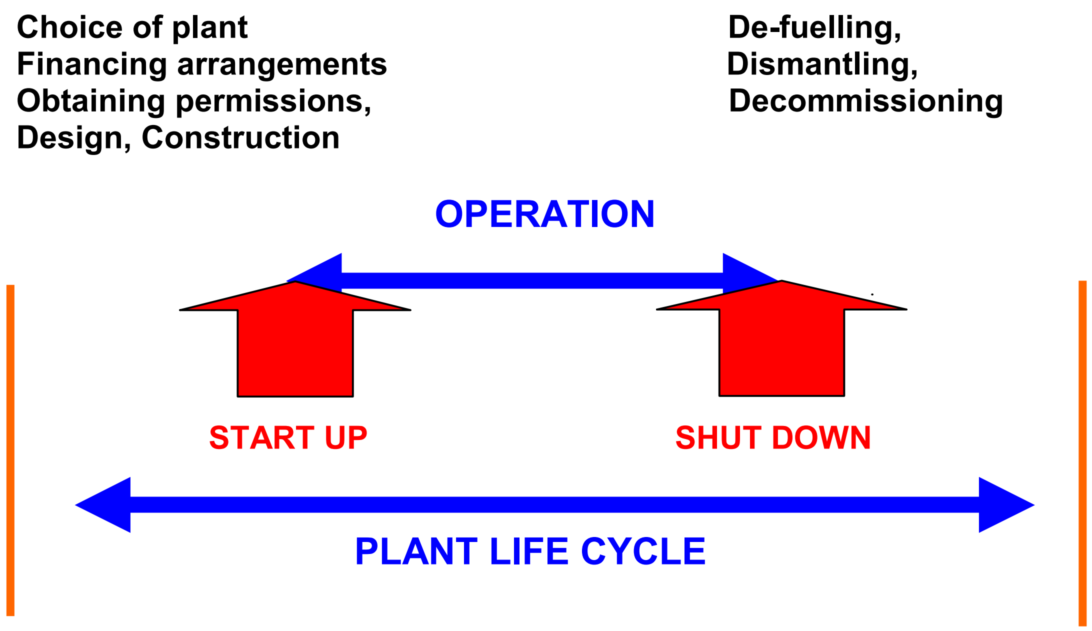

FIG. 1. Diagram showing the relationship between NPP life cycle and the period of operation (or service)

Definitions of Commonly Used Terms.

We can define ***Plant Life Cycle*** as the period of time, which includes pre-operational activities, such as the initial choice of the NPP, its design, procurement, construction, commissioning, operation and also the post-operational activities of decommissioning and the return of the site to a 'green field'. It encompasses the entire period during which any financial charges can be ascribed to the NPP. In the industry jargon, this period is described as 'before the cradle *(conception)* until the grave'.

Plant Operational Life *or* ***Service Life***, refers to the period when electricity is generated and sold providing the source of revenue to cover all the costs during the Plant Life Cycle. This period is not defined either by the designer, the owner or the regulator.

Design Life of a NPP may be defined as the minimum plant operational or service life assured by the designer provided all specified requirements for manufacturing, construction, commissioning, operation and maintenance are observed. During some stage (about halfway in the operation) the condition and performance of the plant are assessed against the design intent and the operational life may be confirmed or re-evaluated.

As the NPP is essentially a business venture, it has to recover its capital costs over a period of time called ***Amortisation Life***. This varies from plant to plant and affects the cost of electricity generation. For example, the Amortisation Life for Sizewell-B in the UK is 40 years. Many plants have an Amortisation Life of 25 years. If a plant is shut down for any reason after an operating period shorter than its Amortisation Life, it would result in a significant financial loss.

For the initial economic assessment Service Life is taken to be the Design Life, which is substituted by the re-assessed life at a later stage. Recently, after a 10 year investigation, the French utility EdF announced that no technical reason could be found for its PWRs not to be operated for 40, 50 or 60 years. Such an increase in the Service Life beyond the usual Amortisation Life of 25 to 30 years will have a direct bearing on the cost of electricity from these plants and significantly enhance its profitability. Similarly in Japan the utilities are reassessing their NPP operational life and the results of early assessment lead to a conclusion about the possibility of an operational life of 60 years.

Because the NPP Service Life is not a defined period the term "Plant Life Extension **(PLEX)"** becomes unacceptable. Due to a realisation of this fact, the term PLEX has increasingly fallen into disuse during the past 5-10 years. There are similar difficulties with the term **Residual Life**, the estimate of which requires knowledge of the criteria and conditions for defining the **End-Of-Life (EOL)**. There could be many criteria to define EOL including design, economic, technical, political, and other criteria. For example, the EOL of a NPP can be reached when politicians decide upon it, or when the cost of electricity it produces is too high, or the continued operation of the plant is declared to be unsafe or when some of the equipment has deteriorated to a level which is deemed as being unacceptable and it is not technically or economically possible to repair or replace.

To summarize, the period of NPP Life Cycle encompasses the early conceptual phase through choice, approval, financing, construction, operation, shutdown and de-commissioning to the eventual restoration of the site to a 'green field' and includes the eventual disposal or re-assignment of responsibility for the fuel and waste. **So, in practical terms the full scope of**

the 'Life Cycle' of a Nuclear Power Plant covers the entire period when financial charges are incurred by the project.

While extending the operating license of an existing nuclear power plant has several advantages, both the owner and the regulator should ensure that the power plant is in the same operating condition on the last day of its license as it is on the first. [2]

To meet these requirements the IAEA is implementing programmes that deal with different aspects of an integrated NPP life management approach.

IAEA Activities

The International Atomic Energy Agency (IAEA) is an independent, intergovernmental, science and technology-based organization, within the United Nations family and serves as global focal point for nuclear co-operation. The three main pillars of its activity are safety, technology and safeguard. According to its mandate, the IAEA assists its Member States (134 at present) in planning for and using nuclear science and technology for peaceful purposes, which includes the generation of electricity. It also develops nuclear safety standards, promotes achievement and maintenance of high levels of safety in all application of nuclear energy. In addition, the IAEA verifies that States comply with their commitments, under the Non-Proliferation Treaty and other agreements, in using the nuclear materials only for peaceful purposes.

In response to the needs of the Member States the IAEA (Agency) is implementing several programmes on engineering and management practices for optimisation of nuclear power plant life cycle. Majority of these activities is concentrated within the IAEA Division of Nuclear Power under the Sub- Programme A1 "Engineering and Management Support for Competitive Nuclear Power". Four projects within this sub-programme deal with different aspects of the NPP life cycle management with the aim to increase the capabilities of interested Member States in implementing and maintaining a competitive and sustainable nuclear power programme.

While all the projects under the IAEA Sub-Programme A1, contain certain issues of PLIM there is one dedicated project on "guidance on engineering and management practices for optimization of NPP service life, including decommissioning", as the focal project on the subject. The Project scope includes different categories of IAEA activities for the purposes of information exchange and technology transfer. These include arrangements for technical meetings, development of databases and guidance documents on proven practices, co-ordinated research projects, technical co-operation projects, direct expert services to Member States, training activities and co-operation with other International organizations.

Technical Working Group on Life management of Nuclear Power Plants (TWG-LMNPP)

The Agency's programme regarding life management is systematically reviewed by the Technical Working Group on Life Management of Nuclear Power Plants (TWG-LMNPP). The TWG is composed of representatives from 27 countries and 2 international organizations. For more than 25 years, it has been providing the IAEA with recommendations on, the current and future activities that are considered important to NPP life management as well as the needs of Member States for information exchange and technology transfer.

The TWG activities are mostly concentrated on studies of understanding mechanisms of degradation and their monitoring, management of maintenance options, economic aspects and human factors in approaches to NPP life management. All these activities are oriented to the assistance to Member States to develop their capabilities for an informed decision making process.

Certain topical priorities have been recommended by the TWG and accepted by the IAEA. These are given below in the order of priority:

1. *RPV Integrity*
2. *Steam Generator Life Management*
3. *Primary Circuit Operation and Integrity*
4. *Reactor Internals Operation and Integrity*
5. *Secondary Circuit*
6. *Containment/civil structures*
7. *Cables*
8. *Other items of importance (valves, pumps)*
9. *Economic aspects of life management programmes in relation to plant management, safety and maintenance.*
10. *NPP pre-shut down and decommissioning activities*

Other important items of general importance to all components include:

- guidelines and recommended practices
- international aspects of codes and standards
- international databases
- quality assurance
- regulation requirements and licensing procedures
- maintenance man-dose management
- component cleaning.

The future programme of activities the Agency follows recommendations of Member States and includes the management of NPP decommissioning as a part of the plant life management issue. These changes are reflected in the forward Agency's programme for the years 2003-2005.

One of the most important activities performed by the Agency is co-ordinated research in different subject areas. Within the framework of the TWG-LMNPP there are four Co-ordinated Research Projects (CRP) under way dealing with the studies of degradation mechanisms and in particular with regard to irradiation embrittlement of RPV materials. They are as follows:

- Surveillance Programme Results Application to RPV Integrity Assessment
- Mechanism of Nickel Content Effect in Radiation Embrittlement of RPV Materials.
- Evaluation of Radiation Damage of WWER Reactor Pressure Vessel using the IAEA Database on RPV Materials.
- Verification of WWER Steam Generator tube integrity.

Operating experience has shown that NPP operational life is affected by component degradation and ageing and therefore highlighted the need to develop the methodology allowing for an improvement in the understanding of these processes.

The latter in turn will provide the possibility to manage ageing effects in a timely and planned way and also to create a strategy in the NPP Life Management with the aim to achieve economic viability of the plant while observing necessary safety and operational margins. The Agency mainly looks at technological and phenomenological aspects of ageing and degradation processes and takes into consideration practical advice on maintenance, mitigation, repair, replacement and refurbishing. It has been recognized recently that this approach should be complemented by relevant economic analysis to achieve an effective decision.

The assessment of the component life relies on an ability to assess its condition and predict future degradation trends. This is a background for the strategy of the plant operational life assurance that to a large extent is dependent on the availability of relevant data. It has been recognised that for effective management of ageing and degradation related processes a large amount of data is needed. Therefore data availability is the key aspect in evaluation of the components state and therefore in the decision making process related to their life management.

A systematic collection and screening of the information available and the development of databases that could assist in research and decision making process are important activities in the IAEA programme.

Several years ago the Agency started the work on the International Database on NPP Life Management. This is a multi-module Database (Fig.2) first one of which is called the International Database on Reactor Pressure Vessel Materials (IDRPVM) and was implemented already in 1996 and is being used world wide. Four other modules are also completed and the work on receiving the data and filling it into the databases is under way.

One of the important parts of the Integrated NPP Life Management Programme is the issue of maintenance and in-service inspections as the tools for the optimisation of the operational or service life of the plant.

Maintenance Programmes and In-Service Inspection

Assurance on a long-term basis of both functional and structural integrity and reliability of nuclear power plant components is a basic condition for any plant life management activity. Component reliability is undoubtedly necessary to maintain plant safety but also to help guarantee the plant availability. Functional integrity of the active components is possible to be maintained by the plant maintenance programme, and in the context of structural integrity and reliability of passive components (pressure vessels and piping), in-service inspection (ISI) and on line and/or periodic condition monitoring are the basic mechanisms to provide information concerning both the presence and size of flaws as well as revealing other deviations from component operating regimes that could threaten their structural integrity. Consequently, maintenance and in-service inspection are fundamental elements of plant life management and their effectiveness strongly influences the production and safety performance of the plants.

A technical document entitled *Optimisation of NPP maintenance programmes* is under preparation as a part of this activity. The goal of this work is to identify approaches and methodologies in Member States on how maintenance programme optimisation contributes to NPP performance improvement and entire life management. It is recommended to see the process of maintenance optimisation as a continuing process, which is driven by the imbalance between maintenance requirements (legislative, economic, technical) and resources used (people, spare parts, consumable materials, equipment, facilities, collective doses). The process includes details as to how selection of maintenance techniques is achieved to enable the most appropriate type of maintenance is performed on systems, structures and components and at what periodicity to achieve regulatory requirements, maintenance targets concerning safety, reliability and plant availability and cost. This approach can be used also in establishing a preventive maintenance programme.

The initial step of the process provides the benchmark for the maintenance indicators that will be used to recalculate the benefit of the optimisation. Various maintenance optimisation techniques and tools are available and applied to the various extents in the Member States. The process seeks to make the best use of Condition Based Maintenance where unnecessary costly maintenance actions and associated maintenance error induced failures can be avoided. Reliability Centred Maintenance, a methodology aiming at identifying the critical components based on the safety and operational consequences of their failures, plays an important role as well. If a probabilistic risk assessment has been performed, its results can be used to help define the important systems and components.

Preparation of another technical document about *Good practices on ISI effectiveness improvement* is in progress. Results of round-robin tests within various international research projects as well as the appearance of real in-service defects in pressurized components (e.g. cracks caused by stress corrosion or thermal fatigue) have demonstrated the need for improving ISI effectiveness. There are two ways to be pursued, which should be integrated to strengthen each other's potential. The first is selection of those components whose failure has a relatively higher probability, a more severe consequence and where the mitigation action, as a result of a proper NDE, effectively reduces the risk of failure: this is called risk informed ISI. The other way to improve ISI effectiveness is the transition from a prescriptive ISI code to a performance oriented one with the demonstration of the capability of ISI systems that the ISI objective can be achieved. The latter one is the inspection qualification or performance demonstration.

ISI qualification is a world wide issue: majority of the NPP operating countries both developed and developing have established their qualification infrastructure or are in the process of establishing it. Experiences on technical, organisational, regulatory and economic aspects of inspection qualification have been being accumulated. Expansion of risk informed ISI seems much more limited. There are developed countries where risk informed ISI programmes tend to replace those based on the conventional ISI codes like ASME Code Section XI, and are about to find the place in the everyday ISI practice. In other countries, risk informed ISI is under consideration. Of course, there are other options, which may contribute to ISI effectiveness improvement. For example, proper methodology and technique for complicated inspection objectives like dissimilar metal welds or complex geometry ferritic or forged austenitic components. Human reliability associated with moral problems and key staff shortage can also impact overall ISI effectiveness. The specific concept of the technical document is to treat these two issues in their complementarities, define fields and the logistics of their interaction and to assess the impact of such concepts on the NPP life management.

2. OPERATIONAL ISSUES

An important role in the PLIM integrated approach and namely in the optimisation of the operating or service life plays the issue of the outage management.

Outage management is a complex task, which involves in respect of the plant policy, the co-ordination of available resources, safety, regulatory and technical requirements and, all activities and work before and during the outage.

The fundamental basis for outages during the lifetime of a nuclear power plant (NPP) is heavily affected by plant design and layout. The choice of fuel cycle length, desired mode of operation, maintenance periods for the different components, requirements of authorities and the electricity market may significantly affect duration and frequency of outages.

Although, most of the main components of a nuclear power plant are designed for plant lifetime operation, some other equipment might need to be updated or exchanged in a plant operating for as long as 40 or 60 years. Refurbishment programs are being planned as long-term activities and in accordance with cost benefit analyses and their importance to safety and availability. It is a key factor for outage optimization to coordinate refurbishment programme and long term outage planning.

Improving the overall economics of a nuclear power plant requires a comprehensive understanding of the relationship between O&M costs and the performance of the plant. It should be recognized that there is a real cost associated with poor performance (lost opportunity for receiving revenues, higher then necessary cost to generate, etc.) as well as the corrective maintenance cost associated with repairing equipment. In addition, there is a mutual interaction between O&M spending and the performance of the plant. Too little pro-active (preventive) O&M spending results in a high frequency of unplanned breakdowns with high corrective maintenance cost and high cost associated with unavailability. The practice of too much O&M spending can put the plant behind the point of diminishing returns. The goal, therefore, is not to minimize O&M cost or to maximize performance (availability, etc.) but rather to minimize the total cost by optimising the O&M cost.

In this context, the Agency has implemented a number of activities directed to outage management and strategy and outage optimisation. The Agency has also developed and implemented in cooperation with the Nuclear Committee of Electric Utility Cost Group (EUCG), the Nuclear Economic Performance Information System (NEPIS) to directly support this optimisation process by providing insight into each of the three steps listed above. In this first phase of its development NEPIS focused on operating and maintenance costs (O&M).

3. CONTROL AND INSTRUMENTATION

The majority of NPPs in the world still have the greater part of their original analogue and aged digital I&C systems in operation. A major driver for the modernization of I&C systems is the need for more cost-effective power production, for improved competitiveness, and for replacing obsolete equipment. In addition, plants, which are pursuing life cycle optimisation and license extension, need to modernize their I&C equipment in order that a number of the systems would be able to satisfy the extended operation requirements.

As NPPs receive license extensions and look forward to decades of continuing operation, they will inevitably modernize their aging and obsolete I&C systems. However, this modernization will, for the most NPP's I&C, be implemented within several years, and careful planning and maintenance of aging equipment is necessary in the meantime. The I&C modernization program is usually focused on the cost effective modernization project management and impact of the technical and financial success of the transition to updated digital I&C systems.

The aging monitoring program requires significant technical and scientific background activities. The basic question is whether the overall cost of operation and aging management together ensures a lower electricity price than replacing a large amount of components would do. These calculations shall be performed periodically and the lifetime management strategy has to be modified according to the changes of technical and economic conditions.

4. PRESERVATION OF KNOWLEDGE

Management of the NPPs long-term safe and efficient operation requires that their personnel possess and maintain the requisite knowledge, skills, and abilities, to do their jobs. This includes managers, operators, technicians, electrical and mechanical maintenance staff, instrumentation and control technicians, health physicists, chemists, engineers, supervisors, scientists, and supervisors and others depending on the individual NPP organization.

The nuclear power industry in Member States has invested, and continues to invest considerable resources in implementing Systematic Approach to Training (SAT) and SAT-based training programmes, consistent with guidance provided by the Agency in its published Guidebook TRS #380 "Nuclear Power Plant Personnel Training and its Evaluation" [3].

On the basis of experience gained worldwide in the SAT application, SAT - based training represents now a broad integrated approach emphasizing not only technical knowledge and skills but also human factor related knowledge, skills and attitudes.

The increased control and accountability features of the SAT process provide management and the regulator with the means of applying standard QA procedures and processes at any stage of the training process. The requirement for the training process to conform with the plant QA programme provides management and the regulator with far greater confidence in the qualifications and competence of personnel than that provided by a purely examination-driven assessment. SAT is a QA approach to training and therefore plays a significant role in the overall nuclear power plant QA programme.

SAT based training eliminates or minimizes the competency gaps that affect nuclear power plant safety and efficient operation. A SAT based training system provides continuously inputs for other processes to enhance NPP safety and reliability, such as the upgrading of plant procedures, systems and organizational structure, as well as human resources management during the whole Life Cycle of the plant.

5. ECONOMICS OF PLANT LIFE CYCLE OPTIMISATION

An operating NPP needs to pay for the construction and design costs, in addition to operating costs, from the revenue received. In addition it needs to ensure that adequate funds are generated and retained to cover future decommissioning and waste liabilities that will need to be paid when the plant is no longer generating revenue.

Generally plants are designed to achieve an identified operating life, and the capital costs are paid from revenues received during this identified period. If the plant is able to operate beyond this identified period then there are no costs associated with the initial capital investment and hence the plant is able to operate with much greater potential profit margins.

The decision to permit the extension of a license for an NPP is quite complex. It may involve a number of political, technical and financial issues, and may require significant investment to justify the license extension.

To address Member States needs on the economic aspects of license extension/operational or service life optimization the Agency embarked in a number of activities aimed to gather world wide existing experience to assist utilities in Member States. Some of them are given below.

A technical document on Cost drivers for the assessment of NPP life extension (TECDOC-1309) [4] has recently been published The objective of this technical document is:

i). To provide an understanding of the various cost elements and drivers in NPP life management.
ii). To present cost data collected through a questionnaire sent to IAEA Member States and to discuss and identify the basis of the available cost estimates of different activities. This will allow users to draw their own conclusions for input into the economic assessment.

The Agency is also involved in the development of some practical tools in the economic assessments. One of them is *The computer model for economic assessment of NPP Plant License extension.* The objective of this activity is to develop a PC based computer model to assist plant owners in Member States with the assessment of economic effectiveness of extension and other generation options including consideration of deregulated/privatised electricity markets. What level of nuclear performance will be required to become and remain competitive; what will the market price of electricity and other energy products be; what will be a sufficient return on investment and at what level of risk; what's the value of a NPP at auction; how much can one bid and still make a profit; and what generation alternatives should be considered. The ongoing activities and the objective and the scope of the package were agreed and a detailed plan of further actions was prepared. The plan received the name FINPLAN and is intended for the use in the development of the above mentioned computer model. It is expected that the model will be completed in the second half of 2003.

6. DECOMMISSIONING

In accordance with the scope of its activities the TWG-LMNPP pursues also the work o NPP decommissioning in a different aspects of the issue but first of all from the point of view of preparation of these activities and their cost analysis.

In a technical document, the *Decommissioning costs of WWER-440 NPPs* were presented and analysed in a uniform manner, using the cost item and cost group system of the joint EC/IAEA/OECD-NEA Interim Technical Document on Nuclear Decommissioning, “A Proposed Standardised List of Items for Costing Purposes” to provide a basis for understanding decommissioning costs differences. It was shown that the standardised list might facilitate communication, promote uniformity and help to avoid inconsistency in results of decommissioning cost assessments. The technical document describes the first exercise in which decommissioning costs for NPPs were converted to and presented in accordance with the structure recommended.

The document comprises a presentation and analyses of the costs for two decommissioning options for WWER-440 NPPs, i.e., immediate decommissioning and safe enclosure. The specific characteristics of individual cost items could be identified and understood. When interpreting individual cost items in the recommended cost structure, also the boundary conditions for the individual decommissioning projects were clarified.

The total costs for the immediate decommissioning option vary from 219 MUSD (Finland) to 1,370 MUSD (Germany). This large difference is mainly due to country and site specific conditions. In the case of Finland the possibility for on-site disposal of all dismantled materials reduces the costs dramatically. In the case of the Greifswald project (Germany) major costs for post-operational and site support activities, as well as the construction of a large interim storage on the site are included. For the safe enclosure option the cost figures vary from 210 MUSD (Czech Republic) to 469 MUSD (Hungary). In this case the difference in the cost estimations is smaller, but still significant. The reason for this is the difference in scopes that are included. Comparison of the cost groups (Labour costs; Capital, equipment and material costs; and Expenses) has demonstrated that about 50 % of the total decommissioning costs is due to labour requirements. Comparison of these results with OECD/NEA cost study results has shown quite good agreement. It may be concluded, therefore, that WWER-440 NPPs are certainly not ‘unique’ from the point of view of their decommissioning costs.

The Agency also co-operates with other International organizations in the studies related to NPP Life cycle management, and in this case on decommissioning policies, strategies and costs. It joined the study initiated by Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (OECD/NEA) on “Decommissioning Policies Strategies and Costs” for the period 2001 - 2002. The main goal of the study is to provide policy makers with comprehensive information on key issues related to decommissioning, based on an international survey, that they might use in support of developing national approaches and strategies in the field.

The specific objectives of the project are:

- To compile and analyse comprehensive information on economic and financing aspects of decommissioning nuclear reactors, with emphasis on commercial power plants.
- To review policies implemented or planned for decommissioning nuclear reactors, with emphasis on commercial power plants.
- To prepare a report for policy makers on decommissioning strategies, costs and financing.

The study is carried out by an Expert Group composed of representatives from Member Countries of the OECD, the Agency and the European Commission (EC).

The Group elaborated the objectives and scope of the project and developed a working method for carrying out the study. In particular, the Group agreed to build upon information already available to the NEA, the Agency and the EC within their respective working groups and studies, and to use a questionnaire for collecting additional up-to-date comprehensive data on decommissioning policies, strategies, cost estimates and funding from OECD and non-OECD countries in order to provide a world wide review on the subject.

24 countries provided responses, out of which 10 non-OECD countries. All questionnaire responses cover policy and strategy issues with various degrees of detail. Cost data, at different level of details, were provided for 50 NPPs.

The first draft report was completed by the end of September this year and will be reviewed during the last Expert Group meeting on the 4-5 November 2002. The study is expected to be accomplished by the end of 2002.

7. TECHNICAL CO-OPERATION

Among other activities of the highest priority there is the Agency's involvement in the technical co-operation with its Member States. Within these activities there is a number of projects related to the issues of NPP Life cycle management. These Projects (implemented and current) are aimed at development of National infrastructures that would help in various ways in either implementation of different parts or development of the National programme on NPP Life Cycle management. Below there are some examples of recent projects in this area.

TC Project BRA 4/050, "Structural Integrity Analysis of Nuclear Reactor Components"

The objective of the project is to transfer the technology and knowledge on the assessment of the NPP mechanical components integrity in particular with regard to the influence of operating regimes and stressors that induce their ageing and degradation. The Project includes usual components of fellowships, scientific visits, procurement of necessary equipment and expert missions. The Project will improve National capability in monitoring ageing phenomena and NPP component life management and assessment.

TC Project MEX4/051, "Structural Integrity Programme of Reactor pressure Vessel and Internals for Laguna Verde Nuclear Power Station"

The objective of the Project is to improve the National infrastructure with the aim to design, prepare and operate test equipment and procedures for irradiated material to undertake integrity assessment tests and develop experimental ageing program, for inclusion in the life extension program of Laguna Verde Nuclear Power Plant.

The project foresees provision of the testing equipment for hot cell to be used for evaluation of irradiated RPV material properties. Corresponding training for the ININ employees and exchange of experts is being implemented within the Project.

TC Project BUL 4/010, "Upgrading the equipment for mechanical testing and microstructure investigation of Kozloduy NPP surveillance specimens"

The main goal of this Project is to extend and update the equipment of National testing laboratory for mechanical and microstructure investigations of irradiated specimens from NPP Kozloduy surveillance programme. As a result the National capabilities in RPV integrity assessment and safe operation of NPP will be improved. Within the Project new equipment for micro-structural investigation of samples will be provided to the Bulgarian counterpart.

TC Project CZE 4/009, "Evaluation of radiation damage Attenuation in WWER Reactor Pressure Vessel and Core Internals"

This project is intended to investigate the problem of neutron flux attenuation and therefore irradiation damage of RPV material through the thickness of RPV wall. This will allow for addressing a specific problem of radiation embrittlement of RPV steels and will improve the accuracy of RPV integrity assessment within the period of life cycle of the plant.

The project foresees carrying out a large-scale irradiation experiment and subsequent testing of different RPV materials of WWER-1000 type. As a result a guidance/procedure for a precise evaluation of the irradiation embrittlement evaluation through the RPV wall should be elaborate to help in more realistic RPV lifetime assessment.

TC Project ROM 4/026, "Technical Support for Chernavoda NPP Operation Management/Development of an Ageing and PLIM Programme"

The project is aimed at the improvement of NPP Life management and operational safety. It includes the development of the ageing and life management group,

TC Project HUN 4/014, "License Renewal and Plant Life Extension"

Implementation of this project will improve the infrastructure of nuclear industry and assist in creation of the regulatory basis for license renewal/extension of NPP Paks operation as well as in the development of the ageing management programme for the plant. The Project is focused on the on the use of the existing international experience and good practices.

8. SCIENTIFIC FORUM

The 5th Scientific Forum, organized during the 46th Regular Session of the IAEA General Conference, in September 2002, focused on three topical issues, one of which was Nuclear Power Plant Life Cycle Management.

From the discussions it emerged that the nuclear industry is, at present, at a crucial juncture, where it has to decide about the future of the first generation of nuclear plants, which are approaching the end of their licensed service life. At the same time, long term experience and new advances have established that it is possible to extend the life of nuclear plants beyond their initially licensed life by another 20-30 years. While some utilities and regulatory bodies have already gone ahead with license renewal or extension, many others are still exploring various possibilities concerning these processes.

The session addressed key issues, concerns and trends in the life cycle management of nuclear power plants – from construction to operation and then to decommissioning. Measures to cope with ageing plants, licence renewal, expected growth in electricity demands and the need to find sustainable long term solutions for closed or ageing nuclear facilities were presented, including examples of experience from FORATOM, Japan, the United Kingdom, the United States, the Russian Federation and Hungary.

The Scientific Forum considered that the IAEA should act as a catalyst to enable the dissemination of experience in licence renewal activities to all Member States. In addition, the IAEA should identify proven practices in licence renewal and procedures that have been demonstrated, to achieve efficient review of applications. The IAEA should produce guidance on the scope of safety and environmental reports in support of licence renewal.

9. CONCLUSIONS

For the past decade the IAEA has continued to be the focal point on information exchange and technology transfer for plant life cycle management programmes and related activities.

The comprehensive nature of the Life Cycle, from concept to activities after plant closure, have been described, including the need to ensure that the costs of all aspects of the Life Cycle are taken fully into consideration.

Agency activities, the role of the Technical Working Group on Life Management for NPPs and many other activities in support of improvements in the management of all aspects of the Life Cycle have been described.

In accordance with the advice from the Scientific Forum the IAEA will continue to act as a catalyst for the dissemination of experience and proven practices and guidance in support of licence renewal.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Power Reactors in the World, Reference Data Series No. 2, IAEA, Vienna, (April 2001).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Scientific Forum during the 46th IAEA General Conference 2002, Session 1: Nuclear Power Life Cycle Management, IAEA, Vienna, (September 2002).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Power Plant Personnel training and its Evaluation A Guidebook, IAEA, Vienna, (1996).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Cost drivers for the assessment of nuclear power plant life extension, TECDOC-1309, IAEA, September (2002).

[5] EUROPEAN COMMISSION, Green Paper Towards a European Strategy for the security of energy supply. Directorate-General for Energy and Transport, European Commission, Brussels (November2000).

[6] DAVIES, L.M., GUEORGUIEV, B., LYSSAKOV, V., Plant Life Management Programmes- Defining the milestones of NPP Life Proceedings of Plant Life Management and Plant Licence Extension, Nuclear Engineering International Conference, London, (November 2001).

[7] DAVIES, L.M., GUEORGUIEV, B., LYSSAKOV, V., Plant Life Management, Definitions, Categorisation of components and Methodology for PWRs and WWERs, pp. 1-16, Proceedings of NATO ARW on the Scientific Fundamentals for the Lifetime of Reactor Pressure Vessels, (22-25 April).

[8] DAVIES, L.M., LYSSAKOV, V., A current selected overview of PLIM and re-licensing of NPP, Proceedings of 7th International Conference, Prometey, Pushkin, St.Petersburg, (June 2002).

EURATOM COMMUNITY RESEARCH IN NPP LIFE MANAGEMENT

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Abstract

One of the main objectives of the 5th Euratom framework (FP-5) programme (1998-2002), key action “nuclear fission” is to enhance the safety of Europe’s nuclear installations and to improve the competitiveness of the European nuclear industry. Community research in this field is carried out through both "indirect actions” (organised by DG Research) and “direct actions” (carried out by DG JRC). Co-operation amongst a variety of public and private organisations is enforced, allowing the optimal utilisation of the available resources. Under the current FP-5 “reactor safety” dedicated research is conducted under the « Operational Safety of Existing Installations » component of the “nuclear fission” key action. “Plant Life and Extention Management” « PLEM » is one of the three clusters of research projects undertaken in the area (the others being “Severe Accident Management” and “Evolutionary Concepts”). Thirty three projects have been financed within the PLEM cluster with a total Community funding of approximately € 19 million and are presented here. They focus on the following three key issues: “Integrity of equipment and structures”, “on-line monitoring and maintenance” and “organisation and management of safety”. In the 6th FP (2002-2006), Euratom research is expected to continue to address items concerning safe operation of ageing plants, mitigation of severe accident consequences, as well as other safety relevant concerns raised by the stakeholders, such as decrease in dedicated human and financial resources, or spreading of a common safety culture throughout an enlarged European Union. A new challenge to Euratom research, in line with the European Research Area concept (ERA), is to reorganise itself applying the new instruments that is : Integrated Projects that will target ambitious objectives involving research, demonstration and trainning, whereas Networks of Excellence will integrate existing resources (manpower as well as installations) to form durable pools of expertise.

1. INTRODUCTION

At the end of 2000, in the EU[1], in 8 of the 15 Member States, a total of 143 reactor units were in operation with a total capacity of 123 net GWe and a total gross generation of 863 TWh (i.e. 35 % of the EU electricity generation). Despite the historical safety record of NPPs of Western design, the key players believe that research is still needed to further increase both the safety and the performances of these power plants in line with the steadily growing pressure of regulatory and market forces as well as of the public opinion.

[1] With the EU enlargement 18 NPPs will be added to the European nuclear park.

The activities of the European Commission in the field of nuclear energy are governed primarily by the Treaty establishing the European Atomic Energy Community (Euratom). The Council Decision concerning the "Fifth Framework Programme for research and training activities in the field of nuclear energy" (Euratom FP-5) was made on 22 December 1998 and covers the period 1998 to 2002 [1].

The program is implemented in two different and complementary modes. The "direct actions" run by the DG-JRC[2] (Joint Research Centre) the EC's own research centre and the "indirect actions" managed by the EC's Directorate General Research (DG-RTD).

The part of the Euratom programme implemented by DG-RTD (matter of the present paper) comprises two key actions, *controlled thermonuclear fusion* and *nuclear fission* as well as generic research on *radiological sciences,* and *support for research infrastructure.*

1.1. Key action on nuclear fission

Among the main objectives of the key action "nuclear fission" are the enhance of safety[3] of Europe's nuclear installations and the improvement of the competitiveness of the European industry. Besides the traditional technological challenges, socio-economic concerns are also taken on board, such as public acceptance and cost of the nuclear option as well as plant simplification and man-technology-organisation interaction.

The research supported under this key action focus on the following items:

- Operational safety of existing installations
- Safety of the fuel cycle
- Safety and efficiency of future systems
- Radiation protection

The research under this key action is carried out principally by laboratories and research bodies, electrical or engineering companies, the nuclear industry, nuclear regulatory authorities, small and medium-sized businesses and other public or private undertakings which can contribute their experience to attain the objectives of the programme. On the other side, the principal end-user of the results is the nuclear industry in general terms (power station designers and users, safety and regulatory authorities, etc…).

Co-operation amongst a variety of public and private organisations is thus enforced, allowing the optimal utilisation of the available resources and the enhancement of the nuclear research fabric within the Community.

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[2] information on the Joint Research Centre (JRC) which, with its seven institutes, contributes to the implementation of the framework programmes by carrying out the so called "direct" actions, can be found on the JRC homepage http//:www.jrc.org. One of the main actions of DG JRC in reactor safety research consists in operating European networks in the field of structural integrity for nuclear components, namely : the network on Ageing Materials European Strategy (AMES – homepage http://www.jrc.nl/ames), the European Network for Inspection Qualification (ENIQ– homepage http://www.jrc.nl/eniq) and the Network for Evaluating Structural Components (NESC – homepage http://nesc.jrc.nl/).

[3] safety is used here in the broadest of senses and embraces health, environmental, organisational and technological aspects. Nuclear safety is used in a similar manner and encompasses the safety of facilities, waste management and disposal and the protection of people and the environment against the effects of ionising radiation.

1.2. Operational Safety of Existing Installations

Reactor safety dedicated research is conducted under the "Operational Safety of Existing Installations" component of the "nuclear fission" key action.

The objective of the research is to provide improved or innovative tools and methods for achieving evolutionary improvements in the design and operation of existing installations improving their safety as well as the competitiveness of Europe's nuclear industry.

The following three areas of research are covered:

- Plant Life Extension and Management (**PLEM**) focusing on integrity of equipment and structures, on-line monitoring and maintenance, and on organisation of safety as well as on harmonisation of practices in EU and Central / Eastern European Countries.
- Severe Accident Management (**SAM**) including assessment of severe accident risks, and severe accident management measures.
- Evolutionary concepts (**EVOL**) including evolutionary safety concepts, and advanced fuel technologies such as high burn-up and MOX fuel.

A total of 73 multi-partner projects (with a total budget of around € 85 million and Community's contribution of approximately € 44 million) have been selected for funding by the DG Research of the European Commission (in charge with the programme implementation), in this area. Most of them are shared cost projects (the European Commission and the project partners provide each 50 % of the necessary funding), concerted actions or thematic networks. In addition training activities (fellowships, grants and support to training courses) and accompanying measures have been funded.

Plant life and extention management "PLEM" is one of the three clusters of research projects undertaken in the area of "Operational Safety of Existing Installations". The emphasis of the work programme is on the reactor's very high radioactive inventory that is perceived as a high potential danger for people and environment. The additional challenge driving the Euratom FP-5 implementation that is to share the same safety culture amongst European Union (EU), and Central and Eastern European Countries (CEEC) in the framework of the EU enlargement[4] has been reflected strongly in the PLEM projects. Whenever possible, close co-operation with Tacis and Phare projects has been encouraged.

In the following paragraphs the focus is to the PLEM cluster of projects that are presented one by one together with the most important information.

2. PLANT LIFE EXTENSION AND MANAGEMENT

Nowadays, a large number of nuclear reactors is being operating for longer than 20 years. In the EU only, a total of 65 reactor units were put in commercial operation before

[4] As the EU enlargement process is now well engaged, a number of Central and Eastern European Countries (CEECs) become of particular interest, as far as nuclear power production is concerned. The Czech Republic, Hungary, Slovakia and Slovenia, which are particularly active in FP-5, are operating all together 16 nuclear units (15 VVERs-440 and 1 PWR) with a total capacity of 7.3 net GWe providing 30 % of their electricity. Lithuania operates 2 more. FP-5 pays specific attention to plant safety assessment of Russian design reactors and to the development of new safety culture in the candidate countries in co-operation with similar activities run at the Commission, especially under the programmes of Tacis/Phare and of the EC "direct" actions run by JRC.

1980 and a total of 78 after this date. As a consequence, the nuclear industry is increasingly interested in research activities aimed at better understanding and managing ageing phenomena (i.e. changes in microstructural and mechanical properties due to irradiation, etc.). More importantly, the optimisation of the operational conditions of aged reactors (using, for example, appropriate prediction tools for evaluating the safety margins) and the decision process about plant life management (involving, for example, replacement of equipment) are becoming key issues for those in charge of plant safety and performance.

In the frame of the FP-5 of the Euratom, thirty three selected projects belonging to the cluster PLEM have been and co-financed by DG Research with a total Community funding of approximately € 19 million. The following three key issues have been identified:

- "Integrity of equipment and structures".
- "on-line monitoring and maintenance".
- "organisation and management of safety".

These projects are listed in table 1 and are thematically represented in figure 1.

1. Integrity of equipment and structures

To prevent any in-service failure of the RPV and in general of the reactor cooling system, stringent operational rules are necessary, based on deep understanding of irradiation embrittlement, of various types of fluid-structure interaction effects (corrosion processes or dynamic loading from waterhammer impacts), on accurate safety margins evaluation etc…

Understanding of irradiation embrittlement behaviour and remedial actions

Irradiation embrittlement is the subject of a number of different projects, focusing on reactor pressure vessel aging aspects.

Within **FRAME,** research is conducted to improve the assessment of the most important parameter used to measure the embrittlement conditions of the RPV. Currently this is done through indirect measurements in a rather conservative way (the so-called reference temperature methodology, which makes use of Charpy-V notch impact testing). The work focuses on the development of a method, which allows to measure directly the fracture toughness. This should result in a better and more accurate estimation of the embrittlement conditions of the RPV material.

RETROSPEC develops procedures and guidelines for retrospective dosimetry to evaluate the neutron doses induced in reactor structural materials in those cases where no or unreliable data from surveillance specimens are available (for example the older generation of VVER-440 type reactors).

The objective of **PISA** is to improve the predictability of the impact phosphorus segregation to internal grain boundaries, assisted by irradiation at elevated temperatures, can have on embrittlement in RPV steels.

As support to the transnational access to research infrastructures, the project **RENION** (homepage www.nri.cz) offers the Czech experimental LR-0 reactor to researchers to conduct experimental projects related to VVER and PWR reactor physics in order to extend their experimental databases and to validate computer codes.

Most of the MTRs will be more than 40 years old by 2010. The objective of the thematic network **FEUNMARR** is to determine the future European irradiation needs in MTRs.

Optimisation of operational conditions focusing on corrosion issues

Two projects deal with irradiation assisted corrosion cracking of austenitic steels for reactor internals. Within **INTERWELD,** the radiation induced damages that promote cracking in the heat affected zones of PWR and BWR core internal components is studied, focussing on parameters such as neutron fluence/irradiation conditions, microstructural and microchemical conditions and giving particular emphasis on the residual stresses. Further work is performed in **PRIS** to produce materials data (in particular J_{IC}, J-R, tensile properties and irradiation induced microstructural changes) for LWR internals as a function of fluence up to 70 dpa.

Within **CASTOC,** environmentally assisted corrosion of low alloy steels, for RPV, under static and cyclic conditions is studied with the aim to improve service operation and code implementation.

Prediction of structural safety margins focusing on fracture mechanics

Fracture mechanics is the subject of a number of different projects, focusing on the development of predictive tools.

The project **ADIMEW** is extending the work done in BIMET (under FP-4) in connection with the prediction of the safety margins for bimetallic welds, to confirm findings under prototypical loading conditions (scale 1 to 1 components, 300 °C and bending conditions) using fracture mechanics codes and new tools, such as advanced flaw assessment methods, more advanced J methods (i.e. calculating the crack driving force in terms of the stress intensity factor, the J-integral or the crack tip opening displacement), and the Local Approach (i.e. damage models of the micro-mechanical type using constitutive equations derived from basic material properties).

Defect assessment techniques are further improved in **VOCALIST** to better predict safety margins, in particular with respect to the constraint effect (i.e. the pattern of crack-tip stresses and strains causing plastic flow and fracture), which gives rise to an effective toughness for components higher than that measured on test specimens.

The effect of the Warm Pre-Stress (WPS) will be studied by **SMILE**. Data will be gathered, or will be created by experimental and numerical work. The objective is to propose a harmonised way of taking into account this generic phenomenon in the safety studies.

THERFAT (in connection with the recent Civaux 1 incident) proposes to review field data and perform advanced thermohydraulic flaw simulations and stress and fracture analysis. Critical elements of the procedure will be investigated by targeted verification tests. Proposals will be made for improved thermal fatigue assessment procedures, screening criteria and for establishing a European Methodology on Thermal Fatigue.

Optimisation of operational conditions focusing on water hammer loads

WAHALOADS deals with dynamic loadings during operation (generated, for example, by condensation induced waterhammers in pipes and open networks).

FLOMIX-R aims at performing a set of mixing experiments that are supported by CFD calculations. Emphasis will be on slug mixing phenomena relevant for local boron dilution scenarios, and mixing phenomena of interest for operational issues and thermal fatigue. Improved measurement techniques with enhanced resolution in time and space will be tested.
Materials testing reactors (MTR)

Concrete containment ageing

MAECENAS will create an advanced analysis tool that will allow the structural integrity of aged, reinforced, pre-stressed concrete NPP structures to be assessed in a rational manner. A generalised thermo-mechanical constitutive model for concrete will be constructed, and a safety-cost analysis tool will be developed.

The **CONMOD** project is aimed at improving the interaction of NDT and finite element calculations in order to optimise maintenance activities for concrete containments. Emphasis will be placed on the identification of possible critical defects and damage mechanisms, using laboratory-scale and full-scale experiments.

2. On-line monitoring and maintenance

In FP-4, a concerted action, AMES-NDT, was aiming at verifying to what extent non-destructive testing (NDT) techniques can be used to assess material damage. Some of the NDT techniques applied showed promising results. More development and validation work is required to evaluate the in-service applicability and the potential to support the decision-making process for failure prevention (i.e. change of operational parameters, increased inspections or replacement of components). Under 5FP, this work is continued as a full-scale shared cost action within **GRETE** focusing on applications such as irradiation damage in RPVs or thermal fatigue in piping.

In **LIRES** robust reference electrodes are under development to allow on-line monitoring of the corrosion potential (important parameter for IASCC) within the harsh operational conditions of reactors.

In **SPIQNAR,** attention is devoted to improve performance of ultrasonic inspection aimed at detecting and sizing of cracks in structural components. Specific issues addressed are development of signal processing techniques and a reliable methodology for producing synthetic defects and “virtual defect” signals to improve inspection qualification methodologies.

The **REDOS** project aims to improve dosimetry for irradiated steel and qualify methodology for radiation field parameters monitoring. Benchmarking as well as combined experimental and computational techniques will be used taking in account deep penetration and space energy dependent radiation field, complex geometry, as well as gamma irradiation.

As far as the decision process related to inspection, maintenance, operation and repair of NPPs goes, special attention is devoted within the shared-cost action **VRIMOR** to innovative support tools based on virtual reality.

NURBIM aims to develop improved procedures to identify where the highest likelihood of damage/failure is located in plant and provide quantitative measures of the associated risk. It focuses on the definition of best practice methodologies for performing risk-based analysis and establishing a set of criteria for the acceptance of risk quantities.

ENPOWER aims to develop weld repair procedures and alternative post weld treatments that minimise residual stresses and shorten repair time scales. Assessment methodology for treating defects in residual stress fields will be refined to give more accurate and informed sentencing of defects in ageing plant.

3. Organisation and management of safety

Digital instrumentation

From a technological point of view, one important challenge is the implementation of digital instrumentation and control tools (substituting the original analog systems) and the subsequent training required for the operating staff. Two projects are devoted to software modernisation:

BE-SECBS dealing with computer-based systems embedded in a nuclear installation to support I&C functions important to safety**,**

CEMSIS aiming to develop a safety justification framework for the refurbishment of systems important to safety (SIS) that is acceptable to different stakeholders (especially licensing bodies and utilities)

Organisational factors

In many studies it is recognised that organisational factors are often the root of incidents.

The main objective of **LEARNSAFE** is to create methods and tools for supporting processes of organisational learning at the NPP. This has become increasingly important for the nuclear industry in its adaptation in a changing political and economic environment, changing regulatory requirements, changing work force, changing technology and changing organisation of NPPs and power utilities. The focus will be on the management of change.

The objective of the concerted action **SPI** is to review and evaluate the application of safety performance indicators – in combination with other tools, like PSA – in order to maintain and improve safety of NPPs. It will also seek methods that can be used in a risk-informed regulatory system and environment.

More generally, the objectives of the accompanying measure **EUROSAFE** are to support the convergence of nuclear and radiological safety practices (safety culture) in Europe, while developing the idea of a European scientific and technical pool in the fields of reactor safety and radiation protection.

Emphasis to VVER reactors

The project **VERSAFE** brings together utilities from some of the Central and Eastern European Countries with the aim to produce common guidelines for the implementation of techniques for both plant modernisation and severe accident management.

The **IMPAM-VVER** project will address a safety relevant issue identified in recent studies on VVER safety. It investigates effective means and criteria for primary depressurisation during small loss of coolant accident (SBLOCA) including feed and bleed operation. The issue was raised using analytical tools but the resolution requires experimental investigation as well as specific computer code validation.

VERLIFE will create a “unified procedure for lifetime assessment of components and piping in VVER type nuclear power plants” based, in a first step, on former Soviet rules and codes. Later on, a critical analysis of possible application of some approaches used in PWR type components will be done, to incorporate into the prepared procedure as much as possible of such approaches with the aim of a harmonisation of VVER and PWR Codes and Procedures.

ATHENA, the AMES thematic network on ageing, aims, within the enlarged Europe, at reaching a consensus on important issues that have an impact on the life management of nuclear power plants. ATHENA creates a structure enhancing the collaboration between European funded R&D, national programs and TACIS/PHARE programs.

Knowledge management

It is worth presenting here some projects that, dealing with knowledge management, stretch across the three areas PLEM, SAM and EVOL.

JSRI (Joint Safety Research Index) concerning dissemination of the information and main achievements,

ENEN concerning the conservation of the nuclear knowledge and expertise by creating a European higher education space in the area of nuclear engineering. It is aimed at proposing a global network strategy, formulating proposals for best practices and performing pilot education sessions.

CERTA, a network seeking to provide a consolidated framework for the preservation of the integral system experimental data bases for reactor thermal-hydraulic safety analysis acquired in the context of research carried out by European institutional and industrial research organizations. It includes the main experimental programmes and databases relevant to reactors in operation in Europe (homepage http://lunar.jrc.it/stresaWebSite/).

Worth mentioning also are a series of Eurocourses, co-organised by DG Research and national hosting organisations, with the aim to disseminate specific research results and/or to discuss the latest achievements in a given area, thereby contributing to the objectives of “education and training”, that are becoming increasingly important. Here is the list of Eurocourses organised under FP-5 in the PLEM area:

- MASC: “Use and application of the master curve method for determining fracture toughness” (co-organised by VTT, Helsinki, 12-14/06/02)

- IPC: "Integrity of pressurised components of nuclear power plants" (co-organised by GRS, Cologne, 17-21/09/01)
- SMIRT-17 : "17th International conference on structural mechanics in reactor technology" (co-organised by Brno University of Technology, Czech republic, Prague, 18-22/08/03 – homepage http://www.teris.cz/SMiRT17).

Finally, 3 individual Marie Curie grants for post-PhD students are also part of the programme.

The mid-term achievements of most of projects, were also discussed in the **FISA 2001 Conference** [2][5] held at Luxembourg the 12-14 November 2001. More information on FP-5 projects can be found **in the Cordis web-site** www.cordis.lu/fp5-euratom/src/projects.htm.

3. RATIONALE AND OBJECTIVES FOR FUTURE RTD

Contributing by 35% to the electricity produced in the European Union, nuclear energy is an element of the debate on how to combat climate change and reduce the energy dependency of the EU. Nevertheless the European nuclear industry has these last two decades changed profile from construction / exploitation, to exploitation / service for maintenance. No major change in the number of operating plants in the EU is expected in a foreseeable future and the EU nuclear park is becoming old. Today, most of these reactors are either approaching or are already in their second half of design life.

Continued safe operation as well as optimisation of the decision making processes about plant life management are today key issues for the industrial sector and for the regulators concerned with plant safety and performance.

Despite the excellent safety performance observed within the European operating plants, mainly due to their design, conceived to limit the consequences of any accident within the confinement barriers, and the appropriate regulatory frame, the very unlikely case of a severe beyond design-basis accident, is always in the agenda of the designers and regulators.

The many public and private organisations concerned with reactor safety research are faced with the following challenges:

- keep or even increase the safety performance levels despite ageing of the NPPs,
- mitigate consequences in the worst scenario of a severe accident,
- work under economic pressure due to the globalisation of the electricity market.

Euratom research is expected to support EU policies in the fields of health, energy and the environment in the enlarged EU. It is thus expected to continue to address items concerning safe operation of ageing plants, both in EU and candidate countries, mitigation of severe accident consequences, as well as other safety relevant concerns raised by the stakeholders, such as :

- decrease in dedicated human and financial resources, or
- dissemination of a common safety culture throughout an enlarged European Union.

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[5] In this bi-annual conference, as it is the tradition now, all the EU co-financed running projects are presented by their co-ordinators to an audience of around 300 nuclear reactor safety concerned people. The **next FISA** conference will take place the 10-13 November 2003 in Luxembourg.

Euratom research is also expected to contribute towards the creation of the European Research Area (ERA) [3], [4]. The challenge is to reorganise itself applying the new instruments (Integrated Projects or Networks of Excellence) in addressing the above items.

The priorities for the FP-6 [5] as determined by the European Council in the “reactor safety” area [6] are:

i) *Innovative concepts :* to evaluate the potential of innovative concepts and develop improved and safer processes for the generation and exploitation of nuclear energy that have been identified as offering longer term benefits in terms of *safety, environmental impact, resource utilisation, proliferation resistance, or diversity of application.*

ii) *Education and training :* to combat the decline in both student numbers and teaching establishments providing the necessary competence and expertise for the continued safe use of nuclear energy (development of a more harmonised approach for education in the nuclear sciences and engineering in Europe and its implementation, including the better integration of national resources and capabilities).

iii) *Safety of existing nuclear installations :* to improve safety in existing nuclear installations in Member States and candidate countries during their remaining operational lifetimes and subsequent decommissioning, making use of the considerable knowledge and experience gained internationally from experimental and theoretical research.

In order to implement FP-6, the Commission will use various instruments. Two of them, have been designed as the main instruments for implementation of the programme in line with the ERA concept realisation. These are :

- Networks of excellence, aimed at strengthening and developing Community scientific and technological excellence by means of the integration, at European level, of research capacities currently existing or emerging at both national and regional level, and
- Integrated projects, designed to give increased impetus to the Community's competitiveness or to address major societal needs by mobilising a critical mass of research and technological development resources and competences

The Community's budgetary intervention in indirect actions is aimed at governemental and industrial research centres, universities, businesses and national or international bodies situated mainly in the Member States and the European Associated States which carry out research activities. Nevertheless, where this proves necessary to achieve the objectives of the programme, also bodies in the Newly Independent States (NIS) and international organisations may exceptionally receive Community funding.

4. CONCLUSION

Today, nuclear energy provides 35% of the electricity produced in the European Union, contributing to the debate on how to combat climate change and reduce the energy dependency of the EU.

The strategic goal of the Euratom program is to help exploit the full potential of nuclear energy in a sustainable manner, by making current technologies even safer and more economical and by exploring promising new concepts. Euratom framework research in reactor safety currently concentrates on “operational safety of existing installations”. It covers a broad range of physical phenomena and safety measures ranging from structural integrity, inspection and maintenance, to severe accident management. An overview was given here of the projects selected within FP-5 in the area of plant life management. A total of 33 multi-partner projects have been co-financed by DG Research with € 19 million. Technological requirements as well as organisational and socio-economic aspects are considered. Specific attention is given on developing a common safety culture throughout the EU Member States and the Candidate Countries.

In the future, Euratom research is expected to continue to contribute to maintain nuclear power as a sustainable option, boosting both the safety and the competitiveness of the European nuclear industry. In FP-6 particular attention is given on education and training in the nuclear field, on innovative concepts development and on the safety of the existing nuclear installations. The new challenge is the reorganisation of the Euratom research in applying the new instrument for the programme implementation contributing thus to the ERA realisation.

More information on the FP-5 implementation (DG-RTD and DG-JRC) as well as on the priorities of the 6th FP can be found in the **Cordis web-site** http://www.cordis.lu . Additional information on the European Commission research policy can be found on http://www.europa.eu.int/comm/dg12/rtdinfo.html

REFERENCES

[1] Council Decision 1999/64/Euratom of 22 December 1988 (OJ L 26, 1.2.1999, p. 34).

[2] FISA 2001 - EU research in reactor safety conference, Proceedings of the symposium on FP-5 shared-cost and concerted actions, held in Luxembourg, 12 -14 November 2001. EUR 20281 EN, OPOCE Luxembourg 2002

[3] "Towards a European Research Area", Communication of the European Commission, COM(2000)6, January 2000.

[4] "Making a reality of the European Research Area : Guidelines for EU research activities (2002-2006)", Communication of the European Commission, COM(2000)612, October 2000.

[5] Council Decision 2002/668/Euratom of 3 June 2002 (OJ L 232, 29.8.2002, p. 34)

[6] Council Decision of 30 September 2002 (to be published in OJ L)

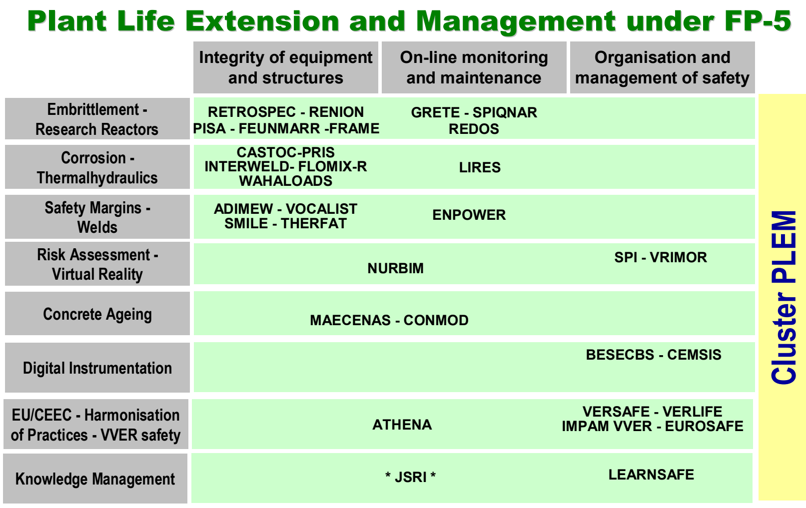

FIG. 1. Thematic representation of PLEM projects

Table 1. Projects selected in the area of plant life extension and management

Acronym	Proposal title of negotiated indirect RT action	Coordinator	Country	Project type

General

Acronym	Proposal title of negotiated indirect RT action	Coordinator	Country	Project type
JSRI	Joint Safety Research Index	Gesellschaft für Anlagen und Reaktorsicherheit (GRS) mbH	D	Concerted action

Integrity of equipment and structures

Acronym	Proposal title of negotiated indirect RT action	Coordinator	Country	Project type
FRAME	Fracture Mechanics Based Embrittlement Trend Curves for the Characterisation of Nuclear Pressure Vessel Materials	Technical Research Centre of Finland (VTT)	FIN	Shared cost
RETROSPEC	Retrospective Dosimetry Focussed on the Reaction 93Nb(n,n')	NRG	NL	Shar. cost
PISA	Phosphorus Influence on Steel Ageing	AEA Technology Plc	UK	Shar. cost
INTERWELD	Irradiation Effects on the Evolution of the Microstructure, Properties and Residual Stresses in the Heat Affected Zone of Stainless Steel Welds	Nuclear Research and Consultancy Group (NRG)	NL	Shared cost
PRIS	Properties of Irradiated Stainless Steels for Predicting Lifetime of Nuclear Power Plant Components	ABB Atom AB	S	Shared cost
CASTOC	Crack Growth Behaviour of Low Alloy Steel for Pressure Boundary Components under Transient Light Water Reactor (LWR) Operating Conditions	Staatliche Materialpruefungsanstalt (MPA Stuttgart)	D	Shared cost
ADIMEW	Assessment of Aged Piping Dissimilar Metal Weld Integrity	Electricite de France	F	Shar. cost
VOCALIST	Validation of Constraint-based Assessment Methodology in Structural Integrity	AEA Technology Plc	UK	Shared cost
SMILE	Structural Margin Improvements In Aged-embrittled RPV With Load History Effects	Electricité de France	F	Shared cost
THERFAT	Thermal Fatigue Of Piping System "Tee"- Connections	E.ON Kernkraft GmbH	D	Shar. cost
WAHALOADS	Two-Phase Flow Water Hammer Transients and Induced Loads on Materials and Structures of Nuclear Power Plants	Université Catholique de Louvain	B	Shared cost
FLOMIX-R	Fluid Mixing and Flow Distribution in the Reactor Circuit	Forschungszentrum Rossendorf E.V.	D	Shared cost
FEUNMARR	Future EU Needs in Materials Research Reactors	CEA	F	Them. Net
MAECENAS	Modelling of Ageing in Concrete Nuclear Power Plant Structures	University of Sheffield	UK	Shar. cost
CONMOD	Concrete containment management using the Finite Element technique combined with in-situ Non-Destructive Testing of conformity with respect to design and construction quality	Force Institute	DK	Shared cost
RENION	Reactor neutronic investigations on LR-0 reactor	NRI - REZ	CZ	Support Res.-Infr.

On-line monitoring and maintenance

Acronym	Proposal title of negotiated indirect RT action	Coordinator	Country	Project type
GRETE	Evaluation of NDT for Monitoring of Material Degradation	Electricité de France	F	Shar. cost
LIRES	Development of Reactor (LWR) Reference Electrodes	Belgian Nuclear Research Centre	B	Shared cost
SPIQNAR	Signal Processing and Improved Qualification for Non-Destructive Testing of Ageing Reactors	Mitsui Babcock	D	Shared cost
REDOS	Reactor Dosimetry: Accurate Determination and Benchmarking of Radiation Field Parameters, Relevant for Reactor Pressure Vessel Monitoring	Tecnatom S.A.	E	Shared cost
VRIMOR	Virtual Reality for Inspection, Maintenance,Operation, and Repair of Nuclear Power Plant	NNC Limited	UK	Shared cost
NURBIM	Nuclear Risk-based Inspection Methodology	GRS	D	Shar.cost
ENPOWER	Management of Plant Operation by Optimising Weld Repairs	Institut de Soudure	F	Shar. cost

Safety management and culture

Acronym	Proposal title of negotiated indirect RT action	Coordinator	Country	Project type
BE-SECBS	Benchmark on Safety Evaluation of Computer-Based Systems	EC – JRC - IE	NL	Shar. cost
CEMSIS	Cost Effective Modernisation of Systems Important to Safety	British Energy (Generation) Ltd.	UK	Shared cost
LearnSafe	Learning Organisations for Nuclear Safety	VTT	FI	Shar. cost
SPI	Evaluation of Alternative Approaches for Assessment of Safety Performance Indicators for Nuclear Power Plants	GRS	D	Thematic network
EUROSAFE	International approach towards convergence of technical nuclear and radiological safety practices in Europe	IRSN	FR	Accomp. measure

Emphasis on VVER reactors

Acronym	Proposal title of negotiated indirect RT action	Coordinator	Country	Project type
VERSAFE	Concerted Utility Review of VVER-440 Safety Research Needs	Fortum Engineering Ltd	FIN	Concerted
IMPAM-VVER	Improved Accident Management of VVER Nuclear Power Plants	VTT	FIN	Shar. cost
VERLIFE	Unified Procedure for Lifetime Assessment of Components and Piping in VVER NPPS	NRI - Ustav Jaderného Vyzkumu Rez A.S.	CZ	Thematic Network
ATHENA	AMES Thematic Network on Ageing	Tractebel S.A.	B	Them. Net

SAFE AGEING MANAGEMENT OF NUCLEAR POWER PLANTS - A EUROPEAN SYNTHESIS

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Abstract

Ageing of nuclear power plants means evolution of material or equipment properties on one side, and evolution of personnel skill and procedure adequacy on the other side, both of which, after a certain time, may not be compatible with the required safety provisions, or with an economic operation of the plant. Repair or replacement of components, as well as change in service conditions for a better compatibility with component reduced capabilities can be used to mitigate ageing effects. The paper summarizes the results of a study conducted in this field with the support of the European Commission. It presents:

- the synthesis of the work done under international auspices, and in the European context,
- the comparison of ageing management approaches used in several European countries with international recommendations,
- the summary of the main potential phenomena and their governing parameters, the methods of in-service ageing identification and possible mitigation methods,
- illustrative ageing management practices, taking material ageing aspects as examples.

1. INTRODUCTION

As every human being unfortunately knows, ageing is a universal phenomenon. In the particular case of nuclear power plants, ageing effects shall be closely managed for obvious safety and economical reasons. This has led the European Commission to support a synthesis work [1] providing recommendations for the development of a methodology to monitor, control and anticipate the ageing of Nuclear Islands, in order to maintain their level of safety during the whole NPP life cycle. This work includes a synthesis of the various international studies, which are presented in the first part of this paper.

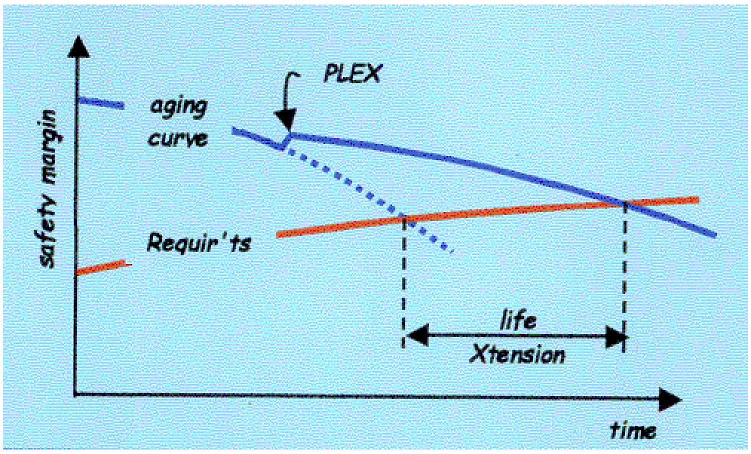

FIG. 1. Illustration of the impact of ageing management on plant life extension (PLEX).

As illustrated in figure 1, evolution of material or equipment properties may, after a certain time, not be compatible with the required safety provisions, or with an economic operation of the plant. A better ageing management may lead to an extension of the life duration during which the applicable requirements can be met. At equipment level, this may be obtained through ageing reduction, restoration of properties, reduced working conditions, or repair. At plant level, ageing management may involve planned replacement of components. In parallel, safety requirements, public acceptance, or economic conditions may evolve with time, limiting as a result the economic interest of continued operation. The plant life will then be the result of the consideration of ageing in a changing regulatory, political, technical and economic context.

Ageing of equipment is nevertheless not the only effect to be considered and ageing management shall be assessed at various levels, including strategic and organizational aspects. This constitutes the second part of this paper, where ageing management approaches used in several European countries are compared with international recommendations.

The paper summarises in a third part the main potential phenomena and their governing parameters, the methods of in-service ageing identification and possible mitigation methods, and illustrates ageing management practices, taking material ageing aspects as examples.

2. SYNTHESIS OF WORK DONE UNDER INTERNATIONAL AUSPICES

2.1. Presentation of international organizations

The synthesis prepared in the context of project [1] includes the presentation of the various European and international organizations dealing with aspects related to ageing management.

2.1.1. OECD activities

The OECD Nuclear Energy Agency (NEA) primary objective 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. Four committees develop actions relating to ageing or life management:

- the Committee on Nuclear Regulatory Activities (CNRA), set up in 1989 as a forum for the exchange of information between nuclear regulators and for the review of developments which could affect regulatory requirements,
- the Committee on the Safety of Nuclear Installations (CSNI), set up in 1973 includes scientists and engineers. Its purpose is to foster international co-operation in nuclear safety, review the state of knowledge, initiate and conduct programs based on these reviews in order to reach international consensus. Working groups have been established on Risk Assessment, Analysis and Management of Accidents, Integrity of Components Structure, and Operating Experience,
- the Nuclear Development Committee (NDC) is more focussed on the contribution of nuclear energy to overall energy demand. It has established an Expert Group on Nuclear Power Plant Life Management in 1991 to assist decision makers in evaluating the economics and politics of plant life extension by providing reports on key issues and organising workshops,
- the Nuclear Science Committee (NSC) has some relevant activities on neutron dosimetry and material degradation modelling.

2.1.2. IAEA activities

The IAEA is an autonomous intergovernmental organization founded in 1957 with the objective to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world and ensure that assistance is not used to further any military purpose.

Activities in the field of ageing and life management are carried out in the Division of Nuclear Power for life management and technological aspects and in the Division of Nuclear Installation Safety. Activities include:

- the organization of specialists meetings by the International Working Group on Life Management of Nuclear Power Plants established within the Division of Nuclear Power. Priorities are Reactor Pressure Vessel integrity, Steam Generator Management, Primary Circuit and Reactor Internals operation and integrity, Secondary Circuit, Containment/civil structures and cables considered from design, material, fabrication, monitoring, testing, inspection, degradation mechanisms, assessment and strategic, economic and administrative aspects,
- the issuing of safety standards, guidance and awareness documents, programmatic guidance reports on data collection and record keeping, management methodology, implementation and review of programs, or equipment qualification. Component specific guidelines are also prepared and ageing management review guidelines are issued for IAEA Ageing Management Assessment Teams (AMAT) assessment or utility self-assessment,
- research contract programs initiated in 1989 (Pilot studies) with the purpose to facilitate exchange of information and collaboration among international organizations engaged in ageing management and evaluation projects.

2.1.3. WANO activities

The World Association of Nuclear Operators (WANO) includes all nuclear electricity operators in the world. Its objective is to facilitate exchange of operating experience, so that members can achieve the highest possible standards of safety and reliability in operating their NPP.

The establishment of WANO followed the Chernobyl accident in 1986 which, beyond the immediate effects of the accident, had far reaching repercussions for the nuclear power industry as a whole. It caused nuclear operators to reassess safety issues and made them aware of the need for international cooperation. WANO's mission is implemented through 5 programmes on Operating Experience Information Exchange, Operator to Operator Exchange, Performance Indicators, Good practices and Peer Reviews.

2.2. Synthesis of International reports

56 reports issued by the above organizations have been identified, 35 being summarized in an appendix to report [1].

2.2.1. OECD reports

The OECD reports consider ageing issues as important for the future, with no particular concerns expressed in regard of safety. A distinction is made between physical ageing of components and structures, and other ageing concerns, such as ageing of analytical techniques and documentation, rules, standards and technology, organizations and personnel. The first domain is linked to the demonstration of the structural integrity of components and structures throughout the plant lifetime. The second domain deals more globally with the management of change.

It is considered that functional aspects are correctly managed by overall maintenance and surveillance programmes. Ageing management strategies are consequently mainly relative to passive components. Databases on experience feedback and failure data exchange have been established by the OECD Committee on the Safety of Nuclear Installations (CSNI).

2.2.2. IAEA reports

The IAEA has developed a set of useful guidance for systematic ageing management process of physical and non-physical ageing. Identification of critical components and structures and opinions on physical ageing management broadly support those of OECD Nuclear Energy Agency. Moreover, IAEA guidance documents provide help for ageing management evaluation and improvement. In addition, an International database on NPP life management has been set-up in the frame of the International Working Group on Life Management. The ageing practices applied by the contributors to study [1] have been evaluated according to guidance [2]. Data bases have been or are being set up on reactor pressure vessel materials, pipe work, steam generators, concrete containment and other components.

2.2.3. WANO data bank

WANO provides a data bank on all reported events, which is accessible to all member operators. Since the beginning of 1998, a codification system has been set up, including event categories (8 fields), event consequences (10 fields), affected systems (107 fields) or components (45 fields), status of reactor (11 fields) and activity (23 fields) performed at the time the event occurred, staff involved in or likely to learn from the event (31 fields), direct causes (55 fields) and root causes (190 fields). When there are a lot of sub fields, they can be grouped in 2 or 3 sub levels.

2.3. Needs for further research and developments

2.3.1. OECD recommendations

Needs for further research and developments have been identified by OECD initiatives in the following areas:

- Metallic components and structures: better understanding of ageing phenomena affecting the primary pressure boundary (fatigue, thermal and irradiation embrittlement, thermal shocks, corrosion, erosion, crack initiation and propagation under various environmental conditions) and tests of materials from decommissioned reactors,
- Concrete structures: in-service inspection techniques of thick sections and inaccessible areas, anchorage and sensitive parts, durability of remedial measures and repairs, loss of pre-stressing force in tension or post-tension structures, and validation of finite element analysis of degraded structures,
- Organic components: research is needed to develop more realistic end of life criteria or forecast lifetime in terms of functionality and condition monitoring methods,
- Management of changes: a technical basis for long-term operation is necessary to update plant documentation and avoid gaps in knowledge, establish a system of information retrieval to bridge gaps between today's and previous design and manufacturing standards and safety rules, determine functional requirements to allow qualification of new equipment, ensure safety of replacement of ageing instrumentation and control, manage ageing of organization by identifying and transferring specific skills, ensure transfer of knowledge and know-why through several "generations" of personnel during the expected plant lifetime, enforce international clubs of users.

2.3.2. IAEA recommendations

The results of pilot studies initiated by IAEA are expected to have application in the following areas:

- Monitoring the degradation and preventive maintenance of the selected components, including the development of criteria for the type and timing of preventive maintenance actions,
- Prediction of component performance and remaining service life under all expected service conditions,
- Future design, material selections and amendments to applicable codes, standards and regulatory requirements.

2.4. European studies

There were no specific studies made under European Commission auspices on ageing management, which was the reason for the study [1]. Nevertheless, numerous studies covering topics of interest to ageing management have been identified, and the EC framework programs presented.

The strategic goal of specific program "Research and training programme in the field of nuclear energy" of the 5th Framework Programme (FP5) [3] is to help exploit the full potential of nuclear energy by making current technologies even safer and exploring new concepts. Under key action 2 "Nuclear fission", one of the first research objectives is related to improve methods for understanding and managing the effect of ageing on equipment and structures, for on-line monitoring , and for risk informed approaches to plant modernization and inspection.

Aspects covered include integrity of equipment and structures, on-line monitoring, inspection and maintenance, organization and management of safety and networking. Achievements of the 3rd and 4th Euratom Framework Programmes have been presented in [4] and [5] together with the orientations of the 5th programme. Projects are listed in Table 1.

Table 1. FP-4 and FP-5 EC projects in the field of Plant Life Extension and management

Strategy PLEX + PLIM		FP4: INTACT FP5: VERSAFE
Integrity of equipment and structure	Embrittlement	FP4: REFEREE, RESQUE, MADAM, AMES-DOSIMETRY FP5: FRAME, PISA, RETROSPEC
	Corrosion	FP4: MODAGE, DISWEC FP5: INTERWELD, PRIS, CASTOC
	Fracture mechanics	FP4: BIMET, VORSAC FP5: ADIMEW, VOCALIST
On-line monitoring, inspection and maintenance		FP4: AMES-NDT FP5: GRETE, LIRES, IQNAR/SPICRACK
Organization and management of safety		FP4: EURIS, ISANEW, ORFA FP5: BE-SECBS, CEMSIS
Dissemination		FP4 and FP5: JSRI

Other activities organized within the European Community deal with pre-harmonization of national rules and practices. To this aim, working groups and networks are established, some of them being reorganized:

- the "Nuclear Regulator's Working Group" (NRWG) made of national Safety Authorities representatives, and the European Nuclear Installations Safety Group (ENIS-G) extended to operators, deal with safety rules including feedback of experience, risk analysis and operational safety fundamentals and culture,
- the former "Working Group on Codes and Standards" (WGCS) established within DG "ENV" did cover in-service surveillance, design and materials. It is being re-defined with new objectives within DG "TREN" including convergence between nuclear and conventional practices,
- European networks have been established, each one being dedicated to a specific aspect of the fitness for purpose of materials or techniques: the Ageing Materials Evaluation and Studies (AMES), the European Network for Inspection Qualification (ENIQ), the Network for Evaluating Steel Components (NESC) provide valuable inputs for equipment ageing assessment.

Other initiatives are organized, such as the EPERC Workshop on In-Service Inspection and Life Management of Pressure Equipment [6], which may provide valuable information for nuclear application. A more extended bibliography on ageing management is given in [1] on the basis of documents exchanged between contributors. More than 360 reports and 300 publications have been identified.

3. COMPARISON OF AGEING MANAGEMENT APPROACHES

3.1. Safety and regulatory aspects

Ageing management approaches have been considered from the regulatory point of view and from the utilities management point of view. Contributors to the study covering Belgium (Belgatom), France (EDF, IPSN and Framatome-ANP) and Spain (Unesa) have presented the practices applicable in each country, and identified a general consensus in Europe, with no limited time operating authorization, the safety being a utility responsibility under continuous surveillance by the regulatory authority. However, authorizations given by the safety authorities to the plant operator are not associated to the same process in the various countries: formal ageing management evaluations exist for quite short periods in Spain and UK, where in other countries there is a more general requirement of ability for safety demonstration at any moment (in France and Belgium).

Practical ageing management methods include:

- Periodic safety reviews (PSR), widely accepted in the international community, even if not always required by the regulation, a ten years periodicity being a common practice. This review is completed by continuous ageing management, taking into account safety and industrial anticipation needs,
- The implementation of life-time management programmes for the most important (non-replaceable) components. It could be the framework for periodically reporting ageing safety issues developments and improvements, able to prove that safety margins are maintained,
- Generic evaluations of specific important issues on the main ageing phenomena, with their own approaches and programs: examples are vessel failure assessment, stress corrosion cracking of 600 alloy, and thermal ageing of cast parts.

Safety objectives consider global probability figures for the classification of Plant conditions, but generally express the requirements applicable to components using a deterministic approach. Margins relative to component damage are specified at design stage, and defects may be accepted during service subjected to the condition that appropriate safety margins may be demonstrated.

The surveillance shall take into account the importance of the component for plant safety, the results of available studies, the return of experience obtained on similar configurations, and any doubt which may exist, in particular the degree of knowledge of the applied loading. The overall process is "risk informed", but remains based on an engineering judgement. There are strictly speaking no failure or reliability data considered in a model permitting the verification of safety objectives expressed under the form of core damage or environment radioactivity release probabilistic objectives.

3.2. Comparison of ageing management approaches with international recommendations

In the interest of obtaining comparable descriptions of practices, a standardized form of questioning was taken from the reference document for the IAEA Ageing Management Assessment Teams [2], including in part II (Guidance for Ageing Management Process (AMP) programmatic review) a list of areas to be reviewed: Table 2. Forms were completed by each contributor, except for II.4 "results" which was not considered relevant for the study. The following main conclusions were derived from this comparison.

Table 2. Areas of review (SSC = Systems, Structures and Components)

II-1	**II-2**	**II-3**	**II-4**	**II-5**
AMP STRATEGY	AMP ORGANIZATION	AMP ACTIVITIES	AMP RESULTS	AMP MONITORING
- Regulatory policy and requirements - AMP policy - International guidance - Scope of AMP	- AMP organization and programme description - Resources: (a) human (b) financial (c) tools and equipment (d) external - Provisions for understanding SSC ageing	- SSC screening method - List of SSCs - Operational procedures - Surveillance - Assessment - Maintenance - Data collection and record keeping NB: assessment includes data analysis	- Physical condition of SSC - EQ established and maintained - Performance indicators	- Self-assessment programme - Peer reviews - Comprehensive reviews - Continuous improvement process

3.2.1. Comparison in terms of strategy

Each country have defined or refer to documents on regulatory policy and requirements, in the contexts of ageing management processes described in Unesa documents in Spain, the "Continuous Operation of Belgian NPPs" project in Belgium, and the "Service Life and Ageing of Pressurized Water Power Plants" project in France. International guidance documents are considered in each case.

In no case is a life time limit pre-determined and, provided that the safety level can be justified, the decision to continue to operate is taken by the Owner on economical grounds. The safety status of the units is subject to a continuous assessment process, validating the conformity to the Safety Analysis Report. In addition, Periodic Safety Reviews (PSR) are performed and may include a Safety Reassessment. This PSR is required under the terms of the licence for Spain and Belgium. Reviews and reassessments are performed on a 10 years basis.

The principle is in every case to define a list of sensitive components that have a risk of failure due to ageing before the forecast life duration. Sensitive equipment is ranked according to the degree of importance for plant lifetime management, taking into account the importance for safety and availability. The Ageing Management Programs of all countries go beyond the IAEA objectives, which are limited to safety aspects, to cover all components which have a significant impact on reliability, replacement and cost. For these components, the programme includes the analysis of ageing phenomena, the justification of behaviour, including rate of deterioration and determination of critical defects and the assessment of maintenance practices.

3.2.2. Comparison in terms of organization

Differences have been found due to the context of the companies: Unesa and Belgatom do not have direct responsibility for the units: they are in charge of methodologies, advice, studies which should be taken up and put into force by the management of the units. EDF has to manage standardized series of units, so it imposes practices on all units of the same type from a corporate level. In terms of internal organization, the activity is managed in the three companies by specific teams: Belgatom and Unesa use project type organizations. They submit their product to their respective clients, Electrabel or the Spanish Units, for decision and implementation. EDF has established a co-ordinating group, which puts recommendations to the Steering Committee in charge of making all decisions for modifications or maintenance.

The personnel in charge of these activities, either directly involved in the specific teams or maintained in their original entities, are chosen for their skill and experience in relation to the task assigned to the team they belong to. Life management demands a wide variety of skills and no dedicated organization has been set up in the engineering teams. The specific team thus co-ordinate a large number of actions assigned to specialized teams. Execution depends on the capacity of the teams, laboratories and subcontractors. Accordingly, training for personnel is decided in observance of the internal procedures of the entity.

In addition to internal resources, each entity can decide to complement the skills needed by external support from specialized companies and experts. Exchanges between utilities and engineering companies are numerous and organized. Periodic meetings or congresses are held within the scope of the EC, IAEA and WANO discussed above.

3.2.3. Comparison in term of activities:

Methodologies

The descriptions of Belgium, French and Spanish methodologies show that IAEA recommendations are fulfilled in every case, although the practices described go beyond IAEA methodology defined in Technical Report Series N° 338 [7], which is limited to systems, structures and components (SSC) that are safety-related or whose failure can prevent performance of safety functions. However, systematic ageing assessment is only applied to the population which, based on industry and plant specific experience, are subject to failure by ageing and are not covered by appropriate maintenance for ageing mitigation and/or monitoring. The assessments are aimed at determining the severe effects of ageing, and at confirming maintenance efficiency to mitigate and/or monitor said effects, keeping components within safety margins. Lack of precise knowledge of degradation mechanisms that generate uncertainties in the initial "provisional" assessment of ageing effects require detailed analyses to clear said uncertainties.

According to Unesa methodology, initial selection includes safety-related SSC as well as those having a significant impact on availability, replacement and cost. The entire population selected is submitted to systematic ageing assessment; when ageing effects are severe, maintenance efficiency is assessed. The entire process is summarized in the component degradation data sheets and in the maintenance practice data sheets that are prepared to facilitate the systematic evaluation of existing practices with the subsequent improvement proposal.

In Belgium, safety principles are applied in everyday operation, following lessons learned from incidents, and ageing is systematically investigated during regular reviews, in order to demonstrate that the safety of the installations is guaranteed during the next decade. A specific ageing management project was created for passive safety-related components and components important for plant availability, in order to determine the conditions required to maintain, safely and economically, the unit in operation. Equipment Ageing Summaries are established, summarizing the knowledge gathered on the various topics related to SSC ageing and the situation of each NPP with respect to these ageing phenomena.

The purpose of EDF ageing assessments covers three aspects closely interconnected: technical, economic and safety. Solving technical issues requires taking into account design and manufacturing file, operating conditions, maintenance strategy, experience feedback and support from R&D programmes. Trends in "standard" and "exceptional" maintenance costs are assessed. Assessments are aimed at determining the potential life of components and structures by taking into account all available aspects. Priority is given to preventive maintenance activities implying an "anticipation approach". The objective is to identify and assess generic-type defects of components and systems. Beyond the analysis of equipment ageing, overall generic approaches are being developed, focusing on specific generic-type damage.

Operating procedures are written in compliance with design data, such as the number of transients. Provisions are added so as to limit ageing effects, such as limitation of transients profiles and occurrences, water chemistry improvement or arrangement of core for low neutron losses. Of the topics checked through the list in the AMAT guide, this one is particular in that it aims to avoid ageing. All the other topics, without exception, try to detect and correct the effects of ageing.

Surveillance programmes exist in the three countries, including periodic tests. They have been issued independently from the ageing programmes, and completed after specific reviews. For example, in France, the «basic preventive maintenance programmes» were decided subsequent to analysis of the failure rates. They are complemented by the feedback from experience on incidents (specific check point added) and progressively cost-optimized.

Maintaining equipment qualification: The life time of qualified components is determined by the hypotheses adopted in their qualification programme. Utilities focus on the problem of obtaining spare parts, as many suppliers are abandoning the production of nuclear-grade components. A process to inform the maintenance team of particular points needing care during maintenance operations, so as to avoid impairing qualification has also been identified.

3.2.4. Comparison in terms of monitoring

The programmes established by the utilities are internally assessed and presented to the Safety Authorities. Results are included in periodic (generally yearly) reviews by the owner and submitted to the Safety Authorities for approval.

4. MANAGEMENT OF AGEING PHENOMENA

4.1. Technological aspects

Management strategies of equipment ageing include:

- identification and prioritization of components of importance to safety and plant life,
- selection and identification of significant ageing mechanisms,
- ageing prediction criteria,
- surveillance and periodic testing methods,
- mitigation of ageing effects,
- maintenance programs and updating of operating parameters.

Ageing management strategies followed in the contributing countries on component degradations surveillance, maintenance practices, maintenance evaluation and issue resolution have been presented and compared [1]. Strategies were illustrated on typical ageing mechanism examples common to all participants: neutron irradiation embrittlement of reactor pressure vessels, thermal ageing of austenitic-ferritic castings, stress corrosion cracking, fatigue, reactor building ageing and electrical component and cables ageing.

4.1.1. Prioritization of components

Prioritization of components is done according to a common approach, considering the safety risks, which depend on the potential degradation phenomena and the associated risk of component failure, the potential consequences of such a failure, and the difficulty or cost for component repair or replacement. There is a general agreement on the main classification of important components obtained according to these methodologies, with non-replaceable components at the top and differences on classification details which are essentially the result of differences on utilities strategies, and the corresponding weight given to the different aspects. An example of ranking of components is given in Table 3.

Table 3. Example of Equipment ranking according to the French methodology

MARK	LIST OF COMPONENTS TAKEN INTO ACCOUNT	
1	Life Time	Reactor Pressure Vessel
2	Project	Containment
3		High diameter primary piping
4		Class 1, 2 and 3 piping
5		Steam generators
6		Pressurizer
7		Reactor Internals
8		Instrumentation and control, Converters
9		Electric cables
10		Turbine
11	Current	Generators
12	Maintenance	Primary coolant pump
13	and	Control rod system
14	Exceptional	Vessel supporting structure
15	Maintenance	Anchor bolts
16		Cooling towers
17		Handling facilities
18		In-core flux thimble
19		Bimetallic connections
20		Alloy 600 areas
21		Charging pumps, SG feed pumps
22		Nuclear heat exchangers
23		Condenser
24		Turbine heater & separators-reheaters
25		Circulation pump speed reducer
26		Feed pump
27		Emergency diesels
28		Transformers
29		Civil works for nuclear island
30		Valves

4.1.2. Identification and selection of ageing mechanisms

The identification and selection of ageing mechanisms is based in every case on systematic procedures which serves as the basis for the orientations given to the in-service surveillance and maintenance. This evaluation is further enriched by the consideration of the field experience, including the evaluation of international information. The first in-service inspection programs are based on design assumptions and engineering judgement, quickly completed by ageing experience, which generally appears obviously where not anticipated. The main identified degradation mechanisms are listed in table 4.

Methods of in-service ageing detection and identification include material properties surveillance programs, periodic inspections and testing. Examples of evaluation methods are given in Table 5.

Table 4. List of Main Degradation Mechanisms in LWRs

Potential degradation	**Metals**	**Concrete**	**Electronics**	**Polymers**
1. Irradiation	X		X	X
2. Thermal ageing	X		X	X
3. Creep		X		
4. Fatigue				
4.1 High Cycle Fatigue	X			
4.2 Low Cycle Fatigue	X			
4.3 Thermal fatigue	X			
5. Corrosion				
5.1 Corrosion without mechanical loading				
- Uniform Corrosion Attack	X	X		
- Local Corrosion Attack (Pitting, Wastage, Crevice Corrosion)	X	X	X	
- Selective Corrosion Attack (Intergranular Corrosion)	X			
5.2 Corrosion with additional mechanical loading				
- Stress Corrosion Cracking (Intergranular, Transgranular)	X	X		
- Primary Water Stress Corrosion Cracking	X			
- Hydrogen Induced Stress Corrosion Cracking	X			
- Strain Induced Corrosion Cracking	X			
- Corrosion Fatigue	X			
5.3 Flow Accelerated Corrosion (Erosion/Corrosion)	X			
5.4 Irradiation Assisted Stress Corrosion Cracking	X			
5.5 Microbiologically Influenced Corrosion	X			
6. Wear (Fretting, Abrasion, Vibration, Cavitation...)	X	X		
7. Loss of prestressing		X		
8 Environmental effects				
- Freeze-Thaw Cycling, Wetting and Drying		X		
- Chemical Attack		X		
- Oxidation				X
9 Concrete Degradation (Shrinkage, Leaching of Calcium Hydroxide, Reaction with Aggregates)		X		
10. Differential Settlement		X		
12 Oxidation				X

4.1.3. Methods of Ageing Mitigation

Methods for ageing mitigation have been grouped into three main categories:

- Changes of Structure, System and Component (SSC) design, to make them immune to given degradation mechanisms or minimise the effects of said mechanisms. Examples include: new layout of specific piping sections for fatigue prevention, chemical requirements of steels for erosion / corrosion prevention, modification of anti-vibration devices for wear prevention, or optimization of ventilation and cooling systems for electrical component thermal ageing.
- Change/recovery of material characteristics. Examples include: protective coating of material or stress relief (stress corrosion cracking), material annealing (neutron embrittlement), or various injections for concrete structure protection.
- Change of operating parameters if the degradation has not risen to a level necessitating repair or replacement. This include pressure, temperature and fluid chemistry control, reduction of transients, neutron flux reduction, etc.

Table 5. Examples of Ageing evaluation and mitigation methods

Ageing mechanisms	**Evaluation methods**	**Mitigation methods**
Irradiation embrittlement	- Mechanical test samples taken according to given schedule - Hardness tests - Annihilation of positrons	- Change/recovery of material characteristics: annealing - Change of operating parameters: neutron flux management
Thermal ageing	- Test samples taken from actual components and expertise of replaced components - Hardness and micro-hardness tests - Small angles neutron diffraction (DNPA) - Thermo-electric power (PTE) - Simulation in laboratory conditions	- Change/recovery of materials characteristics: annealing, part replacement
Initiation of fatigue cracks or Corrosion (inter-granular, stress-corrosion) cracks	Non-destructive examinations: - Surface examinations: visual or remote (TV) liquid penetrant or magnetic particle - Volumetric examinations: Eddy current radiography, gammagraphy, ultrasonic Acoustic emission Leak detection	- Change/recovery of materials characteristics, improved geometry, surface treatment producing compressive stresses - Change of operating parameters: reduction of oxygen content, reduction of temperature gradients, redesign of layout and supports
Concrete degradation Differential settlement	- Visual inspection - Topographical levelling - Fissurometers - Leaktightness tests	- Change/recovery of materials characteristics: injection of resins or cement, covering
Electrical cables	- Monitoring of mechanical and electrical characteristics - Qualification under representative conditions - Visual inspection	- Change of operating parameters: reduction of operating current, placing cables remote from heat sources, modification of cable layout

Comparisons of practices given for various typical ageing mechanisms can only be summarized in the present paper. They show the global consistency of the approaches chosen in the three countries, though the applicable provisions may differ on particular points, such as criteria or prescribed safety margins. The general tendency is for Belgium and Spain to refer as far as possible to the US methodology and criteria, where France tends to develop its own methodology due to the standardization of its NPPs and specific regulatory requirements.

4.2. Other aspects

Aspects such as industrial obsolescence and human, organization and knowledge ageing shall also be identified. It was even said that "a bad reactor with a good team is usually better than a good reactor with a bad team". The management of equipment ageing includes firstly the availability of on-site test equipment with which such deterioration can be diagnosed and methods and (material and human) means intended for repairing or replacing the parts whose serviceability has been impaired. This also implies sustaining an industrial fabric that has the means of providing spare parts to secure replacement or repair of degraded components. Obsolescence – the potential unavailability of spares due to the evolution of the market or/and the technical progress – could affect the capability of equipment to perform their required functions.

In addition to the ageing-induced material changes, other evolutions concern developments in techniques and documents since the creation of the power plant (design methods and computer codes, non-destructive testing, industrial codes and standards...), developments in technology (example of analogue systems being replaced by digital systems), developments in safety regulations and standards and acceptable dose limits for staff in terms of radiation protection, developments in information relating to accidents. Other changes, associated with the industrial sector or society, shall also be investigated, such as safeguarding of skills and knowledge following the retirement of experts involved in the design, construction, commissioning and operation of power stations, training of new staff (including non-specialized staff and plant suppliers), preservation of scientific infrastructure (R&D services), development of safety culture over long periods, adaptation of organization models, and work methods.

The human organism as a whole is subject to a still very poorly understood ageing process which degrades the performance of many human functions. However this impairment is generally compensated for by a greater mastery in handling inferences and by a more extensive experience. The operation life-time contributing to the ageing process of components and structures provokes a relevant impact in terms of staff ageing and the need of retirements and substitutions by less experienced people. Criticality of this point increases if, at the same time, the ageing plant needs higher efforts in equipment surveillance and maintenance. A different problem is time relaxation of worker habits and behaviours. Work practice trends to reduce the adaptation capability to new situations, new instruments, or design modifications, which could require different operators training.

The above-mentioned problems are covered by the requirements concerning the qualification and training of the NPPs operators. Preservation of the knowledge related to ageing, their contributing factors and their evolution, as well as the mitigation, control and monitoring methods and techniques shall be part of the new staff training. Rotations of working places to reduce stress and increase interest, ensuring a sufficient overlap between new workers and elder ones, use of retired workers as a source of experience and advice, and involvement of experts from other industries, which could help to integrate new knowledge are also considered. A great importance has to be given to technical Information to support staff actuation. A well structured and updated information covering all the facets of the significant degradations is the best guarantee to support a precise condition evaluation keeping the historical data available for the NPPs staff along their service life.

5. EXAMPLE OF MATERIAL AGEING MANAGEMENT PROGRAMS

Material ageing management programs are given to illustrate the above strategies. Taking into account the physical understanding of these phenomena, zones of components concerned and potential effects are identified, prediction laws established, surveillance programs prepared, criteria applied and mitigation methods proposed.

Taking the resistance to fast fracture as example, certain materials used in pressurized water reactors may evolve over the life of the plant, because of thermal ageing or neutron irradiation.. This may influence the allowable defect size / loading combinations. Consequently, such an evolution shall be managed during operation on the following three aspects:

- Follow of material properties evolution,
- Follow and updating of applied transients, following return of experience,
- Periodic non-destructive examination of potential defects, with qualification objectives based on "critical" defect sizes evaluated taking into account the applied transients, the material properties in aged condition, and the applicable safety margins.

Examples of tools available for ageing evaluation and surveillance are given in [12]. Non destructive examination qualification practices can be derived from the ENIQ consensus defined at the European level. For the practical application of the regulatory objectives relative to fast fracture analysis, methods are provided in industrial codes, such as RSE-M [10] or ASME XI [11]. In the following paragraphs additional details are given concerning two particular material ageing phenomena: the irradiation and thermal ageing embrittlements.

5.1. Reactor Pressure Vessel Irradiation embrittlement

Neutron irradiation effect on RPV material is well known [8]. Ageing effect is essentially a function of neutron fluence, chemical composition of material (in particular copper and phosphorus residual elements), and temperature. The consequence for the reactor vessel material is an increase in the yield stress, which raises the ductile-brittle transition temperature and thus entails an increased risk of fast fracture. Comparisons of practices lead to the following conclusions:

5.1.1. Zones covered

In all cases, the so-called "Belt Line" of the RPV is considered for fast fracture evaluation. This zone may be defined with respect to the neutron fluence likely to be calculated or measured, or on the expected effect (for example, a minimum shift in transition temperature). The second definition was first used in France, which did lead to necessary evolutions and in some cases difficulties when the knowledge on ageing behaviour evolved. This led to an evolution towards an irradiated zone defined as the zone subjected to a fluence greater than a given threshold.

5.1.2. Effects covered

The effects of neutron fluence include a shift of the brittle-ductile transition temperature, as well as a decrease of the upper shelf toughness. All practices consider both effects, the second one being more explicitly considered where fast fracture prevention is not limited to non-ductile (brittle) failure risk, but also includes ductile tearing evaluation, as it is the case in France.

5.1.3. Prediction

Prediction of neutron fluence effects is obtained using US-NRC Regulatory Guide 1.99, rev.1 (included in RCC-M [9] Appendix ZG) in France and rev.2 in Belgium (for old plants) and Spain. French practice do not use rev.2 because this formula is based on in-service surveillance of US materials not representative of materials with relatively low residuals. This is the reason why more precise in-service evaluations are done using FIM and FIS formulas, based on data obtained on representative materials and codified in the RSE-M code [10]. Belgium rules recognise this fact by using FIS and FIM prediction formulas for more recent units.

5.1.4. Surveillance programs

Surveillance programs are defined according to rules generally based on US practices. Specimens representative of base materials, welds and heat-affected zones taken from the belt line region are included in capsules subjected to a neutron flux higher than the one applied to the RPV wall itself, leading to a given "anticipation factor" which shall meet two considerations: guarantee the representativity of ageing measured on test samples, leading to a maximum value for this factor, and provide a sufficient anticipation, leading to a minimum value for this factor.

These capsules are withdrawn according to a pre-determined schedule in each country. There is a global consistency of practices on this topic. The embrittlement is generally monitored by measuring the transition temperature shift at a conventional Charpy V toughness level of 41 J. The material toughness curve is then shifted by the same value.

There is nevertheless a general tendency towards including fast fracture specimens in the surveillance program, allowing a direct determination of the applicable toughness curve of the material, and a reduction of uncertainties, or instrumenting Charpy tests. Evaluation of Chooz A issue in France and of Doel 1&2 in Belgium particularly demonstrates the pessimism of ageing predictions based on the above conventional approach. In addition, reserve capsules are provided with the objective to cover life extension needs, in particular in France and in Belgium.

5.1.5. Criteria

For normal operation, two sets of criteria are used:

- the US approach used in Belgium and Spain, where a reference conventional defect is considered and a safety margin is applied on the part of the stress intensity factor which is due to primary loads (mainly due to pressure),
- and the French approach, which considers more realistic defects and margins applied to not only the primary part, but also to the secondary part. France also considers criteria for brittle failure prevention and ductile tearing. Partial safety coefficients are also codified in the RSE-M, based on semi-probabilistic considerations.

Screening criteria to limit the risk of vessel failure due to pressurized thermal shocks are applied in countries whose practices are based on US approach. These criteria, based on values of RT_{NDT} at end-of-life, were prepared, but not accepted in France.

Demonstrations shall cover the next ten-years interval, evaluations covering the whole life are prepared, but have no regulatory value. There are no real limitations of the reactor vessel life, but the renewal of pressure tests and associated examinations have to be conducted at a pressure higher than the design temperature (1.20 Pc in France), with a minimum margin between test temperature and transition temperature, and test temperature limitation imposed for personnel safety reasons. This may lead to vessels which cannot be used for the reason they cannot be tested safely.

5.1.6. Mitigation

In all cases, mitigation of neutron fluence effects may be obtained through a reduction of fluence by increasing at design stage the vessel diameter, and by using at design and/or in-service stages a core loading pattern with a lower leakage. Another possibility is a vessel annealing, which is identified, but not used by the utilities of the three countries.

The reduction of fast fracture risk may also be obtained through a reduction of thermal shocks and an increase of material temperature during an accident, which can be obtained by an increase of the temperature of the safety injection water (used in France and Belgium). In the case of the Chooz A issue, the criterion retained was a maximum difference between the transition temperature of the vessel material and the temperature of the safety injection system.

Fast fracture risk is also reduced through qualified non-destructive examinations demonstrating the absence of defects or providing a good detection of potential defects in the most severe zone, i.e. the "first 30 mm" of the vessel shell, as retained in France.

5.2. Thermal ageing

The conditions required for thermal ageing include having an "unstable" material in which the atoms are able to rearrange themselves by diffusion [8]. The rate of diffusion depends on temperature and material structure. Cast duplex austenitic-ferritic stainless steels have a dual phase structure consisting of delta-ferrite (typically 10 to 25 %) in an austenitic matrix. The ferrite is there to forestall any risk of hot cracking on casting and to raise the tensile properties up to the level of forged or rolled (wrought) austenitic steels. It also improves the resistance to different forms of corrosion. During ageing, decomposition of the chromium-iron solid solution hardens the ferrite and makes it susceptible to cleavage fracture, even at a temperature as high as 300°C. Research carried out made it possible to establish embrittlement prediction formulas employing the chemical composition, the ferrite content and the operating temperature and duration.

Ageing management practices are presented below according to the same organization than for RPV embrittlement:

5.2.1. Zones covered

The surveillance concerns mostly the reactor coolant pump casing and the cast elbows. Specific files have been established in France for "hot" and "cold" elbows, ageing effects being strongly dependent on temperature.

5.2.2. Effects covered

Thermal ageing leads to a reduction in material toughness (Charpy values and J tearing resistance properties). Practices in the various countries have a similar coverage.

5.2.3. Prediction

Prediction of ageing effects have been established, based on temperature, chemical composition and equivalent ferrite content and on models validated on experiments. French and Belgium provisions are very similar. French studies are based in particular on a very important program including tests and calculations, justified by the standardization of plants, which has no equivalent in the other countries. Samples taken on site or on replaced components are used to validate the predictions.

5.2.4. Surveillance programs

Belgium and Spanish practices follow US practices. Belgium uses Code Case N-481 allowing the replacement of volumetric examinations, subject to material and fast fracture evaluation conditions, for the primary pump casing. French practice is described in the RSE-M, with additional specific provisions coming from the conclusions of the generic studies for the most affected components.

5.2.5. Criteria

There are no specific criteria in Belgium and Spain on thermal ageing of cast stainless steel products. A leak before break evaluation may be acceptable. In France, margins against fast fracture have to be demonstrated, taking into account the defects likely to affect the components. Cast elbows replacement need shall more closely be evaluated when the neighbouring steam generator is replaced and the equivalent chromium content is larger than 23.5%. A LBB evaluation was only used for a complementary technical evaluation according to a "defence-in-depth" approach, but not as a safety criterion. Progresses in safety evaluations are made in France on the initiation risks associated to real defects compared to analyses assumptions.

5.2.6. Mitigation

There are no practical mitigation possibilities, except temperature decrease and material change at design stage, which would reduce material ageing. The accident hypotheses remaining conventional, the only practical possibility to reduce the fast fracture risk is by reducing the defect probability through non destructive examinations, and by replacing the most aged components taking the opportunity of the steam generator replacement.

6. CONCLUSIONS AND RECOMMENDATIONS

A safe control of ageing of nuclear power plants constitutes an important concern for every plant Owner and Safety Authority. Since 20 years, this topic has been the subject of numerous studies which have led in particular the utilities to establish programs or projects specifically dedicated to the management of ageing of systems, structures and components.

The study shows that nuclear power plants are subjected in the participating countries to appropriate ageing monitoring programs or projects, which fulfil IAEA international recommendations on this topic. Contributors agree on the conclusions of the international groups and consider that their recommendations are appropriate. The study also shows the general consistency of the objectives in the different countries, which go beyond safety aspects, with differences in organizational approaches resulting from different industrial and regulatory contexts. There is a general agreement on the service life of the plant which is the life during which the plant may be operated safely and economically. Safety is then a necessary, but not a sufficient condition.

In addition to physical ageing management programs dealing with the gradual evolution of equipment, other factors are considered, such as evolution of market and industrial context, organization changes and conservation of human knowledge. In practice, approaches are different for non replaceable major components, for which in-depth analyses are conducted on life duration, and components subjected to preventive maintenance programmes. Aspects leading to slow evolutions are evaluated during periodic safety reassessments. Those related to more quick changes (in particular for active components) are managed on a continuous basis.

The "ageing" of LWR nuclear power plants can consequently be put under control. Through monitoring and a clear understanding of the different degradation mechanisms, which are common phenomena in an industrial facility, the operator is able to anticipate, via an appropriate programme, the necessary measures in order to operate its plants safely and economically. Considering the existing practices, the results obtained and the research programmes currently running, the ageing management field may be considered as being adequately covered. Nevertheless, one can suggest to particularly emphasize the efforts on the following topics:

- Concerning regulatory aspects, operators have to compete in a de-regulated market and are subjected to national requirements which may be different on some aspects. It is therefore recommended that exchanges between regulators be maintained at the European and International levels on safety standards, particularly on risk-informed aspects, qualification methods and acceptance criteria. Lastly, a certain stability of the safety requirements is recommended to permit full benefits from the analysis of return of experience and plant improvements.
- Concerning management aspects, issuing of periodic syntheses is recommended, which can play a role in the general context of public acceptance. In addition to IAEA reports recommendations concerning design information, the weight shall be put on detailed operating conditions book-keeping covering the entire equipment life. Ageing management shall not be restricted to the oldest plants, where improvements or mitigation measures may have a limited impact. It shall be considered as early as possible in the daily operation of the plant.
- Concerning prediction of ageing, the essential points where progresses are recommended concern the evaluation of degradation kinetics, the description of uncertainties, the improvement of the understanding of local loads and their variations as a function of operating modes, a better evaluation of the effect of fabrication process and knowledge on surface behaviour, the use of expertise conducted on decommissioned plants or replaced components and the integration of additional material, to cover life extension or evolutions in ageing evaluation methods.
- Co-operation between plant operators shall be maintained, in order to identify any signs of ageing precursors and their possible treatment, and relations shall be extended to other industrial sectors, in particular on civil works, advanced tools for monitoring and surveillance for a better evaluation of existing margins and a better allocation of available resources, continuing improvements of traditional non-destructive examination methods, in particular in concrete structures, and industrialization of methods likely to detect ageing risks before occurrence of irreversible damage.
- Recommendations on ageing mitigation include investigating repair methods used in other industrial sectors, in order to share qualification costs, anticipate replacement

where appropriate, through feasibility studies including consideration of potential impacts on safety, and more generally support exchanges between users and manufacturers in order to better appreciate the relations between operation conditions and ageing. Additional work on criteria for possible equivalence (for example for I&C) to cover technical obsolescence is also recommended.

- As far as a significant part of recorded incidents are due to human factors, it is essential to invest on human management aspects, and to the adaptation of safety culture and procedures, taking into account the evolution of people experience, of tools and of the general industrial culture, which gives more and more importance to economical aspects. Safe management during long periods implying to demonstrate and update compliance with the initial safety requirements, it is also necessary to maintain, not only the basic requirements, but also a good understanding of what is behind.

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REFERENCES

[1] Safe management of NPP ageing in European Union. Framatome Final Report EE/S 00.0430. CEC-ENV Contract N°98/872/3040/DEB/C2, September 2000.

[2] Services Series N°4 "AMAT Guidelines. Reference document for the IAEA ageing management assessment teams", March 1999.

[3] Research and training programme in the field of nuclear energy. 5-th Euratom Framework Program 1998 to 2002. OJ L 26, 1.2.1999.

[4] VAN GOETHEM, G. "EU Research in Reactor Safety: Achievements of the 4-th and Prospects for the 5-th Framework Programme" Introductory lecture, Eurocourse-99 on Reactor Safety Advanced Nuclear Reactor Design and Safety, GRS Garching/Munich, 17-21 May 1999.

[5] LEMAITRE, P., VAN GOETHEM, G. "EU research activities in the field of plant life extension and management: Achievements of FP-4 and prospects for FP-5", ASME PVP Conference, Seattle, July 23-27, 2000.

[6] EPERC (European Pressure Equipment Research Council) Workshop on In-Service Inspection and Life Management of Pressure Equipment. MPA Stuttgart, 5 October 2001.

[7] Technical Reports Series No 338 "Methodology for the Management of Ageing of Nuclear Power Plant Components Important to Safety", 1992.

[8] MEYZAUD, Y., SOULAT, P. "In-service aging of pressurized water reactor steam supply system materials, RGN International Edition – Vol.A – July 1996.

[9] RCC-M "Design and Construction Rules for Mechanical components of PWR nuclear islands", 2000 Edition, Afcen, Paris.

[10] RSE-M "In Service Inspection Rules for Mechanical components of PWR nuclear islands", 1997 Edition plus 2000 addendum, Afcen, Paris.

[11] ASME Boiler and Pressure Vessel Code. 2001 Edition, ASME, New-York.

[12] GRANDEMANGE, J.M., MALAVAL, A. "Understanding Aging Phenomena. Plant Life Management". PBNC 2002 Conference, Chenzen, 0ctober 21-25 2002.

AN INTEGRATED VIEW ON PLANT LIFE MANAGEMENT: EC-JRC-IE PROJECTS AND PROGRAMMES

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Abstract:

Nuclear electricity accounts for more than one third of the total EU needs and the life distribution of the operating plants is such that in 2005 more than 70% of the plants will have passed the 20- year lifetime and almost 30% the 30-year age limit. In total, more than 100 LWRs in the EU & Associated States are producing more than 100 GWe. In view of the secure and safe supply of electricity, ageing of power plant is becoming of increasing concern in the EU as well as in the rest of the world. In fact, with increasing operational plant life approaching gradually to the original design life, ageing issues are arising. The list of issues is very long and research and development effort is required in order understand and tackle them properly. Such issues includes: RPV embrittlement, core shrouds, upper and lower head cracking, sticking control rods, cracking in control rod drive mechanism (CRDM) and upper head penetrations, reactor coolant piping issues, steam generator degradation, electric cable ageing, concrete ageing. The JRC-IE with the AMES, NESC, ENIQ, NET projects and the information and competence with the Tacis programmes is playing a key role on the plant life management issues in the various fields of: irradiation embritllement issues, material degradation assessment and monitoring, surveillance, structural integrity, fracture mechanics studies, qualification of in service inspection methodologies for crack detection and, recently, risks aspects. In addition a new network, AMALIA, is in preparation in order to tackle the issues related to core internals and Irradiation Assisted Stress Corrosion Cracking (IASCC) phenomena. Aspects of managing and mitigation of accidents; neutron beams methods for residual stress measurements and other issues related to directives on pressure equipment are also dealt with. The expertise on the subject is also available in the Institute in the Unit supporting the TACIS programs. This papers gives an overview of the EC programmes and JRC-IE activities in particular dealing with the issues of plant life management of nuclear power plants and summarises the obtained key results and plans for future integrated projects. Examples of key results booked so far are presented and discussed. Such results, obtained in the frame of the several networks, shared cost activities and JRC institutional projects, represent the basis and give directions for the future PLIM efforts. Results include gained for example knowledge on irradiation embrittlement, material degradation assessment and monitoring, surveillance data analysis, structural integrity and fracture mechanics and qualification of in service inspection methodologies. In order to tackle the PLIM task an integration effort in JRC is ongoing aiming at defining a PLIM project, SAFELIFE,

integrating the efforts done in the past and the competencies of the various European networks. SAFELIFE will start in 2003 and will last for the whole 6th Framework Programme.

An integrated viewof ageing mechanisms and optimisation of R&D activities for PLIM in view of: Safe & Secure Supply (3S) is clearly promoted together with the prevention, performance and risk informed based approach. The SAFELIFE 6th Framework Programme JRC project is presented.

THE JRC INSTITUTE FOR ENERGY

Nuclear electricity accounts for more than one third of the total EU electricity production. In total, more than 100 LWRs in Europe are producing about 100 GW_e. The life distribution of the operating plants is such that in 2005 more than 70% of these will have passed the 20-year lifetime and almost 30% the 30-year age limit. Given the need for safe & secure supply of electricity, the ageing of nuclear power plant is of increasing concern in the EU as well as in the rest of the world. In fact, as more and more plants approach their original design life, more new ageing issues arise. These need to be understood and R&D effort is required in order to tackle properly such issues which are summarised in the following:

Plant Life Management Issues

A complete list of NPP life management issues is not easy to draw but general agreement is anyhow achieved on several major items:

- RPV embrittlement; effects of elements like Cu, P, Ni, Mn, etc.
- Reactor Internals shroud cracking, bolts cracking
- Thermal fatigue in piping
- Dissimilar metal welds integrity
- Steam generator degradation cracking
- Electric cable and concrete structure ageing

The JRC-IE is promoting an integrated approach to R&D activities on generic issues for plant life management of ageing nuclear power plants, required to support European needs for sustainability and for Safe & Secure Supply (3S) of electrical power. To meet this challenge the European Commission's Joint Research Centre proposes to form a Network of Excellence focussed on structural integrity for plant life management of key components, covering the main R&D disciplines involved and considering all nuclear power plants designs both western and eastern. This is intended to provide a long-term structure capable of addressing generic issues relating to accident prevention, plant performance & risk informed methods and harness the efforts of the leading European R&D. The JRC-IE approach is based on the successfully established European Networks AMES, NESC, ENIQ, also new ones such as NET, AMALIA and future ones, operated by the JRC, as well as the main related activities supported by the Commission (Shared Cost Actions, Thematic Networks and Concerted Actions).

ONGOING ACTIVITIES AT JRC

During the EC's FP4 and FP5, JRC-IE has been co-ordinating and managing the major European Networks, namely AMES, NESC, ENIQ and also NET and their research activities in the area of nuclear plant life management.

- AMES deals with irradiation embrittlement issues, material degradation assessment and monitoring, surveillance, etc. [1,2,3 and 4]

- NESC is dedicated to structural integrity, with principal focus on fracture assessment. [5,6 and 7]
- ENIQ tackles risk-informed approaches to plant life management and the qualification of in service inspection methodologies for defects detection and sizing [8]
- NET deals with the application of neutron based testing for residual stress analyses, defectology, ageing and irradiation damage, repair issues etc. [9, 10 and 11]
- Relevant expertise on the subject is also available in the Institute for Energy in the Unit supporting the TACIS/PHARE programs (SENUF activity).
- AMALIA is the newest network dealing with internals issues and IASCC

The Networks have joined together all key players in Europe in the various fields of ageing, structural integrity and inspection related to plant life management.
A large number of partnership projects in the area have been designed, promoted and carried out in the last ten years. These European Networks have made a substantial contribution to the total R & D effort in the field of nuclear plant life management through a large number of Shared Cost Actions (SCA), Thematic Networks (TN), Concerted Actions (CA) and Contribution-In-Kind projects in the area.

EXAMPLE OF RESULTS

As example of results obtained, some results of AMES in the field of n-embrittlement are shown in the following. In figure 1, for example, the results obtained from the model alloy project are summarized. The results, showing the influence of Nickel on irradiation embritllement in synergy with Copper and Phosphorus, are important element for improved embrittlement prediction formulas for high Nickel steels RPVs like VVER-1000.

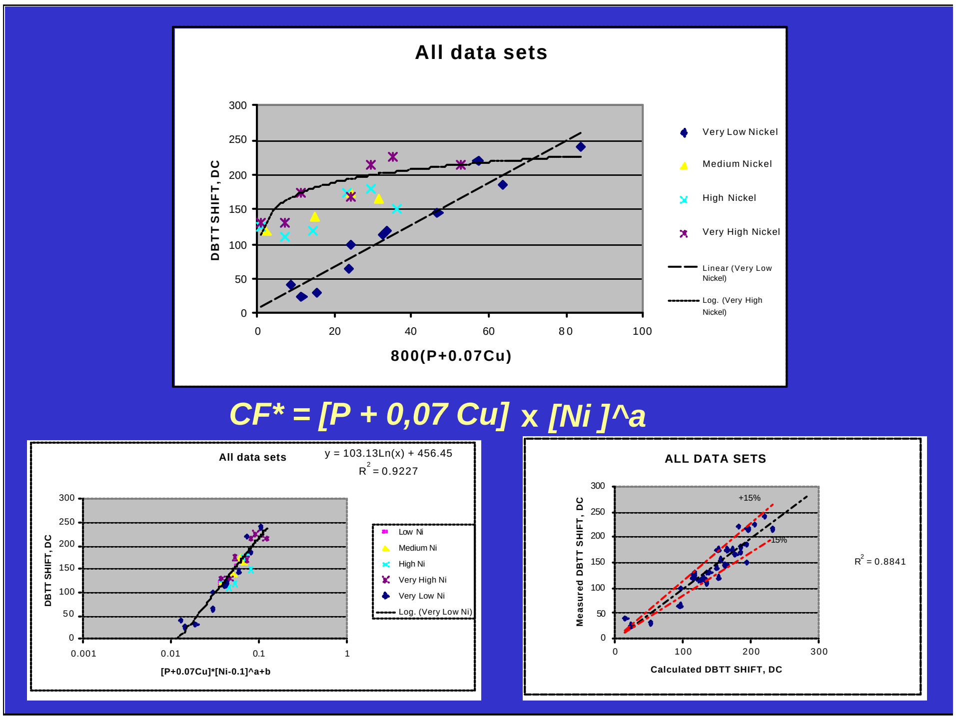

Figure 1 – Model Alloys project; summary of results

An other example of results obtained in co-operation with RRC-KI is the understanding of the embrittlement kinetic of high Nickel commercial steels by evaluation VVER-1000 surveillance data and other relevant research data. The role of Nickel for a broad fluence range is very clear and plays a major role in embrittlement.

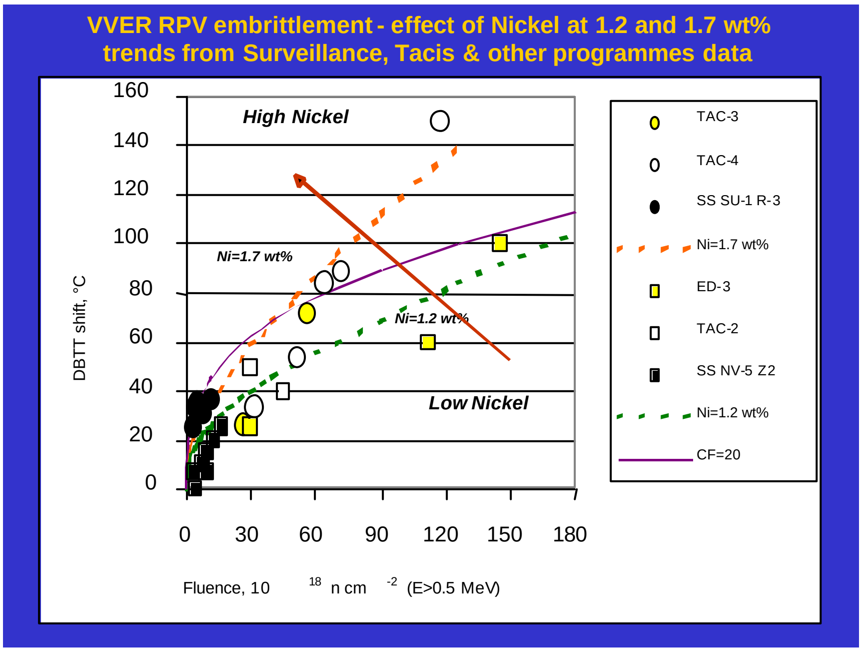

Figure 2 – Model Alloys project; summary of results

In addition the role on Manganese in different VVER-1000 welds is targeted for addition irradiation embrittlement. Welds with Mn contents in excess of ~0.9 wt%, in combination with Ni, are showing increased sensitivity to embrittlement, see Figure 3.

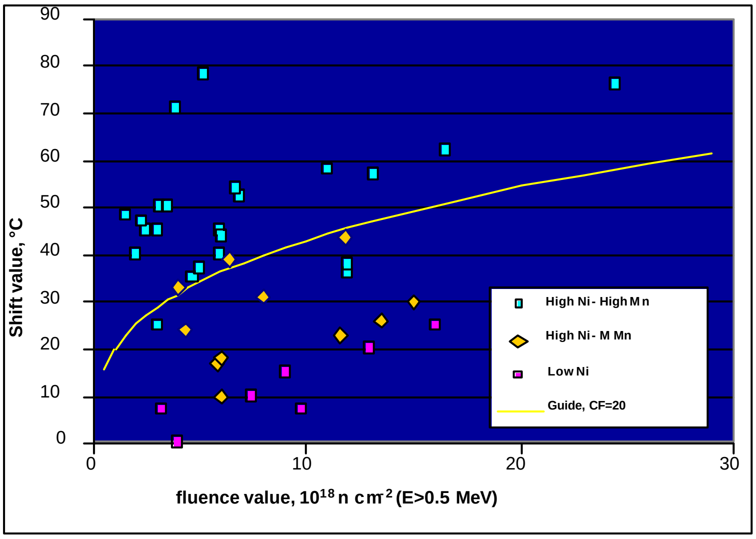

Figure 3 – Role of Mn in embrittlement of high Ni welds (VVER-1000 surveillance data)

JRC PROJECT ON PLIM IN FP6: SAFELIFE

For the 6th Framework Programme, JRC-IE is launching the SAFELIFE project. SAFELIFE will integrate the various existing and designed JRC tasks and networks on PLIM issues. PLIM is seen as a multi-disciplinary issues requiring a subsequent application of the various involved disciplines which range from crack detection, residual stresses evolution, integrity assessment, material properties degradation, IASCC, maintenance practices, etc., see Figure 4.

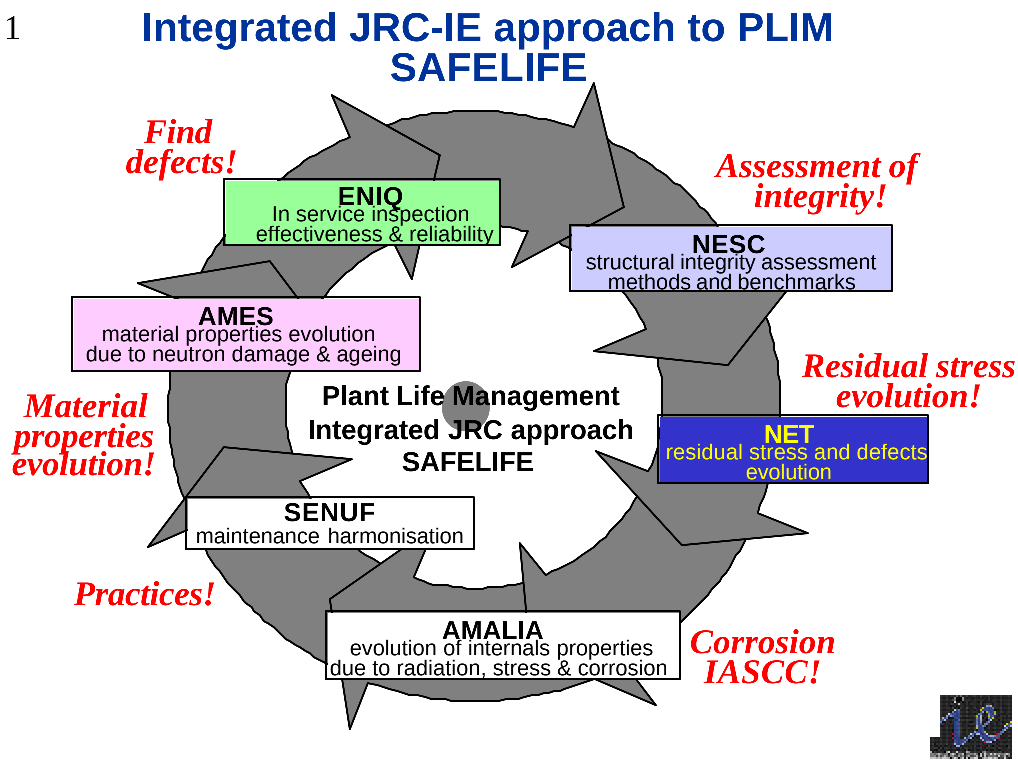

Figure 4 – SAFELIFE JRC project and its structure

The major objectives of the SAFELIFE projects are planned to be as follows:

- **Best practices promotion, development & dissemination:**
  - assessing neutron embrittlement (Ni, Mn, Cu, P, etc.)
  - annealing & re-embrittlement of WWER (PRIMAVERA)
  - primary component integrity assessment
    (Master Curve, dissimilar welds, large-scale benchmark tests, etc.)
  - Inspection Qualification, Risk Based approach, residual life & PLEX
- •**Direct support to other Directorate Generals (DG)**
  - DG-RTD (knowledge man., diss., etc.), DG-TREN (e.g. WGCS)
  - DGs working on Tacis/Phare/Enlargement
- **Enhance competencies/facilities in nuclear safety**
  - support HFR, irradiation technology, rigs & n-beams
  - integrated JRC labs, Reference materials, characterisation, R-R, etc.
  - modeling, TEM, SANS, RPV cladding, etc.

- **Create and implement the Training & Mobility Action**
  - Grant Holders, Visiting Scientists, Eurocourses, WEB, etc.
- **Further exploit Networking (tool for Integration & Dissemination)**
  - completion of GRETE, REDOS, PISA, FRAME, ATHENA, NURBIM, SPIQNAR, ENPOWER, INTERWELD, ADIMEW, THERFAT ......
  - develop new ones coherent to SAFELIFE, using FP6 tools
  - promote & operate the Network of Excellence/I.P. on PLIM
  - IAEA co-operation (CRP, R-R, etc)

To give an example of co-operation with the IAEA, a VVER-1000 reference steels action is recently being jointly undertaken by JRC-IAEA and the component is being stored and characterized at the moment at JRC-IE premises for future IAEA actions, see Figure 5.

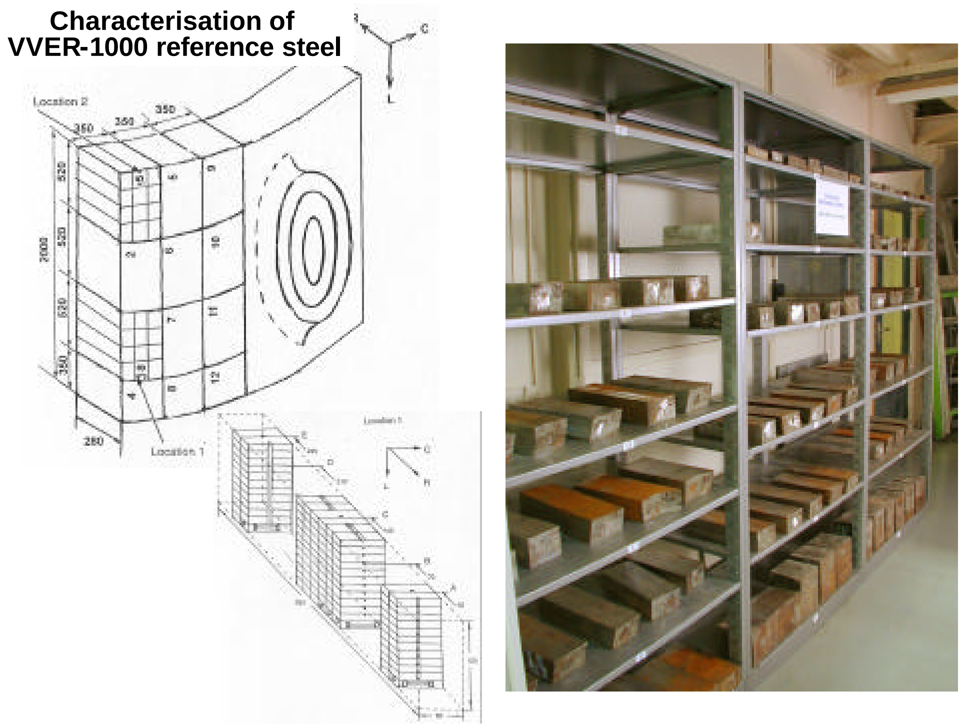

Figure 5 – VVER-1000 Reference steel, joint IAEA-JRC action

CONCLUSIONS

In view of the secure and safe supply of electricity, ageing of nuclear power plant, is an important issue for the EU.

The JRC-IE with the AMES, NESC, ENIQ, NET projects and Networks, complemented by the the information and competence developed within the Tacis programmes, is playing a key role on the plant life management area in the various fields of: irradiation embrittlement issues, material degradation assessment and monitoring, surveillance, structural integrity, fracture mechanics studies, qualification of in service inspection methodologies for crack detection and recently risks aspects. Important results have been booked so far and additional results will become available from the large number of projects still carried out at present.

Examples of key results booked so far have been presented and discussed. Results include gained knowledge on irradiation embrittlement, material degradation assessment and monitoring, surveillance data analysis, etc.

For the 6^{th} FP, JRC-IE is launching the SAFELIFE project on PLIM; integrating the efforts done in the past and the competencies of the various European Networks (AMES, NESC, ENIQ, NET, AMALIA and SENUF).

SAFELIFE will start in 2003 and is planned to last for the whole 6^{th} Framework Programme.

REFERENCES

[1] "The European Network AMES" , F. Sevini, L.Debarberis and M.Davies; 19-23 June 2000, 6th International Conference on Materials Issues in Design, Manufacturing and Operation of NPP Equipment, St. Petersburg, Russia

[2] "Achievements, Open Issues And Developments On VVER RPV Irradiation Embrittlement Assessment & AMES European Network Strategy", A.Kryukov, L.Debarberis, IAEA Specialist Meeting on Irradiation Embrittlement, Madrid 1999, PLIM+PLEX

[3] "Results of Embrittlement Studies on Model Alloys, Reference Steels and RPV materials at JRC-IAM"; L.Debarberis, F.Sevini, B.Acosta and S.Pirfo, A.Kryukov, D.Erak, M.Valo; June 2002, 7th International Conference on Materials Issues in Design, Manufacturing and Operation of NPP Equipment, St. Petersburg, Russia

[4] "The effect of Nickel, Phosphorus and Copper in Irradiation Embrittlement of RPV Steel Model Alloys", L.Debarberis, NATO International Workshop on RP Vessel Embrittlement, September 2000, Varna, Bulgaria

[5] "Overview of NESC-IV cruciform specimen test results, ASME PVP Conference, August 2002, Vancouver, Canada"; N.Taylor, R.Bass

[6] "Networking for Improved Structural Component Integrity Assessment Benchmarks, Proc. ASME PVP 2001 Conference, Atlanta, 22-26 July 2001"; N.Taylor, R.Hurst, D.McGarry and R.Rintama

[7] "Evaluating the NESC-I test and the integrated approach to structural integrity assessment"; Journal of Pressure Vessel & Piping, 78 (2001) 213-224; R.Hurst et al.

[8] "A Status Report on the Achievements of ENIQ"; Proc. 3^{rd} International Conference on NDE in Relation to Structural Integrity of Nuclear and Pressurised Components; Seville, 14-16 November 2001; U.Sandberg, A.Eriksson, T.Seldis, L.Fabbri and J.Whittle

[9] "European Standardization Activities on Residual Stress Analysis by Neutron Diffraction", A.G. Youtsos, C. Ohms, Applied Physics A, 75, (2002) in press

[10] "Towards the development of an International Technical Specification for NDT – Test method for determining residual stress by neutron diffraction", A.G. Youtsos, presented at the CEN/TC 138 meeting in Barcelona, Spain, 19-20 June 2002

[11] NET – European Network on Neutron Techniques Standardization for Structural Integrity, A.G. Youtsos, presented at the 8th ECNDT, Barcelona, Spain, June 17- 21 2002

Session 5

CURRENT NATIONAL APPROACHES TO PLIM

Sub-Session 5.2

Chairpersons

G. Sgarz
Germany

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Bulgaria

THE LEGISLATIVE BASIS AND THE STATE OF THE PLANT LIFE MANAGEMENT PROGRAMMES ON CZECH NPP'S

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Abstract

The information about Country's NPP profile and Design Basis is presented including the history of four WWER 440/213 reactor units in operation of nuclear power plant at Dukovany and two other WWER 1000/320 reactor units under commissioning at Temelin. The application of basic measures of the ageing management and plant life management on both Nuclear Power Plants is described and discussed in relation to the Design basis of components and existing and planned regulatory requirements and guidelines.

1. THE INFORMATION ABOUT COUNTRY'S NPP AGE PROFILE

Actually there are four WWER 440/213 reactor units in operation on nuclear power plant at Dukovany and two other WWER 1000/320 reactor units under commissioning at Temelin.

The design of the CEZ / Dukovany NPP was based on the Agreement between governments of former USSR and CSSR from 30th April 1970 and the contract between ATOMENERGOEXPORT as supplier and Skodaexport Co. as the Customer in this international Contract. As designer of Primary Circuit and nuclear part the LOTEP company from former USSR was contracted.

For the Dukovany NPP Project the standardised design VVER 440/213 U was implemented. As -build Design was completed at 1980 by Energoprojekt Praha. LOTEP was main designer of primary part, Energoprojekt Praha was main designer of secondary part and supervisor of entire construction.

The Design life of this type of unit was determined as 30 years. Design lifetime was specified in the Technical Design (LOTEP - USSR) 30 years. On the base of design lifetime all economic calculation were carried out by designer

Design lifetime stated by manufacturers in passports of equipment is mainly 30 years, reactor pressure vessel 40 years. Determination of Design lifetime by manufactures was based on assumptions of operational condition, expected changes with regard to total design life.

As a part of the design condition the one cold leg instantaneous rupture was determined as the maximal design accident for this project.

2. NPP DUKOVANY – HISTORY

	1st unit	2nd unit	3rd unit	4th unit
Start of construction	1/1979	1/1979	3/1979	3/1979
Start of operation	11/1985	9/1986	6/1987	1/1988

The Temelin Project was authorised in 1980 as art of the Czechoslovak power industry's development programme. This included provision for four Soviet-design VVER 1000 nuclear power plants, of which three were to be located in Czech Republic territory. Construction of the Temelin NPP started in 1986, other three plants projects were discontinued.

The plant was originally designed with four power units, but the post –revolution Federal Government scaled this down to two units in 1990. The Czech Government finally approved completion in March 1993 following a realistic appraisal of the Republic's future energy needs. Investment of the project construction has been undertaken by CEZ (Czech Power Works) who also operates the plant. A number of improvements of the original Soviet design were made following a detailed evaluation by the International Atomic Energy Agency. The most important recommendations were to replace the Russian Instrumentation and Control System by more modern western type and to improve the design of the reactor core by simplifying the nuclear fuel and the fuel assemblies' arrangement in the reactor. These design modifications were contracted and realised by Westinghouse Co.

All aforementioned units were/are build to Soviet standards. Most pieces of equipment in both NPPs were manufactured in Czech factories (e.g. reactor pressure vessels, steam generators and pressurisers, piping, etc.) and some are gradually modified or replaced by fully qualified.

3. THE REGULATORY SYSTEM, INCLUDING REGULATOR-LICENSEE INTERFACE/RELATIONSHIP

The State Office for Nuclear Safety (SUJB) is the independent Regulatory Body, established by Government to supervise the fulfillment of Czech legislation Requirements on the field of Nuclear Safety and Radiation Protection. In relation to lifetime Management of Nuclear Installations, following statements of the Atomic Act} The Law No.: 18/1997 Sb.:

(2) The Office

a) executes the state supervision over nuclear safety, nuclear items, physical protection, radiation protection and emergency preparedness in premises of nuclear installation or ionizing radiation source workplaces and inspects adherence to a fulfilment of obligations as per this Law,
b) approves documentation, programs, lists, limits, conditions, physical protection approach, emergency rules and, subject to a discussion of interfaces to external emergency plans with a relevant District Council, internal emergency plans and their modifications,
c) determines conditions, requirements, limits, limiting values and values for exemption from the jurisdiction of this Law,
d) makes decisions ensuring handling of nuclear items or radioactive waste, in case their owner or an generator proceeds out of accord with this Law and does not eliminate arisen conditions.

Instruments, applied to enforce the legislative requirements set down the SÚJB authority to require the inspected person to remedy the situation, to perform technical inspections, revisions or the functional ability tests, to withdraw the special professional competence authorisation issued and to impose penalties for violating obligations established in the Atomic Act. The SÚJB is an independent central State administration body with its own budget approved by the parliament.

Besides siting, construction and operation a SÚJB licence is prerequisite also for activities such as individual stages of nuclear installation commissioning, restart of a nuclear reactor to criticality following a fuel reload, for reconstruction or other changes affecting nuclear safety, the SÚJB issues authorisations for activities performed by licensee.

4. EXISTING AND PLANNED REGULATORY REQUIREMENTS AND GUIDELINES FOR NPP AGEING MANAGEMENT ARE, MAINLY:

- **Regulatory approach/concept for managing safety aspects of NPP ageing**

There the ageing management is considered by the SÚJB as a constituent part of NPP´s Quality Assurance system.

The Atomic Act does not contain the term Design Lifetime. Instead of that requires proofs that the high level of nuclear safety will be ensured for the time of the license duration.

Basically the Long Term Operation Concept was accepted and applied for all Czech NPP's. The term Design Lifetime has only an informative value. The Atomic Act states that operator has full responsibility during the whole time of operation. The Decree 214/97, following the Atomic Act, defines that for SSC important to nuclear safety the operator is responsible for their ageing surveillance using state of art tools.

Approval for permanent use of the plant in accordance with Civil structure law 1988-90 has unlimited time. Approval for operation is given by SUJB and it is time limited.

Operational permission is given on the base of Atomic Act by SUJB. Atomic Act states general condition and requirements that operator has to fulfil before the request for operation license or renewing of the license is submitted to SUJB.

Relevant regulations, licence conditions and regulatory guides for NPP ageing management

- Act. 18/1997 Coll. on Peaceful Utilisation of Nuclear Energy and Ionizing Radiation
- SUJB Regulation - 195/1999 Coll. on Requirements on Nuclear Installations for Assurance of Nuclear Safety
- SUJB Regulation - 106/1998 Coll. on Providing Nuclear Safety and Radiation Protection of Nuclear Installations at Their Commissioning and Operation
- SUJB Regulation - 214/1997 Coll. (Ref. [1]) on Quality Assurance in Activities Related to the Utilisation of Nuclear Energy

Special SUJB Requirements

- Instructions and Recommendations for Qualification of WWER 440/213 Nuclear Power Plants Equipment Important to Safety (SUJB - Bezpečnost jaderných zařízení - Praha - December 1998).

Existing and planned regulatory inspection assessment and enforcement practices relating to ageing management

SUJB stated basic condition for Long Term Operation:

- all System, Structures and Components (SSC) important to safety have to be qualified for normal operational conditions and conditions during and after the accident through all operational lifetime,
- using a state of art tools, the results of ageing monitoring of System, Structures and Components important to safety have to be submitted,
- all previous conditions of SUJB related to ageing must be solved
- all indicated deviations from recommended nuclear safety standards of higher importance must be solved.

It is expected that Periodic Safety Review (PSR) will be part of the licensing process .

The scope of PSR will come out from IAEA guide 50-SG-O12 (DS307) or decree of SUJB (if issued). Timing and scope must be discussed with SUJB in advance.

General regulatory review practices relating to ageing management

SÚJB assesses the level of nuclear safety also in the course of the so-called ”licensing” procedure to issue licenses for activities identified in the Atomic Act. Moreover, for NPP Dukovany SÚJB also assesses its level of nuclear safety assurance and the safety margins for individual SSCs within the following activities:

- assessment of the periodically submitted Operational Safety Report (requirements for its submittal are specified in the respective SÚJB resolution),
- evaluation of SSC qualification program,
- evaluation of the in-service inspections program,
- evaluation of the program for the enhancement of nuclear installations safety,
- evaluation of feedback from the operational experience and implementation of the latest scientific knowledge and technology.

As a part of periodical assessment of the Operational Safety Analysis Report Modifications and generally in 10 years period issued SAR containing the Periodic Safety Review results, the Main Components Lifetime Monitoring Program results are assessed.

In the lifetime management process, it is the most important to identify, which degradation mechanisms damages the corresponding area of the material in decisive way, to create a mathematical description of the material damage process and subsequently the evaluation of material damage trends and thus the determination of the residual lifetime.

For the Temelin NPP the Ageing Assessment Program is in beginning stage, utilising the modern Design Basis, wide application of monitoring systems and also large experience from Dukovany NPP.

Examples of regulatory review practices relating to ageing.

In both Nuclear Power Plants the equipment qualification keeping programs are assessed by SUJB.

Dukovany NPP Equipment Qualification Programs are completed in subprograms :

- Plant Design Basis
- Operational Conditions of SSC
- Equipment Environmental Qualification Program
- Modification Program
- Ageing Program
- In-service Inspections and Maintenance Evaluation Program
- Safety Performance Evaluation Program
- Other Plant Experience Application Program

Actual problems with this program status is a nonconformity between:

- List of SSC covered by environmental qualification
- List of SSC covered by Technical Specification (LaP)
- List of SSC covered by Safety Classification (Selected Equipment)

In the Dukovany NPP, diagnostic software DIALIFE is created, performing the calculation of the equipment residual lifetime using verified calculation programs based on information from the technological information systems {production units, diagnostics, chemistry, special measurements, SCORPIO in-core measurement system, non-destructive testing results, and material properties database}. In this way lifetime monitoring of the following equipment is performed in the Dukovany NPP:

- Steam generator
- Pressurizer
- Main coolant pump
- Main (coolant) circulation pipeline
- Emergency safety featuers pipelines
- Evaluation of fatigue damage of the reactor pressure vessel.

For the monitoring in DIALIFE pipes of safety class 1 and 2, including the compensation pipe, are prepared.

Great attention is paid to the radiation embrittlement of the reactor pressure vessel. The applied project ”Complementary testimony sampling program ” removes, among others, the inaccuracies of the descending reduction and interpretation of data about neutron fluence, and enables to monitor the lifetime during the whole reactor pressure vessel lifetime in accordance with the legislation and international standards. Gradually the development of the Reactor Pressure Vessel integrity assessment procedures is accepted by SUJB from original

Temperature Approach to Fracture Mechanics Approach and actually the Master Curve Approach.

Erosion/corrosion of piping systems made of carbon steel is monitored in the Dukovany NPP by the CHECKWORKS programs on following systems:

- feed water injection to steam generator
- live steam aftercooling
- feeding tank emptying into condenser
- condensate injection to feeding tank
- pipe 6,7 and 8 of the turboset extraction
- heating of condensate by steam from the high-pressure re-heater
- discharge from condensate pumps pipe to low-pressure re-heaters 1,2,3,4,5

Temelin NPP Ageing Management Program is in initial state, but there are specific advantages:

- complete Design Basis
- wide operational monitoring program on computerised base
- SAR according US NRC Standards
- sharing of experience with Dukovany NPP

The automatic system of operational diagnostic means for primary circuit are

- Digital Impact Monitoring System (DMIMS)
- Reactor Vibration Monitoring System (RVMS)
- Material Fatigue Evaluation System (MAFES)
- Rotating Equipment Monitoring System for the Main Coolant Pumps (RECOP)
- Control Rod Drive Monitoring System (CRDMS)
- Leak Monitoring of Pipes System (LEMOP)

For the monitoring in DIALIFE pipes of safety class 1 and 2, including the compensation pipe, are prepared.

5. CONCLUSIONS

The application of basic measures of the ageing management and plant life management on both Nuclear Power Plants is assessed by licensee and compared with the Design basis of components in main fields as are:

- lifetime assessment and safety margin specification
- solution of deviations, based on relevant international standards and operational experience
- fulfilment of the SÚJB (Regulatory body) requirements
- modification and modernisation programmes.

In actual situation there are no safety indications for apprehensions that the operators goals for longer time operation, based on economical calculations will not be fulfilled.

REFERENCES

[1] Guides and Recommendations for the Important to Safety Equipment Qualification of WWER 440/213 type Nuclear Power Plants, State Office for Nuclear Safety (SÚJB), 1998

[2] Safe Management of the Operating Lifetimes of Nuclear Power Plants, INSAG 14,IAEA, Vienna, 1999

[3] CEZ/EDU Regulation No.: 16/02-„Lifetime Management“, Dukovany 3/2002

[4] CEZ/ETE Regulation No.: 20.05-„Rules for Ageing Management“ Temelin (in preparation)

FRENCH NUCLEAR PLANT LIFE MANAGEMENT PROGRAM REACTOR PRESSURE VESSEL INTEGRITY ASSESSMENT

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Abstract

The process used by French utility, concerning the Reactor Pressure Vessel assessment, applied on 58 PWR NPPs 3 loops and 4 loops Reactor, involves the verification of the integrity of the component by finite element mechanical studies, in all conditions of loading in relation with RT**NDT** (Reference Nil Ductility Transition Temperature), and considering all of parameters. This approach, is based on mechanical safety studies, to demonstrate the absence of risk of failure by brittle fracture. For these mechanical studies two major input data are necessary:

1. the fluence distribution and the Fluence values and RT**NDT** during the lifetime in operation for each NPPs,
2. the thermal-hydraulic and mechanical evaluation and temperature distribution values in the downcomer.

The main results must show significant margins

The major tasks and expertises engaged by EDF were:

- more precise assessment of the fluence and neutronic calculations,
- better knowledge of the vessel material properties, including the effect of radiation,
- the NDE inspection program based on the inspection of the vessel wall, with a special NDE tool to inspect the area in subcladding zone,

1. INTRODUCTION

EDF's Pressurized Water Reactors, 54 units, currently in operation supplied 82 % of the domestic power generation in 1999 (in particular it should be noted that these nuclear power plants provide about important part of all electricity generation).

The first nuclear PWR reactor (900 MWe) built by EDF was put into service twenty years ago. The 900 and 1300 MWe units in operation have an average age of 17 years for the thirty-four 900 MWe units and 12 years for the twenty 1300 MWe units.

These production power plants are now almost 50 % amortized, and represent a technical and financial investment of overriding importance, both for EDF and for France.

Over the last twenty years, the choice of the nuclear program has formed the basis for a wide margin of competitiveness in electricity production, guaranteed energy independence and reduced releases of carbon dioxide into the atmosphere. Therefore control over existing nuclear power plants is of overriding importance and involves the following three objectives :

- maintain current operating performances (safety, availability, costs, security, environment) in the long term, and possibly improve on some aspects ;
- wherever possible, operate the units throughout their design lifetime, in other words 40 years, and even more if possible ;
- consolidate acceptance of nuclear energy, largely based on earning the public's trust.

The financial stakes associated with maintaining the lifetime of nuclear power stations are very high; thus, if their lifetime is shortened by about ten years, dismantling and renewal would be brought forward which would increase their costs by several tens of billions of Euros.

Furthermore, every extra year of operation of a 900 MWe unit should save about 80 million Euros per year on financial charges that would be necessary for a new investment, provided that maintenance costs do not become excessive.

2. APPROACH APPLIED FOR RPV ASSESSMENT

The Reactor Pressure Vessel (RPV) of all of EDF's three and four loop units is an important component to determine the life management of NPPs, it is a major task to justify the RPV and components for all conditions of loading, particularly in emergency and faulted situations, and will remain in operation for lifetime beyond 40 years.

The nuclear boiler was designed based on an engineering lifetime of 40 years, as used in safety reports. However, from a regulatory point of view, French law does not specify any time limit on the operating lifetime of the installations, within the construction authorization decree.

Therefore, EDF should do everything in its power to justify to Safety Authorities and to the public that this lifetime can be reached and, if possible, extended in order to make maximum use of investments already made.

The lifetime of a nuclear power plant may be affected by three main factors :

- normal wear of its components and systems - sometimes called aging - that depends particularly on their age, operating conditions and maintenance conditions applied to them;
- the safety level, that must consistently comply with the safety requirements applicable to the power stations at all times, and which could change as a function of new regulations;
- competitiveness, which must remain satisfactory compared with the competitiveness of other production means.

Within this context, obtaining a lifetime for at least 40 years depends primarily on control of the safety level which must comply with safety requirements at all times, and secondly on all technical and industrial aspects in order to operate units with a good level of competitiveness and safety [1].

Technically, the objective is to derive an understanding of aging effects, and to define and then implement appropriate safeguards to maintain the performance level of units at their current level.

Considering these elements, the general strategy is based on the following two points:

- the ten-year safety reassessment process for each ten years period,
- the implementation of two structured programs in order to make sure that all technical and industrial actions necessary to achieve a lifetime of at least 40 years are actually implemented :

Obviously, there are ongoing discussions with the Safety Authorities about this process and these programs.

3. PROGRAM ON REACTOR PRESSURE VESSEL

Since 1987, EDF has been setting up a "Lifetime" program in order to understand and anticipate aging problems.

This program reviews everything that can have an impact on the lifetime of installations, considering purely technical aspects related to equipment, stet industrial, economic and regulatory aspects.

The "Lifetime" program also identifies progress necessary to improve knowledge about aging phenomena and to support Research and Development actions in order to make a better link between operating conditions and maintenance conditions for components and their lifetime.

This program makes a distinction between [2]:

- Two non-replaceable components : the reactor vessel and confinement containments,
- Other components replaceable included in exceptional maintenance strategies with the objective of reaching a lifetime of at least 40 years in the maintenance policy, primary circuit components for example.

A new development for life evaluation and monitoring is presented for the RPV combined with the vessel in-service inspection on the core zone and mechanical integrity assessment analysis of the vessel.

In fact what input data are necessary to introduce in thermal-hydraulical calculation and mechanical analysis.

4. TOOLS AND METHODS FOR FLUENCE AND RTNDT ASSESSMENT COMBINED WITH RPV IN-SERVICE INSPECTION PROGRAM AND MECHANICAL ANALYSES

The methodology applied for RPV assessment is based on:

1st - The evaluation of RT**NDT** (Reference Nil Ductility Transtion Temperature) at different period (Lifetime period),

2nd - The evaluation of RT**NDT** after mechanical analyse computation and thermohydraulic calculation.

The diagram (Table 1) shows the methodology used for vessel by vessel evaluation.

5. EVALUATION OF FLUENCE AND RTNDT

Neutron-induced embrittlement of vessel materials is measured by the increase in nil ductility transition reference temperature (RT_{NDT}). This is an internationally established practice, since RT_{NDT} is known to be closely linked to fracture toughness. A shift in RT_{NDT} also reflects the envelope shift observed in fracture toughness curves as a result of aging.

The basis for demonstrating an absence of brittle fracture risk is predicting maximum shift in end-of-life RT_{NDT} under radiation, using formulas that allow for the chemical composition of the affected materials.

The fluence value, evaluated at regular intervals, is required to follow and to determine the $RT_{\mathbf{NDT}}$ of the vessel during the lifetime. The diagram presents the relation between the fluence level and the $\Delta RT_{\mathbf{NDT}}$ asses In this paper one example of a presentation of new development for life evaluation and monitoring is presented for the RPV combined with the vessel in-service inspection on the core zone and other part of the vessel,

In fact what input data are necessary to introduce in thermohydraulical calculation and mechanical analysis.

6. TOOLS AND METHODS FOR FLUENCE AND RTNDT ASSESSMENT COMBINED WITH RPV IN-SERVICE INSPECTION PROGRAM AND MECHANICAL ANALYSES

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6.1. Evaluation of the Fluence and RTNDT

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In fact for each vessel with the fluence parameter, the RT**NDT** value at different period of life is evaluated by the relation :

Initial RT_{NDT} + $\Delta RT_{\mathbf{NDT}}$ = $RT_{\mathbf{NDT}}$ **value** (**40-50 years**) (1)

6.2. Fluence Management of Reactor Vessels

Regarding the diagram of the table 1, the first analysis allows the fluence and specific RT**NDT** vessel by vessel obtained after radiation damage evaluation (methodology [**A]**).

The fluence management (reduction of fluence under -15%) is input data to determine ΔRT**NDT** with prediction formulas and specific chemical composition of the vessel.

Use of core physics codes in conjunction with database of recently determined cross sections validated through extensive R & D, enables accurate assessment of the fluence experienced by all zones of the vessel. This information is vital to ensuring coherent calculation of RT_{NDT} shift.

Not to be content with such measures, EDF has now undertaken to optimise fuel loading patterns to reducing fluence, while allowing for two increasingly widespread trends: deployment of MOX fuel in hybrid management schemes (in CP1/CP2 900 MWe reactors series) and fuel cycle expansion to an average 18 months (in 1300 MWe units).

Parallel to these developments, fuel loading patterns are being optimised in terms of maximum fluence undergone by reactor vessels. Depending on the loading pattern and barring any operating contingencies, a decrease of 15 to 40 % in maximum flux can be achieved with respect to re-evaluated design values, for the standard fuel arrangement (Fig. 1). With a special neutronic calculation code " EFLUVE ", each year the fluence level is determined before the arrangement of the refuelling plan unit per unit for all of 3 loop and 4 loop reactors [3].

6.3. Monitoring Radiation Impact on RT_{NDT} *Shift*

Right from the start of 900 and 1300 MWe plant operations, a comprehensive program of radiation surveillance was devised to collect data on RT_{NDT} evolution in each vessel.

This program is based on withdrawal, at predetermined outage intervals, of four radiation specimen coupons previously placed inside the vessel (Fig. 2), at positions exposed to greater neutron flux than the vessel wall itself. Subsequent analysis of the Charpy test samples taken from these capsules yields quantitative data on the impact of neutron-induced embrittlement and enables suitable anticipation of irradiated vessel mechanical "health" at the 40-years service milestone [3].

CP1 and CP2 REACTOR VESSEL SURVEILLANCE PROGRAM - Current Status				
	Capsule 1 U	**Capsule 1 V**	**Capsule 3 Z**	**Capsule 4 Y**
Residence Time in Vessel (years)	**4**	**7**	**9**	**12**
Residence Time in Vessel (years)	**11.2**	**19.5**	**28.1**	**39.1**

With this radiation surveillance program, EDF decided to reintroduce two reserve radiation specimen capsules in the vessel. EDF has decided to begin taking measures to evaluate vessel fluence and RT_{NDT} beyond age 40 years (service life goal for its 900 MWe PWRs: 50 years).

This will entail putting back into vessels the two extra capsules supplied by FRAMATOME the constructor of the Reactors for the radiation surveillance program.

These "reserve" specimens, procured from the same sample ring as the original capsules, will be reinserted after removal of the latter, during upcoming outages. thirteen years of radiation will then be required to obtain a "picture" of vessel fluence for a 40-years duration.

6.4. Presentation of the Organization of the Irradiation Surveillance Program- Results of the Expertises on Specimen Capsules

As it has already been the subject of several presentations[1] , we shall restrict ourselves here to the main characteristics of the organization of the French surveillance program.

Largely inspired by American regulations, practice fulfills the requirements of the French order and its circular dated 26th February 1974. Each reactor vessel is the subject of monitoring of the materials in its core zone, defined as being susceptible to marked embrittlement towards the end of its design life, in this case 40 years. Generally, this core zone is made up of two shells and the associated weld for all 900, 1300 and 1450 MWe reactors.

On the basis that the risk of brittle fracture is highest at the end of design life, a base metal, a weld and a heat affected zone are selected for each vessel. These materials, installed in the reactor by means of capsules at locations characterized both in terms of temperature and neutronic conditions, allow the evolution of the mechanical characteristics of the component to be monitored.

Each capsule contains nuclear instrumentation, based on fissile and activation dosimeters, and thermal instrumentation, based on low melting point alloys, so as to determine accurately the conditions of its stay in the reactor and to optimize the use of the samples for mechanical tests.

Analysis is mainly based on impact strength tests; by hypothesis, the shift ΔTcv, resulting from the comparison of the impact strength curves pre and post irradiation, is taken to be representative of that of the codified toughness curve for this type of material and the temperature scale of which is index linked to the RTNDT of the material. This shift is measured at the 56 Joule level or at 0.9 mm of lateral expansion.

The main objective of this analysis is to verify the conservatism of the hypotheses adopted at the design stage as regards aging, which are based on the determination of the initial RTNDT of each component and on an assessment of its evolution (ΔRTNDT) by means of an empirical prediction formula.

Currently, we use a formula developed by FRAMATOME for the products used in France, known as **FIS** (embrittlement by higher irradiation) :

$$\mathbf{\Delta RTNDT\ (^\circ C) = 8 + [24 + 1537\ (P-0.008) + 238\ (Cu-0.08) + 191\ Ni^2Cu]\ [\varnothing/10^{19}]^{0.35}}$$

where : . $\varnothing$: fluence in n.cm^{-2} (E>1 Mev),
. $10^{18} < \varnothing < 6.10^{19}$
. P, Cu, Ni: content by weight - %
. Cu – 0.08 = 0 if Cu < 0.08
. P – 0.008 = 0 if P < 0.008
. 275°C ≤ Tirradiation ≤ 300°C

Following the results, a new formula, **EDFs**, has been recently developed by EDF concerning the **welds**. This development took into account all the available data on irradiated welds in the chemical and neutronic ranges met in the French reactors. This is :

$$\Delta \textbf{RTNDT (°C)} = \mathbf{22 + [13 + 823\ (P \geq 0.008) + 148\ (Cu\text{-}0.08) + 157\ Ni^2Cu]\ [\varnothing/10^{19}]^{0.45}}$$

where : . $\varnothing$: fluence in $n.cm^{-2}$ (E>1 Mev),
. $3.10^{18} < \varnothing < 8.10^{19}$
. P, Cu, Ni: content by weight - %
. Cu – 0.08 = 0 if Cu < 0.08
. 285°C ≤ Tirradiation ≤ 290°C

Logically, in the light of experience feed-back, organization and practice have evolved over the course of time and different standardized plant series built in France, this mainly concerns the type of fracture mechanics specimen, associated with the impact strength test specimen in the capsules, and the nuclear instrumentation [4].

6.5. Results of the Analysis of the Surveillance Program

To date, 130 capsules, removed from 51 power units, have been analyzed. The content of embrittling elements (copper, phosphorus and nickel) present in the materials concerned is relatively low, with the exception of the welds on the CP0 standardized plant series (6 first 900 MWe power units), as shown in table 2.

The results of mechanical characterization are associated with a neutron dose, obtained from the interpretation of the activity measured on the fissile and activation dosimeters. This metrological process, performed according to a well-established methodology, requires a certain coherence to be obtained between the activity measured, the operating diagram of the power unit and the migration calculations performed according to reactor geometry. This coherence is evaluated at less than 10% over all the results and indicates the quality of methodology used. Furthermore, these results reveal very homogenous operation of all the reactors.

The embrittlement of the various materials monitored in the surveillance program may be expressed in the form of shifts in the transition of the impact strength curves and is illustrated by figures 1 to 3. It remains moderate, yet with relatively high dispersal of the results. As regards the base metal, the maximum value obtained is 83°C and corresponds to a dose of $5.24\ 10^{19}\ n.cm^{-2}$, for the weld, 79°C and $6.91\ 10^{19}\ n.cm^{-2}$.

For both the base metal and welds, the measured shifts reveal a scatter in the embrittlement results of up to 60°C for a given fluence, together with cases of “abnormal” embrittlement kinetics. We can notice that, in the case of HAZ, the scatter is significantly lower. This shows that irradiation is not only the phenomenon at the origin of this scatter.

7. THERMOHYDRAULIC AND MECHANICAL EVALUATION TO DETERMINE RT_{NDT}

For thermal-hydraulic and mechanical evaluation the input data values necessary are :

- the different transient considerations in level A - in level C and in level D ; the most severe situation of loading in this case is the small-break LOCA considered in level C,
- the temperature of safety injection fluid (9°C),
- the coefficients of security considered by hypothesis in level A - level C - level D, the size of the defect postulated at azimutal situation and maximum fluence value and after in-service inspection the defect size and position in the vessel shell.

In fact the Relation B in the diagram (fig. 1) is the following relation

$$\boxed{\mathbf{A(Cs) \cap B(D) \cap C(T) = RT_{NDT}\ limit\ value}\quad (2)}$$

(intersection set A - set B - set C)

A(Cs) is a function of coefficient of security,
B(D) is a function of defect (in subcladding area),
C(T) is a function of transient.

8. IN-SERVICE INSPECTION AND DEFECT SIZE AND LOCATION TO DETERMINE FLUENCE AND RTNDT

In methodology presented in the table 1, the inspection of the core zone and the results in terms of defect size in the under-cladding area or not is input data introduced to determine the fluence and RT_{NDT} values.

In-service inspection of the vessel zone adjoining the core (Fig. 4) is intended to identify subcladding defects that are likely to aggravate end-of-life brittle fracture risk under incident or accident conditions.

A non-destructive, underwater testing method based on "focused ultrasonic probe" technology was thus developed since 1998 to cover the "first 25 millimeter depth" of the vessel subcladding zone and applied from the second 10-years outage NPPs.

As part of its in-service inspection program for French power plants, EDF has now included the systematic testing of all RPVs during outages scheduled every 10 years.

On the basis of the minimum detection threshold, a "reference" defect, flaw 6 mm deep and 60 mm long, has been defined for all vessel integrity studies [5].

Mechanical tests carried out on all 900 MWe-class vessels show that such a defect remains acceptable, regardless of its location in the "sensitive" zone of the vessel, under all transient loadings, after allowance for the effects of irradiation aging.

To supplement this "generic" criterion, the few "real" indications of more than 6 mm discovered during inspection must likewise be taken into account.

Considering the in-service inspection results for the few "real" indications caracterized, it is evaluated, at the situation of the défect, the real fluence value considering the azimutal distribution of the fluence.

The results is fluence decrease in function of the geographical angle. One example is shown in the application after FESSENHEIM unit 1 inspection during the second 10-years outage performed in 2000.

For TRICASTIN unit 1, this methodology was applied. The acceptability of flaws 12 mm deep and over 50 mm long was demonstrated, along with suitable safety margins, for operating lifetimes of 40 years. This result was made possible by good vessel resistance to embrittlement and confirms the acceptability of the approach used by EDF to build its generic safety case.

8.1. RT_{NDT} value in function of the fluence and azimutal distribution of fluence

The program of reduction of fluence engaged with lifetime management program, shows that the target of RT_{NDT} level is lower than basic design values, on the order of 20%.

In correlation with the fluence value associated with defect situation, the RT_{NDT} is determined using prediction formulas and chemical composition.

The application of this methodology (relation [A]) for the FESSENHEIM unit 1 vessel shows that, considering the defect situation (328°), the RT_{NDT} is 28 °C.

The RT_{NDT} obtained by the relation [B] in the worst case is 190°C (level C), and the comparison of these results shows the value of margin interme of RT_{NDT} (Fig.6 - table 2).

9. SYNTHESIS OF MAIN ELEMENTS INPUT DATA TO THERMAL-HYDRAULIC CALCULATIONS AND MECHANICAL ANALYSIS

To perform this demonstration, and more generally to optimize vessel operating life, it is thus necessary:

- to determine the fracture toughness of irradiated vessel materials. This entails upstream knowledge of initial vessel RT_{NDT}, fluence values at all points in the "sensitive" zone and shift in RT_{NDT} induced by radiation. It also means verifying the quality of the correlation between RT_{NDT} and toughness.
- to make suitable flaw size assumptions (based on in-service surveillance results), to both define the size of the minimum defect that will not "escape detection" and provide an exhaustive inventory of the significant defects recorded.
- to perform the computations required to ascertain true safety margins, by constructing an in-depth thermohydraulic model of the most severe transients, and associating it with precise thermomechanical calculations of loadings in the vessel.

This method and tools associated are applied on specific vessel by vessel assessment, and methodology [A] shows fluence and RT_{NDT} levels of the vessels and will compare with RT_{NDT} obtained with methodology [B] .

10. CONCLUSION IMPLEMENTATION OF THESE TOOLS AND METHODOLOGY TO FOLLOW UP THE RPV ASSESSMENT

The evolution during the lifetime in operation of all of EDF's 58 PWR units requires a good knowledge of the evolution of mechanical and metallurgical parameters of each Reactor Pressure Vessel and primary circuit components. The development of specific tools and methodology to follow up the evolution of RPV fluence level and the RT_{NDT} to verify the integrity assessment is a major objective for EDF.

The RT_{NDT} values for at least 40 years is determined by the fluence evaluation using special calculation code, named "EFLUVE", which is combined with the result of ultra-sonic probe (Twenty Five First Millemeter) used for in-Service inspection applied during second 10-years outage.

Following the specific vessel by vessel methodology, it is possible to simulate the RT_{NDT} values for at least 40 years and to correlate them with basic design hypothesis. The result shows the conservatism and the gain obtained in term of margin.

The implementation of these tools and methodology by EDF on PWR NPPs is a major objective for life management.

REFERENCES

[1] CUISINIER, M., DE KEROULAS, F. VALIBUS, L. EDF "Plant Life Management of the French NPPs", ICONE 5 Conference - NICE France 1997.

[2] RIEG, C. and als - EDF "EDF program on aging mechanisms of primary system components", PLIM - PLEX conference - NICE France 1995.

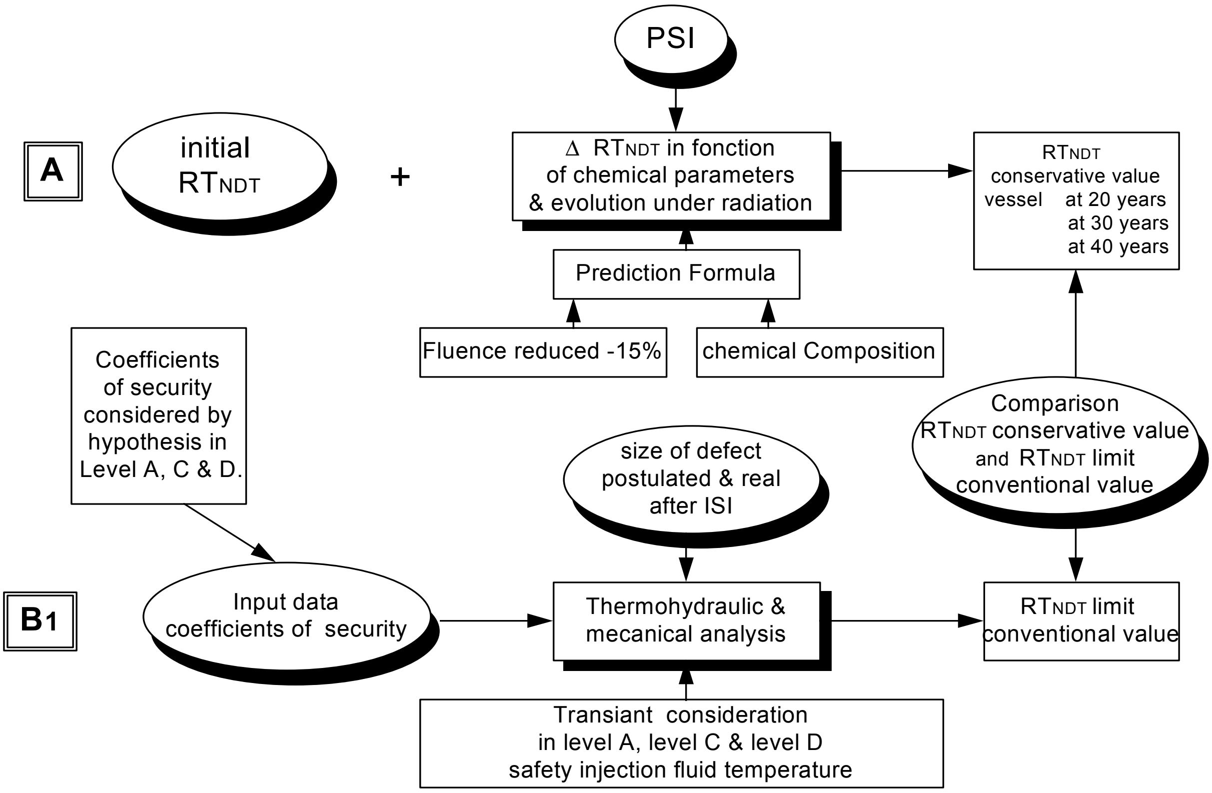
[3] COURTEAU, C. and als - EDF France "assessment of french reactor pressure integrity : Neutron fluence calculations at EDF", ASME PVP Conference - ATLANTA Georgia USA, July 2001

[4] BEZDIKIAN, G. and als - EDF France "PWR Vessel life management on French NPPs - Integrity assessment and maintenance strategy", ASME PVP Conference - BOSTON Massachusetts, USA, August 1999.

[5] BEZDIKIAN, G., MOINEREAU, D. and als EDF France "PWR vessel integrity assessment & life management in french nuclear plants", ASME PVP Conference - SEATTLE – Washington, USA, July2000.

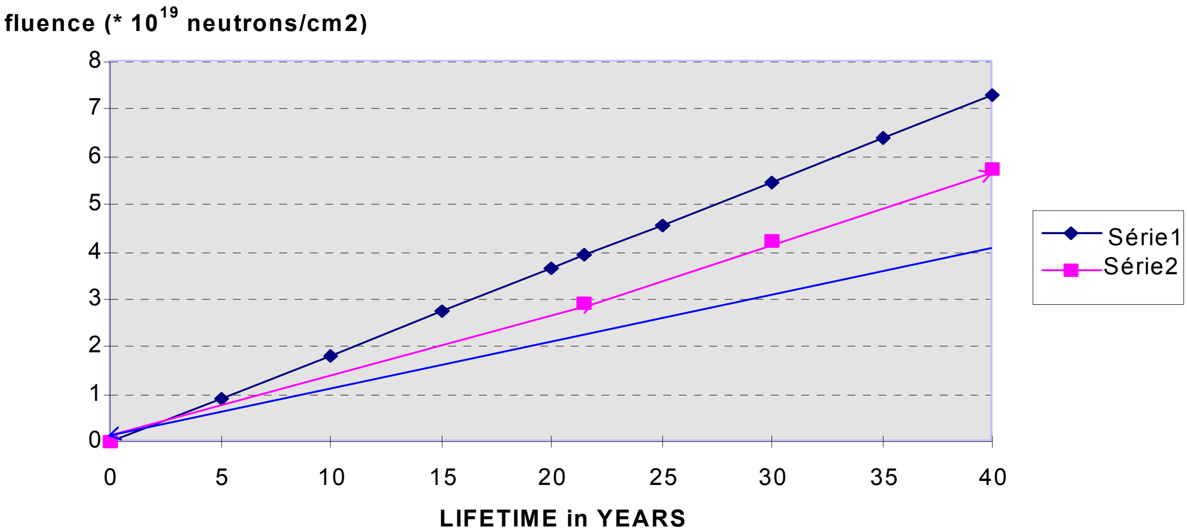
Table1

METHODOLOGY OF SPECIFIC RT_{NDT} EVALUATION VESSEL by VESSEL



- Diagram to evaluate RTNDT vessel by vessel

VARIATION of FLUENCE IN FUNCTION LIFETIME IN OPERATION



0 -> 7,3 Black curve = Basic design fluence curve

0 -> 5,8 Pink curve = Fluence reduced - 15% curve

0 -> 4, 3 Bluecurve = optimed fluence Fluence - 40 % curve

FIG. 1 Fluences curves

Table 2 Minimum, maximum and average contents (acceptance values) of embrittling elements for all French nuclear plant series (900, 1300 and 1450 MWe)

Chemical Content (weight %)		**Copper**	**Phosphorus**	**Nickel**
Base Metal	900 CP0 (6)	0.04 → 0.075 m = 0.055	0.008 → 0.012 m = 0.0097	0.685 → 0.80
	900 CPY (28)	0.045 → 0.09 m = 0.07	0.007 → 0.012 m = 0.0093	0.71 → 0.765
	1300 (20)	0.05 → 0.09 m = 0.069	0.004 → 0.011 m = 0.0073	0.71 → 0.79
	1450 (4)	0.05 → 0.08 m = 0.065	0.004 → 0.005 m = 0.0045	0.68 → 0.74
Weld	900 CP0 (6)	0.08 → 0.13 m = 0.107	0.013 → 0.019 m = 0.0161	0.09 → 0.53
	900 CPY (28)	0.02 → 0.06 m = 0.041	0.007 → 0.014 m = 0.0107	0.52 → 0.73
	1300 (20)	0.01 → 0.06 m = 0.037	0.007 → 0.011 m = 0.0084	0.47 → 0.76
	1450 (4)	0.02 → 0.05 m = 0.040	0.004 → 0.010 m = 0.0067	0.61 → 0.71

CP0 = Fessenheim/Bugey plants - CPY = Other 900 MWe plants

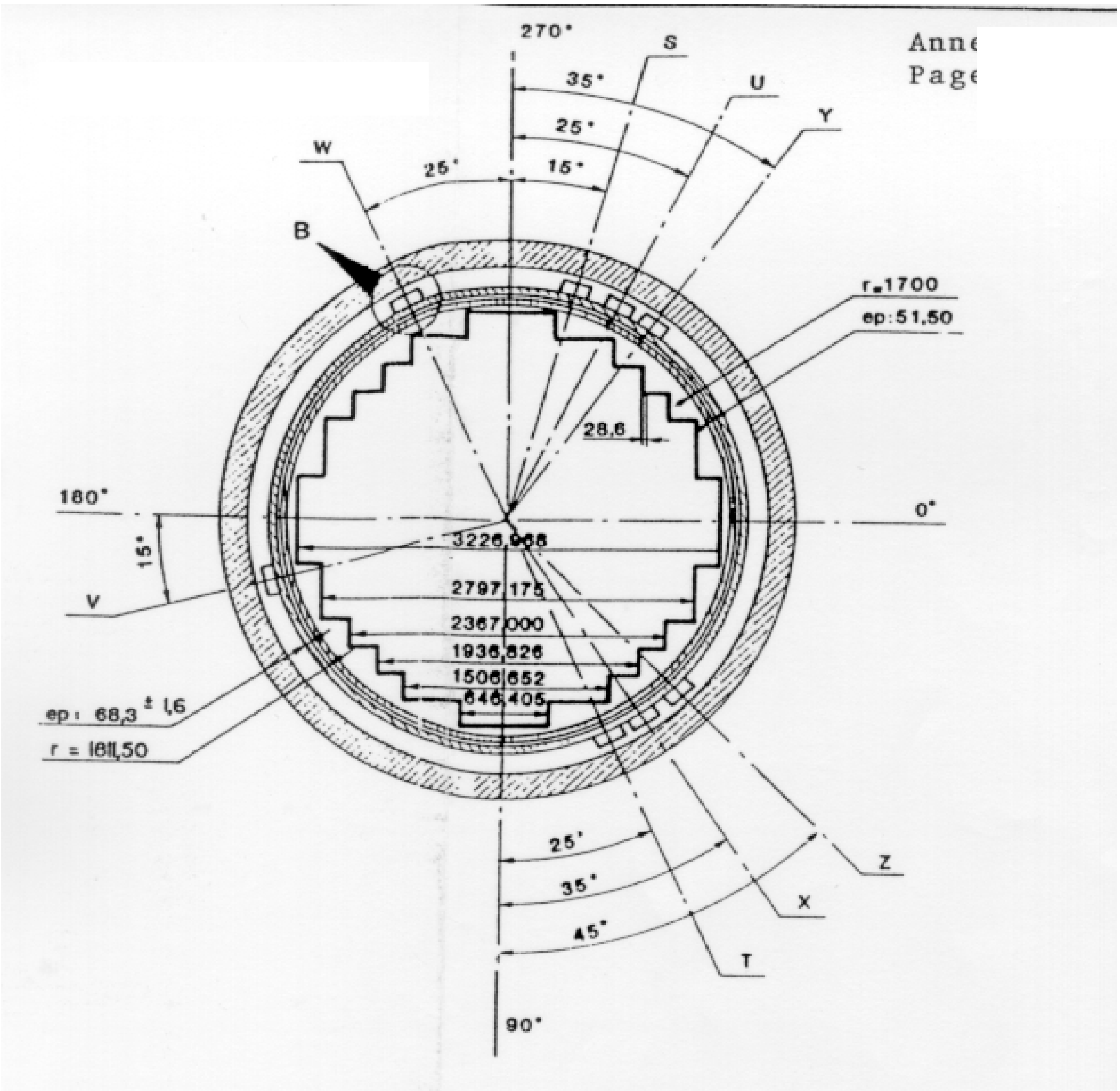
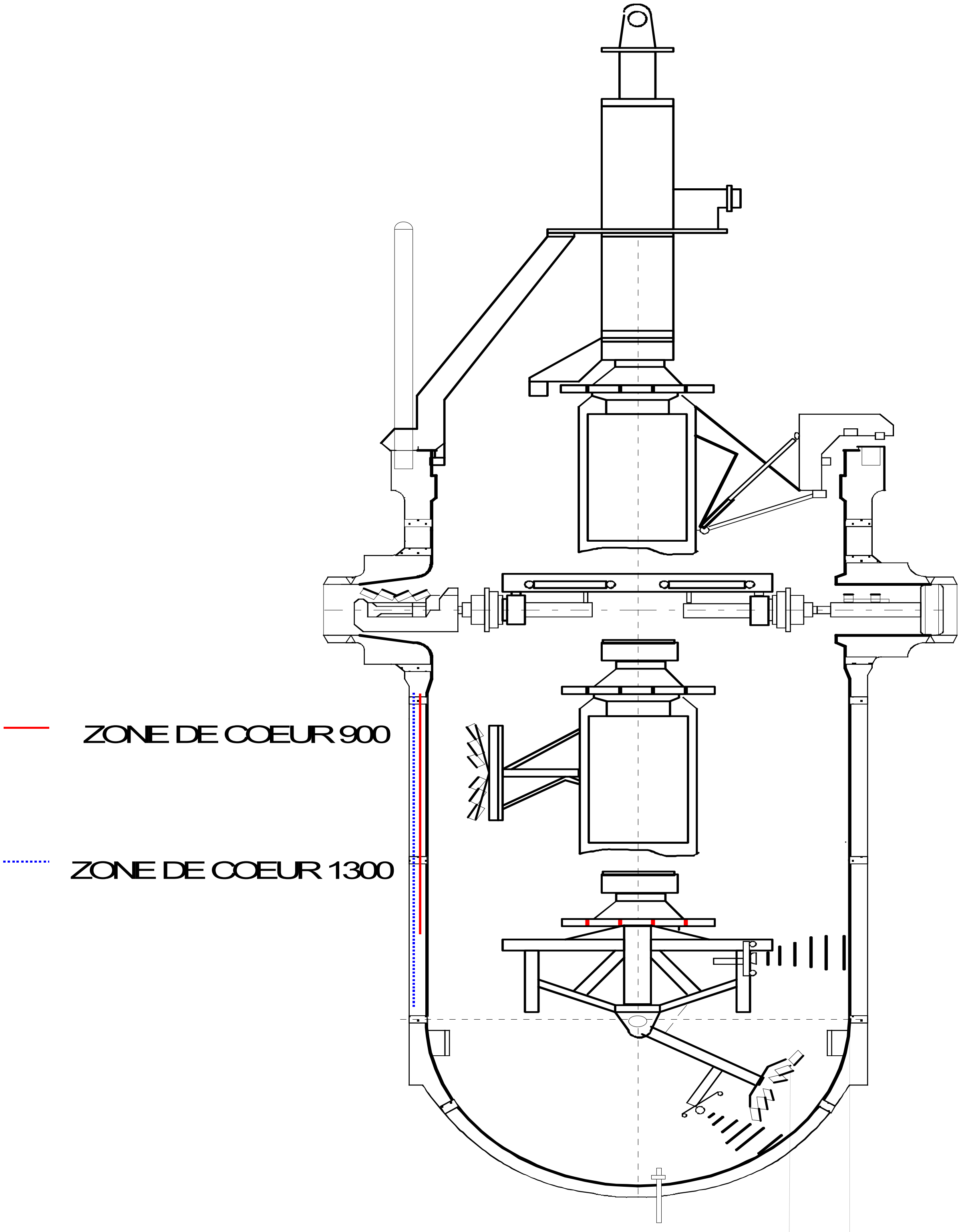


FIG. 2: Situation of radiation specimen capsule in RPV



Inspection de la cuve avec la MIS
(machine d'inspection en service)

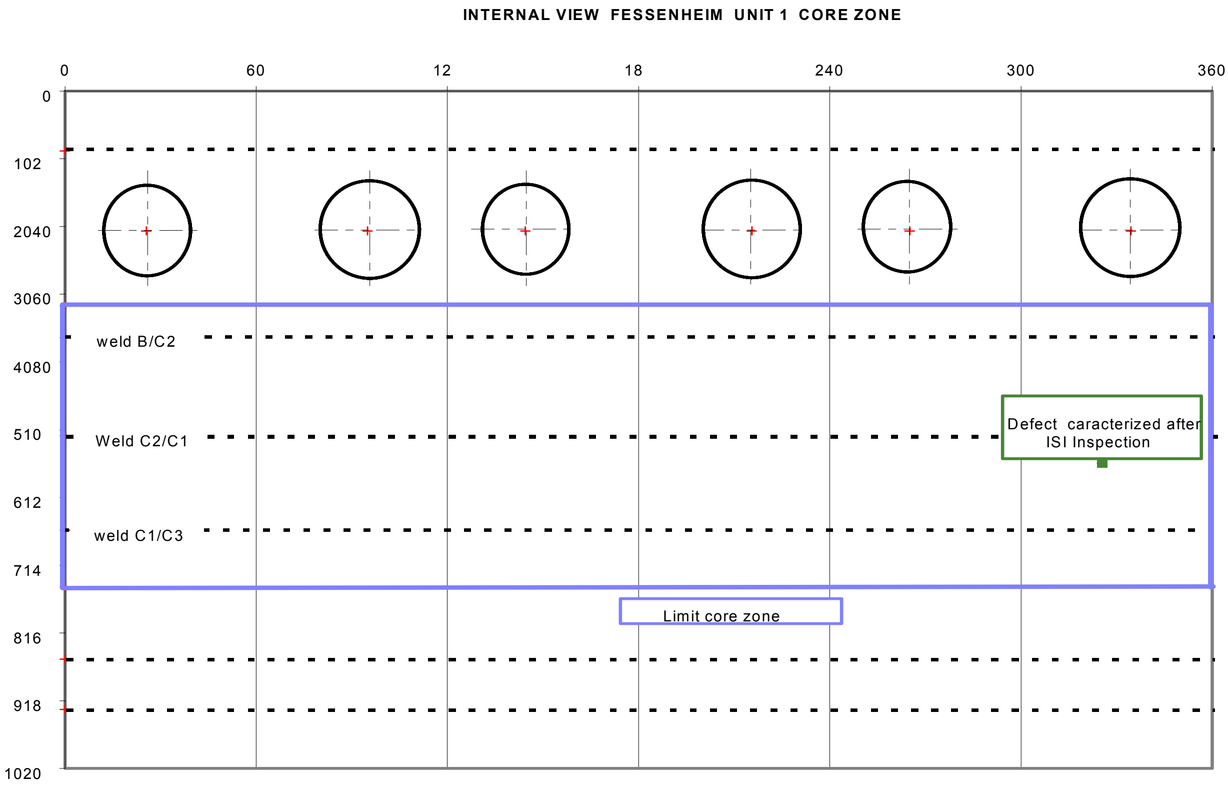


FIG. 3: all core zone inspected by ultrasonic Probe

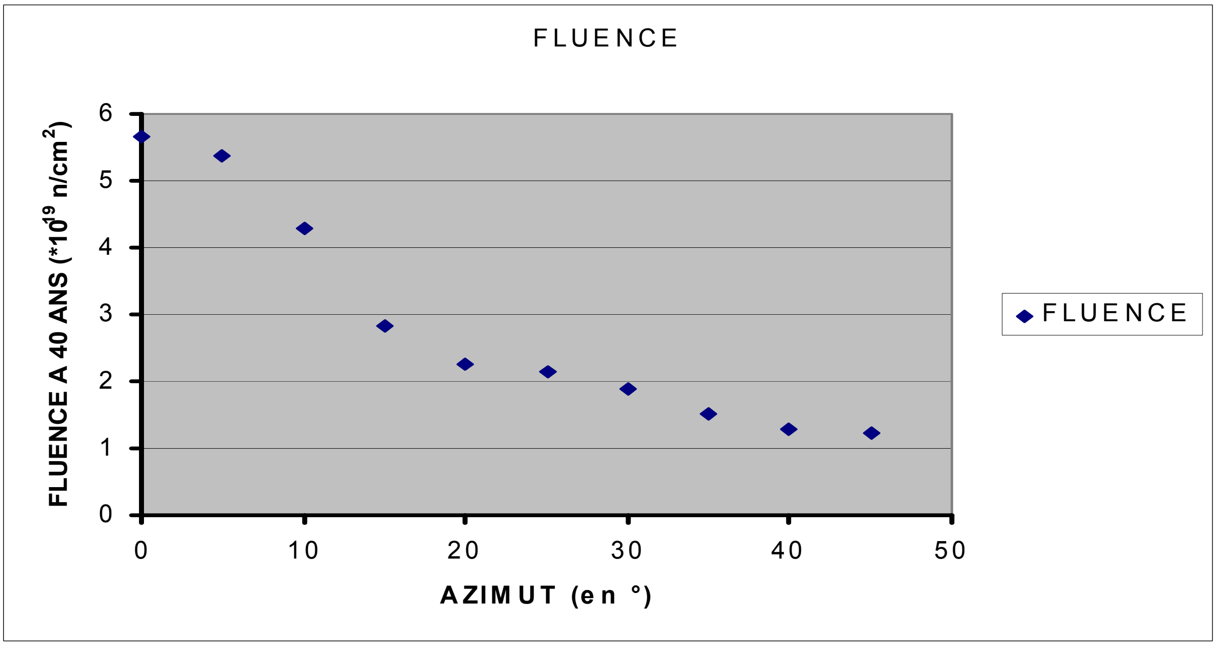


Table 3

	RT_{NDT} "conservative value " in funnction of fluence and the chemical composition at 40 years	RT_{NDT} "limit conventional value" in function of transiant fluid temperature and coefficient of security
Level A	28°C	> 300°C
Level C	28°C	190°C
Level D	28°C	244 °C

BASIC CRITERIA AND APPLICATION EXAMPLES OF GERMAN UTILITY PLIM CONCEPT

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Abstract

In German NPP's no specific ageing management programme documentation exists. The German utilities understand that the subject "Ageing Management" is comprehensively covered by the entirety of the different precaution and maintenance measures and regulations already established. The measures concerning long-term integrity of safety relevant components are performed under different names. However, divergent interpretations as to how to handle technical contents and executing administrative issues require harmonised understanding and handling. Therefore intensive ageing management activities have been initiated in Germany considering international practice. It is intended to provide a plant specific documentation consisting of a general basic report and a recurrent report, describing just the deviations. The scope contains technical subjects and safety culture aspects. For the application of the ageing management-concept the scope has to be defined. According to the international approaches the utilities introduced a systematic ranking of safety relevant systems with high safety requirements until to the level of lower safety requirements. Accordingly to this systematology surveillance measures have been installed. For components with lower safety requirements replacement strategies are preferred.

1. MAIN FEATURES OF THE "NEW GERMAN ATOMIC LAW":

As a consequence of the consensus negotiations between the present Federal German Government and the German utilities the new Atomic Energy Law was set into force in April 2002. The main issues are:

a) Phase out of NPP-operation after a maximum lifetime of 32 years without any claims for compensation.
b) Termination of spent fuel reprocessing and switching over to direct final storage. Stop of spent fuel casks shipment in 2005.
c) Intermediate storage facilities are to be provided on each power plant site.
d) The promotion clause for nuclear energy is cancelled, the construction of new NPP's is prohibited.
e) The NPP safety status has to be kept on a high level standard. A periodic safety assessment must be performed "according to the state of the art" based on up-to-date codes and standards in a 10-year interval.

As a consequence, the future German policies and strategies are based on this law.

2. DEVELOPMENT OF NEW SAFETY RULES WITH A MODIFIED STRUCTURE (KTA 2000)

The current code regulations (KTA-rules) are to be considered as fulfilment rules providing the technical basis to cover the safety requirements for current operating LWR-plant type - the Konvoi-generation. Problems arise in periodic safety assessments for older plants, because their safety goals and their defence in depth concepts are fulfilled by different plant operational and emergency systems. Thus, the objective of the new rules KTA 2000 is to give an assisting guideline concerning the evaluation criteria for future plant assessments.

Additionally, new technical and non-technical features will be included in the new structure, e. g.:

- Analysis methods,
- Consideration of Human Factor,
- New Techniques and Organisation issues.

The elaboration of the new KTA 2000 is supposed to be finalized in 2003.

3. SAFETY CULTURE

In German NPP's the large technical back fitting measures have been completed. Nowadays, in the phase of a liberalised energy market in Europe, it is very important to focus on the safety culture aspect in the NPP's itself and in the utility organisations. Thus, Germany utilities intend to install guidelines to quantify the safety culture standard in order to maintain the current high level and to be prepared for remedial actions, if required, to prevent fading safety culture degradation effects.

4. APPLICATION OF GERMAN UTILITY PLIM CONCEPT

Ageing management (AM) programmes have been launched in different countries in the US. Ageing management activities were initiated to demonstrate the long term integrity of nuclear power plants for plant life extension purposes. In other countries (for instance Switzerland) AM programmes became an important issue as well. In addition, the International Atomic Energy Agency (IAEA) has published recommendations concerning the physical ageing of safety relevant systems. As a consequence of these international developments, the ageing management aspect was introduced in Germany, too, although plant life extension is definitely not the key subject in the German nuclear industry, today. The situation in Germany concerning long term integrity of safety relevant NPP-systems and components is determined by the requirements stated in national regulations and codes. They contain basic requirements for continuous precaution measures according to the current "state of the art" starting already with the plant commissioning. This includes demands for redundant safety and for surveillance measures. A continuous adjustment related to the requirements of the respective "state of the art" is provided by the German utilities and submitted to the responsible safety authority in order to demonstrate an appropriate integrity status of the safety relevant systems.

In Germany, no specific ageing management programme documentation exists. The measures concerning long term integrity of safety relevant components are performed under different names. Nevertheless, the German utilities understand that the subject "Ageing

Management” is comprehensively covered by the entirety of the different precaution and maintenance measures and regulations already established.

Because of this particular attention the German NPP’s are now free from major constraints from age-related degradation.

There are numerous reports being submitted to the safety authority demonstrating reliability of the ageing management actions even being performed:

- Monthly plant operation report,
- Annual plant operation report (including cycle counting for fatigue relevant components),
- Annual report about meeting ‘state-of-the art’ requirements,
- Plant specific assessments concerning incidents from other plants,
- Plant specific periodic safety analyses,
- Etc.

Additionally, comprehensive Safety Reviews have been carried out. The results show that even the older NPP’s built to earlier standards meet the present safety requirements and that sufficient precaution against damage to the environment has been taken. Thus, the safety related work that has been carried out already to bring NPP’s to the current ‘state-of-the-art’, has ensured that safety related obsolescence is not significant.

However, divergent interpretations as to how to handle technical contents and executing administrative issues require harmonised understanding and handling. Therefore, intensive ageing management activities have been initiated in Germany considering international practice. The Reactor Safety Commission (RSK) submitted recommendations containing basic principles. Further activities in RSK Subgroups have been initiated. In parallel, the German utilities have installed a Working Group to clarify the German utility ageing management concept structure and its sound application to give a guideline for future applications.

It is intended to provide a plant specific documentation consisting of the following reports:

- **“Basic report”** containing all general criteria plant specific component classification and the present status concerning the ageing management issues.
- **“Recurrent report”** (annual?) describing just the deviations (‘Deltas’) compared to the basic report.

The first step is to distinguish between ‘Plant Life Management and ‘Ageing Management’ considering the international wording and the liability for the related actions. Essentially the following categories of NPP items have to be considered:

- Equipment (mechanical components, structures, electrical and Instrumentation & Control systems),
- Computer Systems (hardware and software) required for plant operation,
- Plant specifications and documents.

'Plant Life Management' includes all technical and organisational measures on ageing phenomena which are identified and managed by the plant operator and which guarantee achievement of the prospected service life or (identifies) if appropriate action needs to be taken. 'Plant Life Management' aims towards component/system or actions/measures safety and availability issues. Life Management is therefore an activity basically carried out by the utilities for each NPP. Economical reasons have to be considered. Those are:

- Plant availability,
- Reduction of maintenance costs,
- Optimisation of shut down periods,
- Cost reduction in general.

'Ageing management' of safety significant components/systems and all issues related to plant safety are already covered by regulatory supervision together with independent advisory organisations. A continuous adjustment to the respective "state-of-the-art" is provided by the German utilities and submitted to the responsible safety authority.

As a first step for the "ageing management" concept application the evaluation scope has to be defined.

Basically, the handling of ageing phenomena can be divided into:

- The technical and administrative ageing management of safety relevant systems/components and of safety relevant actions (e.g. PSA-Probabilistic Safety Analysis) and its supervision by the responsible safety authority (PLEX activities, but due to the current political situation in Germany, no lifetime extension approval is expected).
- The technical and administrative lifetime management of the remaining system/components and quality assurance actions and maintenance activities to be performed mainly on the utilities responsibility (PLIM-activities).

All PLIM and AM measures have to cover the following topics:

- Define safety/availability significant systems/components or non-technical issues (e. g. mechanical components, I&C components; building structure).
- Determine the current system/component quality status and safeguarding measures.

Safeguarding of quality requirements has to be ensured during plant operation by **proactive** and **reactive** measures. **Proactive** is the monitoring of root causes of potential operational degradation mechanisms in terms of operational loadings. The proactive approach tries to avoid/minimise premature degradation effects. The **reactive** surveillance of consequences of potential operational degradation mechanisms deals with degradation effects after they have already occurred and been detected (e. g. by NDT-measures for mechanical components).

According to the international approaches, ageing management for safety relevant systems can be focused on Class 1-systems and systems required for safe plant shutdown. Within these safety relevant systems, the following safety significant/component/part ranking can be introduced:

Group 1: Components of high safety requirements
("Guarantee" required integrity status by monitoring root causes and consequences of operational degradation mechanisms.)

Group 1 components are the RPV, systems with "Leak-before-break" (LBB) requirements and other components classified in Group 1 due to specific safety or plant availability reasons.

Group 2: Components with medium safety requirements
("Preserve" required component quality by preventive maintenance activities.)

Group 2 components are mainly redundant components such as valves, pumps, electrical, I&C components, building structures and other components with specific safety or availability requirements.

For components existing redundantly, a single case failure is not a safety problem as long as no common cause failure occurs.

Group 3: Components with lower Safety requirements
("Component replacement after failure".)

Ageing management is mainly based on common preventative maintenance measures performed in nuclear power plants (e. g. for valves). Maintenance is performed either as visual inspection, maintenance issues, repair or replacement (either time orientated or based on the actual component condition). Time orientated maintenance means that the components under consideration (e. g. valves) will be inspected in fixed time intervals (e. g. 4 or 8 years according to their safety classification). If the maintenance is based on the existing component condition the time period for inspections will be chosen individually. It is known from experience that maintenance activities performed too frequently may lead to additional ageing effects. For Group 1 components additional surveillance measures have been installed. For Group 3 components the ageing management may be performed by replacement strategies.

5. CONCLUSION

The German utilities understand that the subject "Ageing Management" is comprehensively covered by the entirety of the different precaution and maintenance measures and regulations already established.

LIFETIME MANAGEMENT AND LIFETIME EXTENSION AT PAKS NUCLEAR POWER PLANT

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Abstract

The Paks Nuclear Power Plant strategy is to extend the operational lifetime of the plant and renew the operational licence for 20 years over the designed and licensed lifetime. At Paks NPP the systematic ageing management activities were started eight years ago. It is now a deliberate and programmatic activity, which ensures the plant safety. The programme started with selection of equipment important from both lifetime and safety point of view. The systematic ageing management includes the definition of ageing processes, identification of sensitive parts of the components and the dominant ageing processes, and their control and monitoring. A computerized monitoring system supports the ageing management activity of the plant. The feasibility of plant lifetime extension has been investigated. The study includes a detailed plant assessment, ageing and lifetime prognosis of representative set of structures, systems and components, definition of necessary reconstructions and detailed business analysis. It has been found that a 20 years extension of operation is feasible from both technical and business point of view. Basic results of plant assessment and business analysis are discussed in the paper. A project has been launched by NPP Paks for preparation of plant life extension and licence renewal. The project tasks and conditions of success are presented and discussed in the paper.

1. FOREWORD

The Paks Nuclear Power Plant has a special energy-policy role in Hungary. The diversity of electric energy production is achieved and maintained by the plant in terms of the production technology and the nature and distribution of resources of the primary energy source. As a state owned capacity and dominant public utility the Paks NPP provides 38-40% of domestic generation at lowest price. The nuclear power plant nowadays, and possibly in the long run as well, is a potential tool for market control. The Paks Nuclear Power Plant can significantly diminish the risk of the import dependence of the national economy, as the nuclear fuel does not come from the crisis regions of the world, and can easily be stockpiled for several years. The Paks NPP has, at present, a Russian and – potentially – a British fuel supplier. The storage of a strategic inventory of fuel is still a practice at Paks Nuclear Power Plant.

The structure of the Hungarian electric energy system is presently well balanced. Before 2010, almost exclusively, gas fired power plants are expected to be constructed, and some coal fired ones are predicted to be closed down. A significant change in the structure of energy-production would occur after 2012, if the Paks units were shut down with the expiry of their design lifetime (Fig. 1.). Based on the present constructional tendencies and market automations, the industry is predicted to recover the lack of electricity and the growth of demand with gas fired power plants that produce energy more expensively compared to the nuclear power plants, or would import the electric energy itself, and thus increasing the import-dependence of the company. This way between 2012 and 2019 the gas consumption of electric energy production, as well as its carbon-dioxide emission would double compared to its present values (even in case of an intensive utilization of renewable energy sources). The electric energy import would, in the long run, be an expensive and obviously import-dependence-increasing solution. For the compenzation of the production of Paks NPP it is rather difficult to find a green alternative. As a possible alternative, for example, 11 000 (eleven thousand), environment-friendly windmills, similar to the one operating now in Hungary, would be needed. The strategically unfavourable structural changes can be counterbalanced via upholding the market position of the Paks NPP as well as the power uprating and the lifetime extension of the Paks units.

In 2000 the Paks NPP and the Electric Energy Research Institute looked into the possibilities and alternatives of lifetime extension of the nuclear power plant, as well as the technical feasibility of the alternatives. The business analysis of the feasibility study was completed by Ernst & Young Ltd. Based on this study the management decided on the extension of the operation 20 years over design lifetime for the preparation of which the Paks NPP launched a preparatory project. The most important findings of the feasibility study as well as the preparatory activity are introduced bellow.

2. PRECONDITION OF LIFETIME EXTENSION

It is obvious that lifetime extension is not a strategic decision without precedents. It has been influenced by numerous external circumstances that are independent from the Paks NPP as well as by factors dependent on the characteristics of the NPP and the practice of the operating Company.

The future of the Paks NPP depends greatly on the international tendencies. The energy policy of the USA motivates the lifetime extension activity in Hungary very much. The situation in Western Europe is unclear with respect to the future perspectives of nuclear generating capacities, but some new tendencies in the European Union energy policy are also encouraging. International experience shows that nuclear power plants perform well in the liberalized market. In the Eastern- and Central European countries lifetime extension of plants is intended. This all means that at least 6 units, similar to the ones in Paks, are expected to be in operation until 2030 in Central Europe, so the Paks NPP will not be an isolated, unique phenomenon in the future either.

If the Paks Nuclear Power Plant and the whole of the industry were not ready for the start of lifetime extension programme, the favourable international atmosphere would not be exploited.

Lifetime extension is a strategic decision that is entirely based on the design- and manufacture features of main components; the robustness of the main equipment and of the whole construction; on the system of technical inspections and tests; the maintenance practice; as well as on the good condition of the plant maintained via reconstruction and refurbishment.

The ageing of the plant structures, systems and components (SSC), relevant for safety is treated as a central issue both by the operational – maintenance practice of the Paks NPP and by the nuclear safety regulation. Within the frame of the Periodic Safety Reviews in 1997-1999 it had to be confirmed that the safety functions of the relevant structures, systems and components (SSC) is ensured in spite of the ageing processes. On account of these requirements the systematic ageing-management activity started approximately eight years ago. The activities performed in the nuclear power plant from the beginnings, such as surveillance of the embrittlement process of reactor vessel material, monitoring erosion-corrosion events and the practice of technical inspections were the starting basis to build up a conscious ageing management programme. Already in the course of the Periodic Safety Review the ageing processes of the critical equipment were specified, and ways of tracing changes in status as well as possible correction measures were determined. In addition to ageing management and monitoring of the critical components, a status monitoring of structures, equipment and components is going on in the nuclear power plant, hereby ensuring the required performance of the large number of (although replaceable) components.

Monitoring ageing processes of the critical components and data collection is maintained with advanced computer aid in case of systems and components with enhanced safety significance. The screen of this monitoring system for the pressurizer tank is demonstrated In Fig. 2.

The conscious ageing management has already grained ground during modifications and replacements performed so far. Replacement of turbine condensers is a good example for this, which, since the new condensers have stainless steel tubing, allows for introduction of high pH secondary circuit water chemistry decreasing the rate of deposition raising local corrosion tendency of steam generators.

Safety of the Paks NPP is a prerequisite. A comprehensive safety-upgrading programme is presently in progress at the Paks Nuclear Power Plant, as a result of which the safety level of the power plant complies with the requirements towards nuclear power plant units of similar age operated in developed countries. Safety of the nuclear power plant must satisfy both domestic and international requirements. The safety of the plant, however, is not static, as the new recognition and experiences generate new requirements that have to be met appropriately.

Operation over design lifetime and the renewal of the operational licence is an objective interpreted and licensed within the frame of the Hungarian nuclear safety regulations. The renewal of the operational licence is feasible in the system of the relevant specific requirements, the annual updating of the Final Safety Report and the Periodic Safety Reviews in a correct way, with the guarantees necessary for the society.

In the strategic decision-making process the public acceptance of the Paks Nuclear Power Plant, which is permanently around 70%, had an important role.

In order to clarify technical questions of the lifetime management a significant research work has been in progress for years in several institutions. This activity may serve as basis for the acknowledgement of the technical-scientific competence of Hungarian experts.

It was recognizable already in 1992 that favourable characteristics of the plant, the comprehensive safety enhancing programme, the surveillance and maintenance practice of the operator give an opportunity to enhance the lifetime of the Paks NPP [1]. A complex technical-economical study for the feasibility of lifetime extension was completed in 2000, which fully confirmed this assumption.

3. FEASIBILITY OF LIFETIME EXTENSION

The feasibility study of lifetime extension included the processing of international technical and regulatory information related to the lifetime extension of nuclear power plants with a special respect to the USA experience [2]; assessing the technical condition of the nuclear power plant and specifying the technical and safety measures required for lifetime extension, together with estimating the costs needed [3]; and, furthermore, completing the business plan of the lifetime extension [4].

3.1. Assessing plant status

Assessment of the plant status has been carried out on a representative set (~ 500 items) of SSC. This analysis covered the lifetime-perspectives of SSC, the maintenance, ISI, etc. practice, as well as the data related to ageing and degradation processes. Assessment of the plant status and the process of technical inspection is to be seen in Fig. 3, whereas the system of aspects investigated in Fig. 4.

It has been found that there is no technical or safety limitation to the 50 years of operation of the Paks NPP. In case of most systems and equipment the monitoring, maintenance and regular renewal practice of the plant allows for the lifetime extension without outstanding costs. There is only a well defined number of SSC requiring extensive reconstruction and investment as the possibility of compensating for the effects of ageing is limited, or a significant moral ageing can be expected. In case of some SSC capacity expansion might be needed (e.g. radioactive waste storage tanks).

Findings related to the reactor vessels and steam generators should be dealt with separately on account of their increased significance.

As for the reactor vessels of VVER/213 at Paks NPP the dominant ageing process is caused by embrittlement due to fast neutron irradiation of the material. The vessels are different per units and extending their lifetime can be realized under different conditions. At units 3-4 reactor vessels do not require extra measures even at 50 years of lifetime. At unit 2 the water in the emergency core cooling (ECC) tanks has to be heated up in order to decrease stress levels caused by pressurized thermal shock (PTS) transients. For this purpose cost-effective technical solutions are available. At unit 1, in case of the 50-year lifetime, in addition to the ECC heating-up, the annealing of the welded joint No. 5/6 close to the core has to be considered with 50% probability. Annealing is no longer a cost-critical measure and it has been successfully applied in the practice of VVER plants (i.e. Finland and Slovakia).

Stress corrosion cracking of heat-transfer tubes shall be considered also in case of steam generators at Paks. Considering also the modifications (main turbine-condenser replacement, copper removing, putting into stand-by operation the ion-exchangers of condense-cleaner equipment, etc.) introduced up to now, replacement of steam generators can be excluded with a high confidence also in case of 50-year lifetime. However, local corrosion effects appearing on the secondary side shall be monitored even in case of high pH water chemistry, and entering of erosion products into the steam generator shall be minimized, for example by means of correct selection of the structural materials during replacement of high pressure pre-heaters.

3.2. Business assessment of lifetime extension

The business model of the lifetime extension covered incomes, originated from electricity generation and sales, direct operational- and lifetime extension costs as well as the financing of plant life extension programme.

The operational and maintenance expenses and the costs of investments to sustain the status of the plant were determined in a conservative way, based on practices and actual expenditures between 1994-2000, but with a consideration of unpredictable factors. Assessment of the technical status defined the time and the extent of investments during the extended lifetime. These data served as input for the business analysis.

It had been presumed that as a result of plant life management program the availability of Paks NPP could be maintained at the present outstanding level. In the initial period sales are realized on the basis of the long-term electricity-purchasing contract, and beginning from 2010, already at the totally liberalized electricity market, it will be performed under competition conditions. Data originated from combined cycle gas turbine power plants have been used as a base for formation of competitive market price. In the evaluation of costs the liability to pay into the Central Nuclear Monetary Fund, the safety enhancing investments and the anticipated costs of operational licence renewal were considered in addition to the operational and maintenance costs.

We have assumed that investments of the lifetime extension can be financed from principally from credits and the accumulating operating cash flow, reduced by the dividend.

The logical scheme of business evaluation is to be seen in Fig. 5.

Table 1: Economic comparison of alternatives of power generation

	Coal fired Power Plant	Combined Cycle Gas Turbine Power Plant	20 year life-time extension of Paks NPP
Investment expenses, HUF/kW	340000	160000	58000
Maintenance cost, HUF/kWh	1.32	0.71	2.84
Primary energy-cost, HUF/kWh	3.38	5.67	0.83
Total operating and maintenance cost, HUF/kWh	4.70	6.38	3.67

* Prognosis in case of medium energy-prices, without gas price-tendencies recently experienced.

Table 2.: Business attributes of the lifetime extension project, 20 years of extension

20-year-lifetime	if electricity price, Ft/kWh			
extension	5,85	6,50	7,50	8,50
Project payback %	8,5	17	20	28

Power uprating is not taken into account in the tables.

The calculations show that construction of a CCGTP to replace the NPP Paks could be reasonable, if during the extended operating time the real electricity price level of the CCGT generation will be below the value of 4.52 HUF/kWh. Compared to CCGTP installation the lifetime extension requires lower investment expenses, and the direct operational costs are low at nuclear power plants. This result would not be altered by an increase of nuclear fuel prices. The data for the comparison of different generating alternatives are shown in the table 1.

On the basis of the net present value criterion, the lifetime extension is reasonable if the electricity price is above of 5.85 HUF. The project is doubtlessly more economical if the 20-year life extension is considered. The payback of the project is shown in Table 2.

The Net Present Value of lifetime extension, as a project, is demonstrated as function of electricity price. We would add that the zero NPV of the project at 5,85 Ft/kWh ensures a profit of 8,5%, too.

It is also important to note, that until the acquisition of the license-in-principle for extended operation in 2007 only such costs will arise, apart from project costs, that are also needed for the 30 years of design lifetime. This way the financial risk of the Company due to the project is not significant (compared to the expected benefit).

4. THE ELEMENTS OF PREPARING THE LIFETIME EXTENSION PROGRAM

In order to operate the units of Paks NPP for further 20 years over their design lifetime, their operational licences have to renewed, the first step of which is to obtain the nuclear safety licence-in-principle until 2007 (considering the lifetime of Unit 1 as a basis), then to renew the operational licence in 2012. There is a possibility to get a licence for an operation over the design lifetime, if effectiveness and suitability of the ageing management program approved on the basis of the licence-in-principle, and adequacy of the safety analysis can be verified during the design operational lifetime (that is between 2007 and 2012).

Licence renewal is based on the following principles:

1. during the design lifetime, during the period of preparing the lifetime extension and during the extended lifetime any problems occurring in connection with operation or related to the current licensing basis of the plant should be solved within the frame of the actual operational licence,
2. good technical condition and performance of the SSC should be maintained during both the design lifetime and afterwards,
3. for this an intended activity should be initiated and continued by the operator during the design lifetime, and effectiveness of this activity should be systematically reviewed and evaluated,
4. during the operation of the units over their design lifetime there is never any possibility to utilise the necessary safety margins of the systems and components by referring to the coming end of the licensed lifetime,
5. safety improvement derived from up-to-date international requirements is performed in the frame of the periodical safety review.

In accordance with the common interpretation of the task given by the Hungarian Atomic Energy Authority Nuclear Safety Directorate (HAEA NSD) and Paks NPP, the above system of requirements can and should be met by the following:

(1) ageing management programme
(2) environmental qualification of the equipment and maintenance of their qualified status
(3) maintenance of the required performance and technical status
(4) renewal and annual updating of the final safety report.

It is clear that these tasks exist also during the design lifetime under conditions of the current operational licence, and are included as requirements in the Nuclear Safety Rules, ordered and scheduled in authority decrees relating to the Periodic Safety Review and Final Safety Report. Precondition of licence renewal is that the licensee should meet the requirements under the points (1)-(4) during the design operational lifetime. Licence renewal affects only the time span of the tasks under the points (1)-(4) and gives high priority to the ageing management activities. This logic is illustrated schematically in the Fig. 7.

The principle idea of the definition of the scope of the licence renewal and of the associated ageing management programs is that the availability and performance of active components can be controlled with tests, while the status and performance of passive long-lived components cannot. The ageing processes of passive long lived components should be managed and measures should be made for the compenzation of the ageing effects. With this respect the effectiveness of the maintenance activities shall be evaluated controlled according to safety and performance criteria. A new regulation, similar to the Maintenance Rule in the USA, will be introduced during the next five years in Hungary.

Lack of the environmental qualification of the electrical and I&C equipment is one of the basic issues recognized during Periodic Safety Review of the Paks NPP. The environmental qualification is now a requirement, which is specified in the Nuclear Safety Rules, the relevant guidelines and authority decrees relating to the Periodic Safety Review and Final Safety Report. In accordance with the valid regulations qualification and maintenance of the qualified status is a safety requirement existing independently of the lifetime extension.

Consequently, a the life management programme shall include the ageing management program of passive long-lived SSC and program for ensuring the qualified status and also programme of maintaining good plant status. The lifetime management program has to be optimized in respect of the technical content, schedule and expenses in accordance with the status of the plant.

In usual, property and assets of the plant should be managed with consideration of a 30+20-year operational lifetime. Human resources should be planned and provided, and effective knowledge-management should be performed according to that. It refers not only to the availability of human resources and proficiency for Paks NPP and the companies providing technical support, but it means that technical-scientific and educational potential of the country should be activated and revived. The public and political support, and international acceptance should be ensured. For the latter one we carried out important discussions in several international meetings. A great support for the lifetime extension program of Paks NPP is expected from the International Atomic Energy Agency Technical Co-operation Project, starting 2003. The international co-operation can contribute to international legitimacy and acceptance of the national program.

Operation of the nuclear power plant cannot be considered independently of the problem of spent fuel and radioactive wastes. Interim storage of the spent fuel on the site is ensured for 50 years and it can be resolved for the extended lifetime too. The issue of final disposal of the high activity wastes should be concerned on competent levels in Hungary, but following a "wait and see" strategy. Construction and installation of a storage facility for final disposal of low and medium activity wastes is a very important and actual task.

In order to achieve a successful solution it is essential to acquire political and governmental support in accordance with the public acceptance of the plant.

The structure of the tasks relating to the licence renewal and the obligations of the licensee in general is indicated in the Fig. 8.

5. TASKS OF THE PROJECT FOR THE PREPARATION OF LICENCE RENEWAL

On the basis of a feasibility study the owner made a decision about the future perspectives of Paks NPP, the main element of which is the preparation of the plant life extension and licence renewal for 20 years over the design lifetime. For this work Paks NPP established a preparation project in October 2001. This project prepares the documentation required for the licensing, the required additional regulatory licences, e. g. environmental licence, analyses demonstrating the operability and the adequacy of ageing management program. The preparation of the licence renewal is a complex task that can be resolved only by co-operation with Hungarian technical supporting institutions and with consideration of foreign experiences. From technical and licensing point of view the project tasks are interrelated with the power uprating, renewal of the Final Safety Report and implementation of the regulatory requirements issued after the Periodical Safety Review.

Main tasks of the Project derived from the requirements of the licence-in-principle for the operation over the design lifetime are the following:

- *The scope of the systems, structures and components, which are essential for safe operation of the units should be determined and its adequacy should be verified.*
- *Ageing processes, which should be managed in relation with the licence renewal should be determined and the adequacy of the procedures adopted should be demonstrated.*
- *Status of the systems, structures and components in the scope of the licence renewal of the units should be estimated, the ageing management programs should be evaluated and, if necessary, modified, new programs should be developed and initiated.*
- *Extent of the analyses, which are valid for a limited period and concerned in the operational licence should be determined.*
- *The Time Limited Ageing Assessments, their validity and expandability should be evaluated with consideration of the period of the operation over the design lifetime of the unit.*

The above-mentioned tasks mean that for the licence renewal it should be verified that the systems, structures and components will perform their intended safety functions also during the extended lifetime, their current status makes it possible.

From the point of view of preparation and licensing it is essential to consider and manage ageing of the passive, not replaceable components with long lifetime. In principle, status of these components determines the possibility of the lifetime extension, while maintenance of the status of other replaceable components affects only the expenses of the lifetime extension.

Consequently, a complex lifetime management program should be developed and operated, which ensures the safe and competitive operation, i.e. an ageless status of the plant. The lifetime management program means the introduction of a management system and practices, which specifies inspection and monitoring of the plant status and optimal management of the plant status on the basis of business model developed, that is the system of maintenance, technical reviews and inspections, reconstructions and investments with optimized technical content and schedule. In this program ageing management of the passive, not replaceable equipment with long lifetime has significant role from the point of view of licensing and safety. The program specifies the operational aspects ensuring the planned lifetime.

Scope of the components under the licence is determined on the basis of the scope and extent of the activities performed by the licensee during the operational lifetime and afterwards in order to meet the requirements of the current licence. Logic scheme of how to determine the scope of activities is shown in Fig. 9.

Safety analyses valid for the limited period – considering a 30-year operation – will be included in the renewed Final Safety Report. Review of such analyses, evaluation of their validity, and if necessary, performance of such analyses are also the special task of the Project. Review of the environmental qualification of the components valid for a limited period and development of a programme for maintaining the qualified status also belong to the special tasks of the Project.

Review, modification of the ageing management programmes for passive components with long lifetime and development of new programmes are the special task of the Project.

Additionally, in the frame of the Project the Final Safety Report and the Operational Limits and Conditions (Technical Specifications) shall be updated in accordance with the results of the tasks listed above.

Environmental Impact Study and environmental licence of the prolonged opration are important preconditions for the licence-in-principle. This aspects of the licence renewal are of high importance due the complexity and difficulties, and also political risks of environmental impact studies and licencing processes.

The Project performs methodology preparing work and interpretation of the requirements for evaluating the effectiveness of the maintenance.

The Project is in close relationship with the sections of the Company responsible for human resources, training and PR activities.

The Project for Preparation of Licence Renewal and for the Project for Power Uprating form an entirely Project in order to maintain the interrelation between these two strategic goals and for the better use of synergy.

The environmental licensing process and the site evaluation program – that is the basis of these licences processes – are also among the tasks of the Project.

The system of this relationships and division of the activities are indicated in the Fig. 7 with different colours. Logic connection of the tasks is shown also in the Figs 7 and 8.

6. CONDITIONS OF SUCCESS

An essential condition of the lifetime extension is to comply with the fundamental safety requisites of further permanent operability. This requires from the licensee the continuous priority of safety aspects in addition to successful completion of the safety upgrading programme.

Power uprating of the Paks NPP units, carried out simultaneous to the lifetime extension, is a very advantageous technical (or even economical) circumstance, which significantly enhances competitiveness and provides a better return for both projects.

The licensing, the regulation environment, the development of regulations are crucial conditions of the implementation of lifetime extension. The nuclear safety licensing, amongst the present legal atmosphere, is possible and the development of a detailed, lower level regulation will be completed in 2002.

Public trust must be pertained in the future, too. This, together with the obvious support of the Paks region, serves as social basis of lifetime extension. Nowadays there is a great attention towards lifetime extension. The increase in interest and devotedness can be traced not only among the population but also among the ones competent at its implementation, which must be relied on greatly in the future.

For the favourable international acceptance of the lifetime extension there is a constant exchange of information on international platforms. The Technical Co-operation Project of the International Atomic Energy Agency provides a great support for the lifetime extension of the Paks NPP.

A planned change of generation, a scheduled refreshment of the staff of the nuclear power plant, and also of the technical support institutions must be arranged for.

7. SUMMARY

Due to economic and political circumstances it is necessary to maintain the position of Paks NPP at the domestic electricity market. Following the international tendencies and exploiting the technical capabilities of Paks NPP, this can be realized by means of power uprating and lifetime extension. The feasibility and unambiguous business benefits of lifetime extension have been verified by studies. The preliminary decision for lifetime extension has been made, and preparatory work is carried out within the scope of a preparatory project. This project will demonstrate and verify, in a way transparent for the Hungarian and international public opinion, that the Nuclear Power Plant Paks, can be operated at least up to 50 years in accordance with the nuclear safety and environmental regulations and the international standards. Nuclear Power Plant Paks will stay a safe and clear source of the domestic electricity generation.

REFERENCES

[1] Katona T., Bajsz J.: PLEX at Paks: making virtue out of necessity, Nuclear Engineering International, June 1992

[2] A Paksi Atomerőmű élettartam hosszabbításának megvalósíthatósági elemzése 1. rész: VEIKI, 2000.

[3] A Paksi Atomerőmű élettartam hosszabbításának megvalósíthatósági elemzése 2. rész: A berendezések műszaki állapotának előzetes értékelése Táblázatok, VEIKI, 2000.
[4] A Paksi Atomerő élettartam hosszabbításának megvalósíthatósági elemzése 3. rész. A Paksi Atomerőmű élettartam hosszabbításához tartozó üzleti terv modell kifejlesztése, VEIKI, 2000.

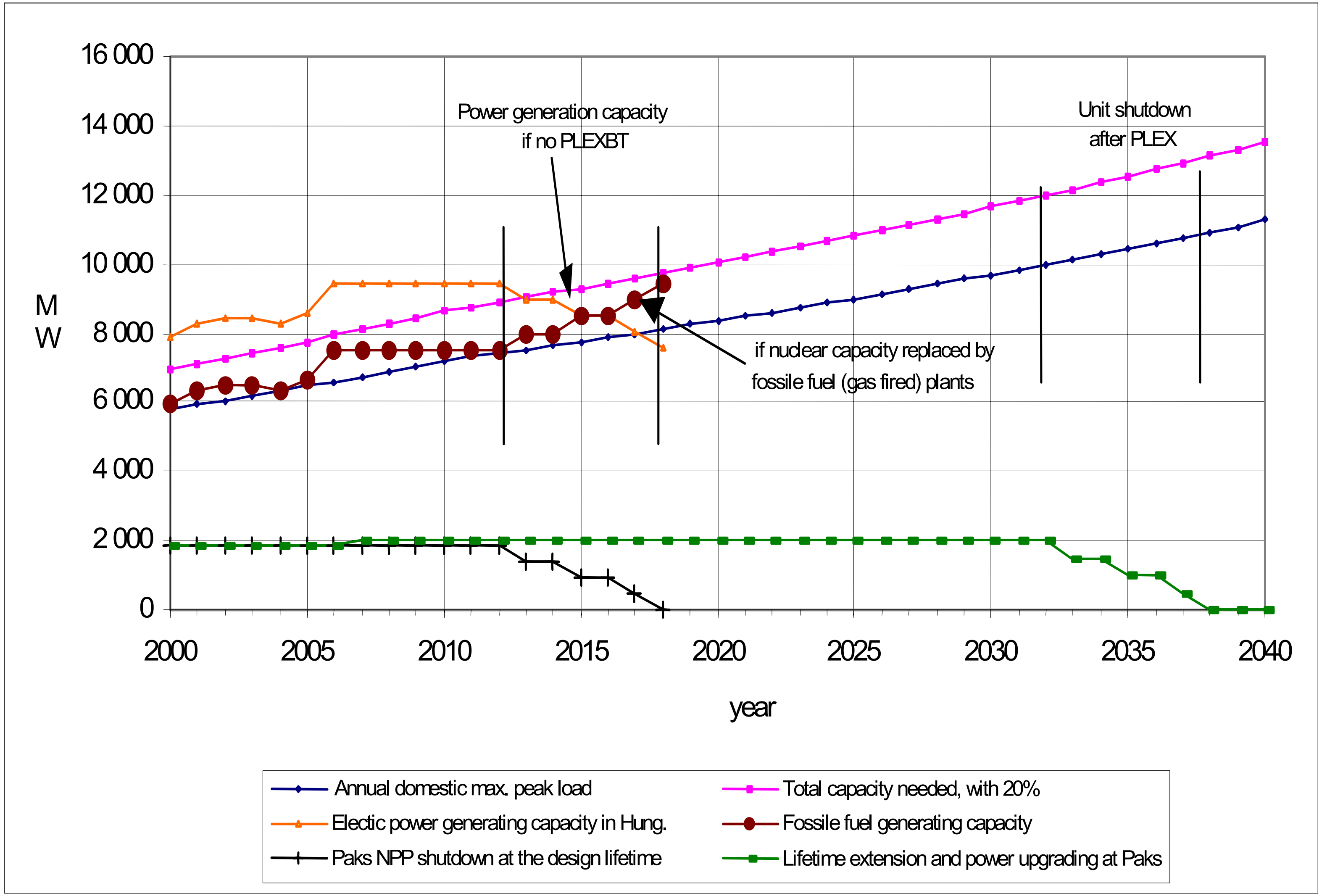


FIG. 1 Electric energy demand and capacity in case of shutdown and lifetime extension

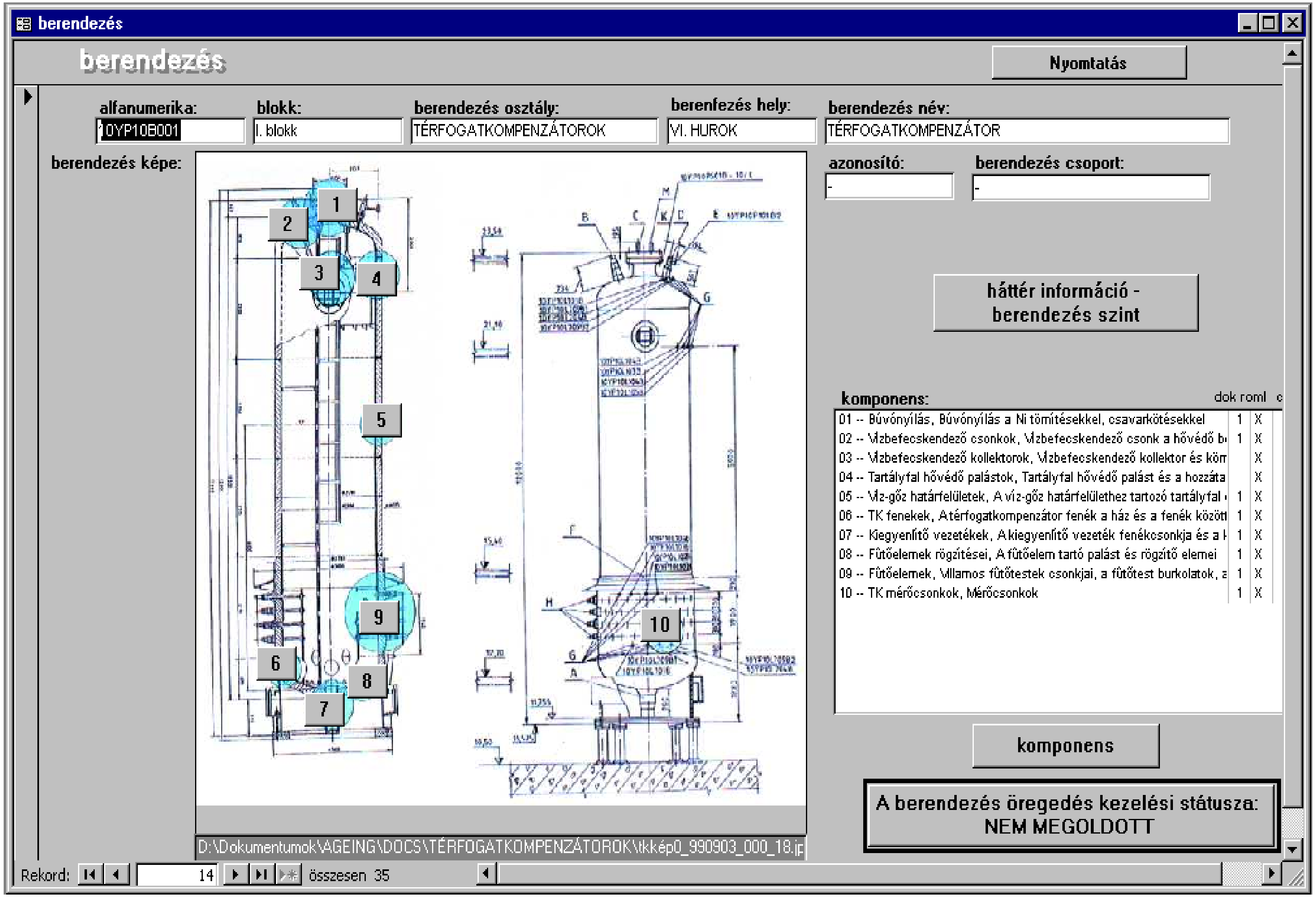


FIG. 2 Computer aid for monitoring ageing processes, pressurizer location of critical elements, identification of the processes

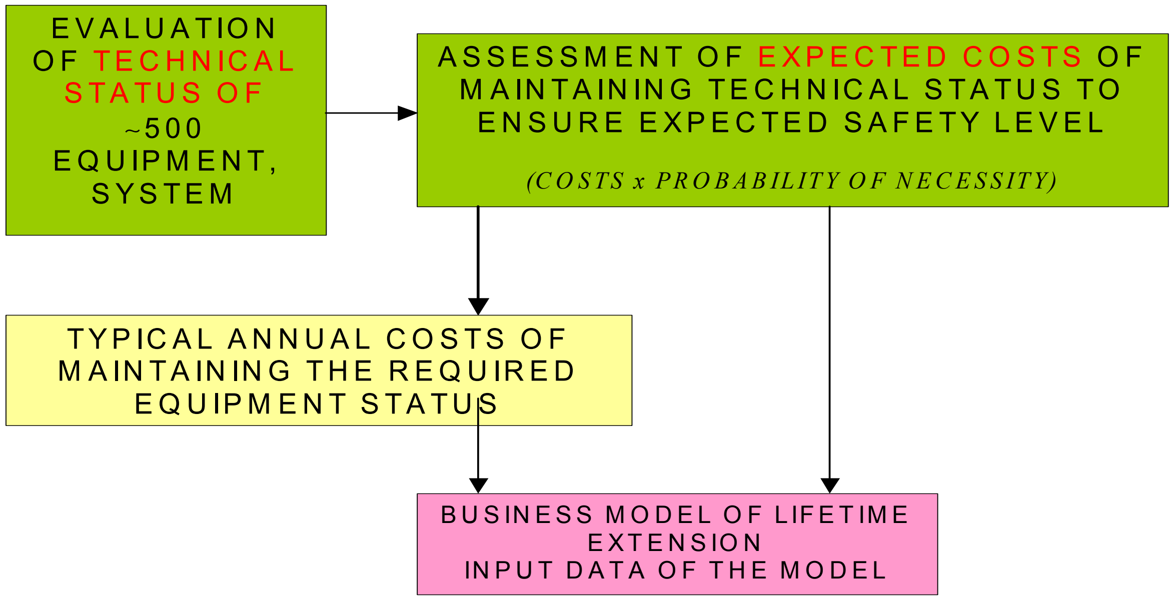


FIG. 3 Process of plant status assessment

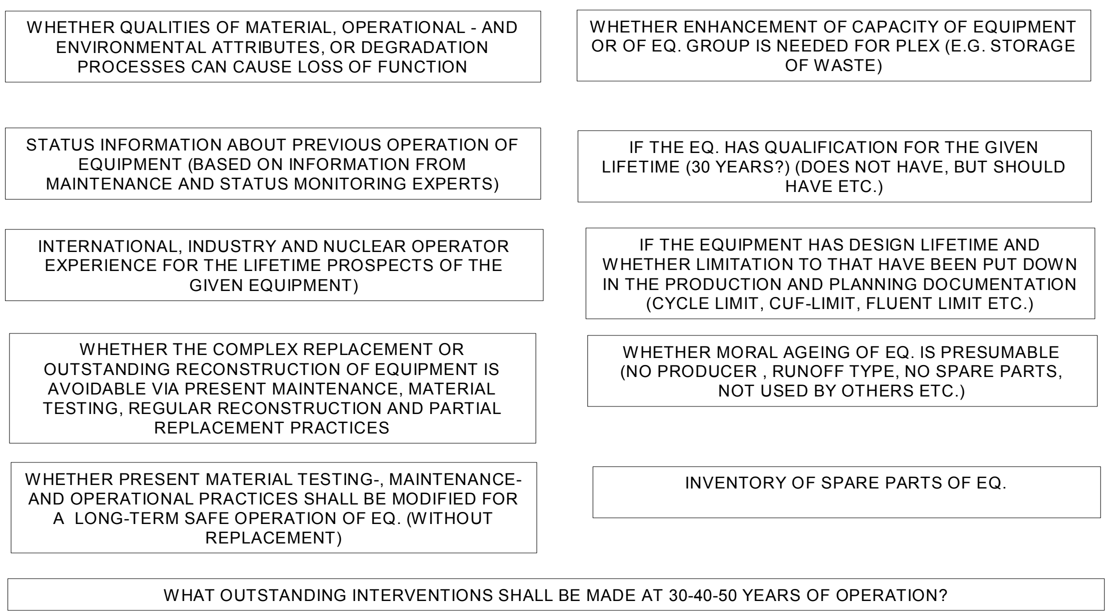


FIG. 4 Aspects of plant status evaluation

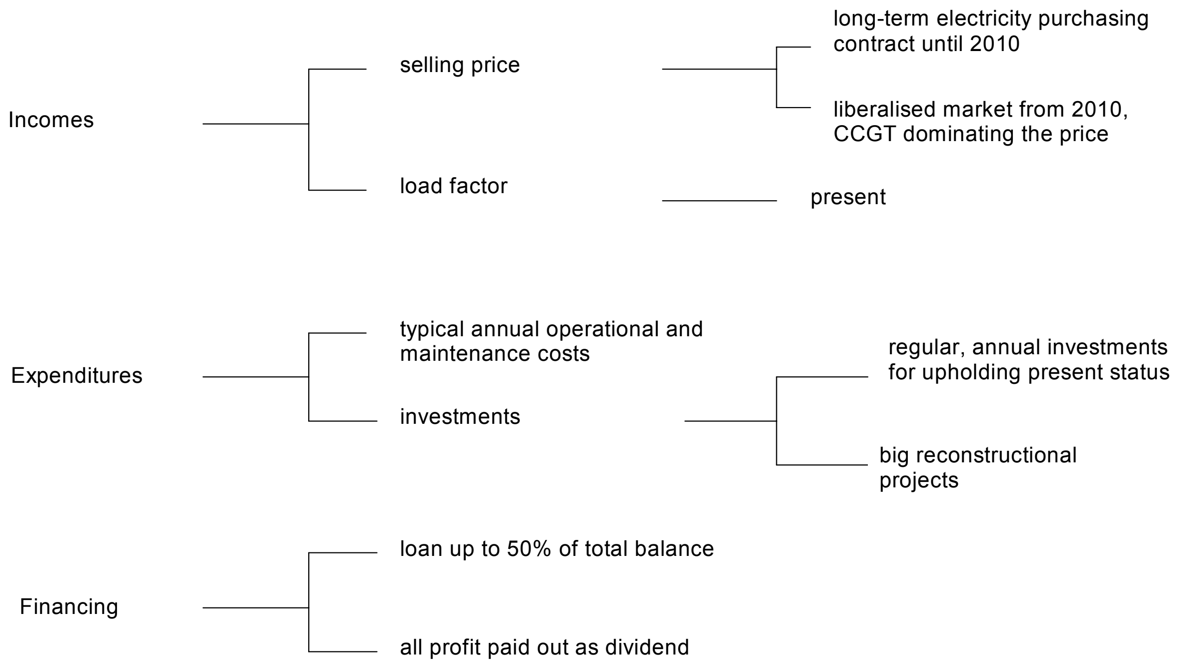


FIG. 5 Business model

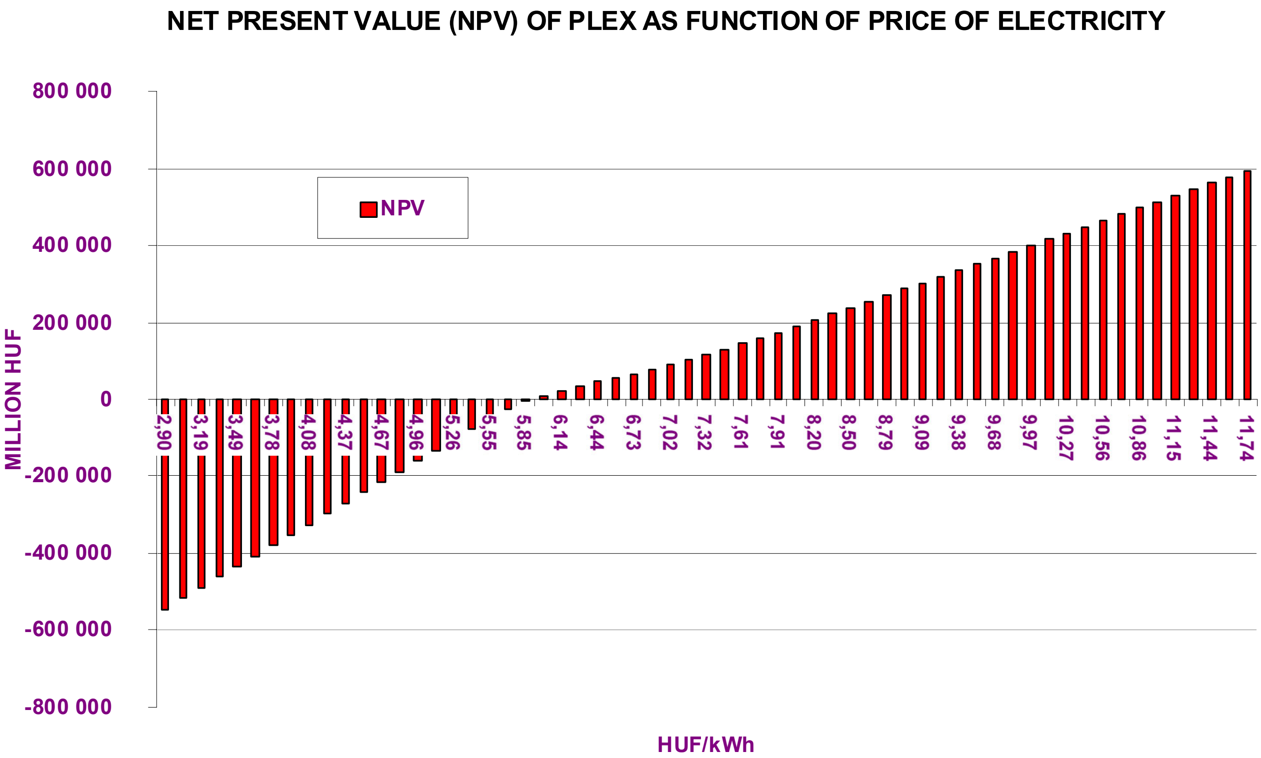
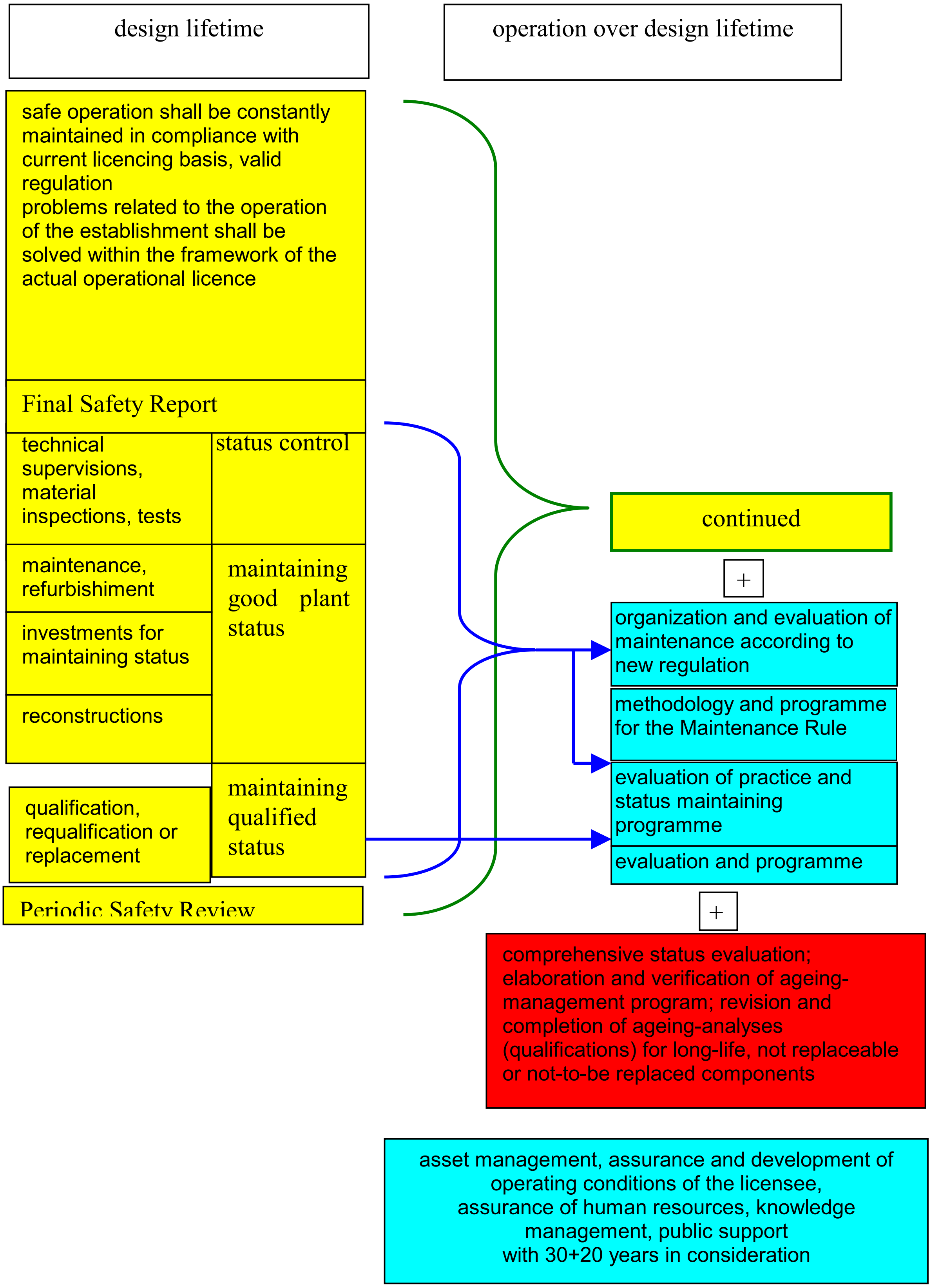


FIG. 6 Net present value of PLEX of 20 years as function of electricity price



Colour code: yellow – normal activities and obligations of the licensee to fulfil conditions of the actual operation licence; blue – a new element in the activity of the licensee related to lifetime extension, competence of organizations of the licensee according to Personnel and Work Regulations, in cooperation with PLEX project; red - a new element in the activity of the licensee related to lifetime extension, implemented within the framework of the PLEX project

FIG. 7 Obligations and basic tasks of the licensee

Nr.	Task	2002.	2003.	2004.	2005.	2006.	2007.	2008.	2009.	2010.	2011.	2012.	2013.-2032.
	Issuance of guidelines, modification of law and of NSR												
	Elaboration of PSR requirements												
	Review of FSR												
	Elaboration of PSR												
	Maintaining plant status												
	Implementation and control of maintenance efficiency												
	Obtaining qualifications, maintaining the qualified status												
0.1.	Completion of methodology and criteria documentation, preliminary definition of the scope, evaluation of data collection												
0.2.	Elaboration of impact studies / environmental licence, site analysing programme												
0.3.	Grounding of application for licence-in-principle for operation over design lifetime												
0.4.	Application for licence-in-principle for operation over design lifetime, address												
0.5.	Licence-in-principle for operation over design lifetime												
0.6.	Fulfilment of conditions of licence-in-principle, implementation of lifetime extension												
0.7.	Completion of application for renewal of operational licence												
0.8.	Application for renewal of operational licence, address												
0.9.	Renewal of operational licence												

FIG. 8 Obligations and tasks of the licensee in the preparation and licensing process

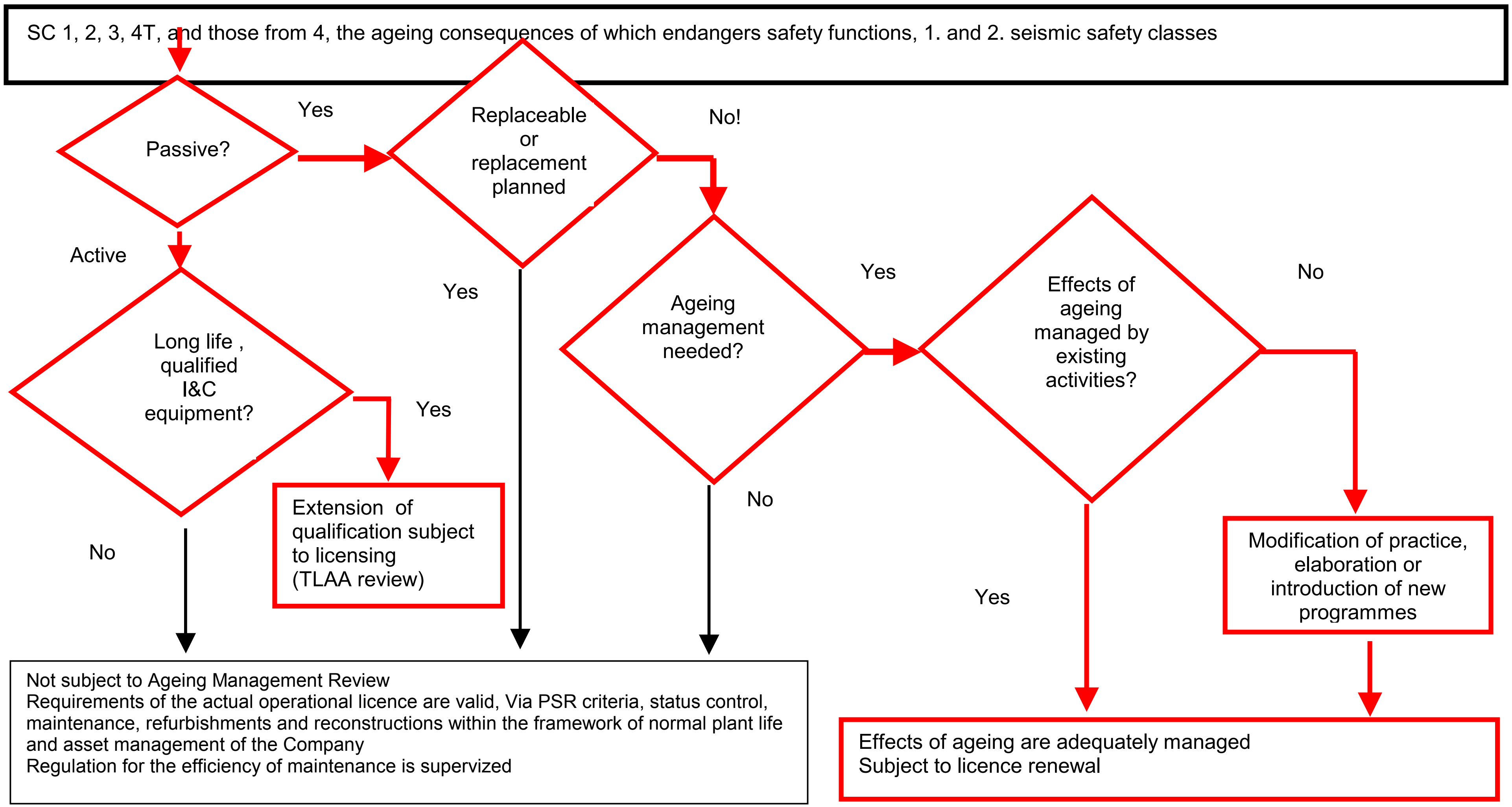


FIG. 9 *Logic tree and scope of preparation project and licence renewal*

PLANT LIFE EXTENSION PROGRAM FOR INDIAN PHWR POWER PLANTS - ACTUAL EXPERIENCE AND FUTURE PLANS

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Abstract

The Nuclear Power Corporation of India Limited (NPCIL) is responsible for design, construction and operation for all nuclear power plants in India. In the earlier Indian PHWRs zircalloy-2 has been used as coolant tube material, which is observed to have useful life of only about 10 full power years. Reactors at Rajasthan-1&2 Madras-1&2 Narora-1&2 and Kakrapara-1 require en-masse coolant channel replacement at least once in their lifetime. For subsequent reactors from Kakrapara-2 onwards the coolant tube material has been upgraded to Zr 2.5% Nb. En-masse coolant channel replacement and other life extension work have been carried out successfully in Rajasthan Unit-2 (RAPS-2). Work for en-masse coolant channel replacement and plant life extension for Madras unit-2 (MAPS-2) has been taken up since January 2002. Since the coolant channel replacement work requires a long plant outage, this opportunity is also used to extend life of existing systems as well as upgradation work. This life extension and upgradation program is based on the results of detailed in service inspection, evaluation of performance of critical equipment, obsolescence and other strategic reasons. This paper discusses in brief the experience of RAPS-2 in carrying out the above jobs as well as the strategies being adopted for MAPS-2 and future reactors.

1. INTRODUCTION

The Nuclear Power Corporation of India Limited (NPCIL) is responsible for design, construction and operation for all nuclear power plants in India. Currently, it has fourteen (14) reactor units under operation. Performance of the units in the last financial year 2001-02 is given in Table-1. NPCIL's overall capacity factor was 85% in financial year 2001-2002, with 96% as the highest capacity factor.

Another eight units are under various stages of construction which will further add a capacity of 3960 MWe as given in Table-2. Two units at Tarapur of 540 MWe PHWR are of new design and four units at Kaiga and Rajasthan are of standard design of 220 MWe PHWR. Two units at Kudankulam are VVER 1000 (PWR) being built with Russian collaboration.

In the earlier PHWRs zircalloy-2 has been used as coolant tube material. Subsequent studies and experience have shown their life to be considerably lower (about 10 full power years) than originally estimated. This means that reactors at Rajasthan-1&2 Madras-1&2 Narora-1&2 and Kakrapara-1 would require en-masse coolant channel replacement in their economic lifetime. Subsequent reactors from Kakrapar-2 onwards would not need this en-masse coolant channel replacement as the coolant tube material has been upgraded to Zr 2.5% Nb.

1.1. Evolution of Plant Design

India has adopted Pressurised Heavy Water Reactors (PHWRs) for the initial phase of its nuclear power program. Rajasthan Atomic Power Station (RAPS) unit-1&2 and Madras Atomic Power Station (MAPS) Unit-1&2 represent old generation of reactors whose designs were evolved in sixties. Moderator dumping is used for shutdown system. Calandaria is housed in vault cooled by air.

In subsequent standardized 220 MWe units, calandria and end-shield are integral components, dump tank is eliminated and water filled calandria vault is introduced. Two fast acting shutdown systems are provided. Numbers of steam generators (SGs) in a unit is reduced to 4 and are of mushroom type with in-service inspection provision. High Pressure injection for emergency core cooling, supplementary control room (SCR), incorporation of computerized systems & programmable logic controllers (PLCs), and visual display units based plant information system are provided. Total segregation of power supplies and cables for the redundant groups is provided. Tertiary cooling loop is provided. These significant design improvements are done to meet current standards.

1.2. Older units needing change

In older reactors Rajasthan Atomic Power Station (RAPS) unit-1&2 and Madras Atomic Power Station (MAPS) Unit-1&2 there is a greater urgency for life extension program as they have seen a long operating life. Rajasthan unit-2 was the first plant where en-masse coolant channel replacement and other upgrades have been carried out. Madras unit-2 is currently undergoing en-masse coolant channel replacement and other life management activities.

Madras unit-1 and Rajasthan unit-1 are slated for coolant channel rehabilitation and plant life extension program in the next phase.

2. IDENTIFICATION OF SYSTEMS AND COMPONENT NEEDING CHANGE

Life extension requirement in a power plant is based on component ageing determined based on in-service inspection results, increase in maintenance frequency, obsolescence, plant performance improvement and safety upgrades requirement. As a long shutdown is required for en-masse coolant channel replacement, major life extension activities are also taken up during this period.

Following are the main focus areas of Plant Life Management (PLIM).

2.1. Coolant tubes

Reactor components in general are the most important areas of concern as far as the PLIM is concerned. Coolant Tube in PHWR is the most affected reactor component. In the old reactors Zircalloy-2 was used as coolant tube material. Also two loose fit garter springs were used to support the coolant tube in the middle. In some channels, the garter springs are observed to have shifted from their design location.

For monitoring health of the channel an in-service inspection (ISI) programme is established. Positions of garter springs are monitored and relocation is done through remote tooling developed indigenously. ‘BARCIS’ has been developed for channel inspection and

‘INGRESS’ for repositioning of garter springs by Bhabha Atomic Research Centre (BARC). These tools have been improved considerably over the years. These tools can be used in a wet channel so that draining of the channel is not required.

Over period of time, due to creep the coolant channel may contact the calandria tube and may becomes brittle due to irradiation and increase in hydrogen pick up concentration in the contacting area. When degradation is observed in many tubes, economical consideration dictates to go for en-masse coolant channel replacement (EMCCR).

In the new reactors Kakrapar unit-2 onwards, the coolant channels are of Zr-Nb and have four tight fit garter-springs to avoid contact between calandria tube and pressure tube. Hence, it is expected that En Masse Coolant Channel Replacement campaign involving long shutdown will not be required. This implies that the planning of PLIM shall have to take into account that only planned annual shutdowns will be available for implementing PLIM program.

2.2. Safety Upgradation

Several important safety upgrades were done to meet latest safety requirement. These include retrofitting of high pressure emergency core cooling system, introduction of supplementary control room, segregation of power supplies and cables, provision of emergency power supply in case of flood situation, and emergency water injection to steam generators & end shields, etc.

2.3. Heavy Water Heat Exchangers

Heavy water heat exchangers are critical components and are designed and fabricated to the highest nuclear safety standards. Any small leak in these heat exchangers may result in tritium leakage, which is a potential radiation hazard. Thus ISI for health of equipment and corrective measures are important. In RAPS lake water is directly used as the ‘cooling water’ for the heat exchangers, which deteriorates the health of heat exchangers. All the heavy water heat exchangers have therefore been replaced with new ones of latest designs. In subsequent plants demineralised water in closed loop is used for the cooling of heavy water heat exchangers and degradation observed is much less and are expected to last for the planned life.

2.4. Control and Instrumentation

This is another important area for PLIM. There has been rapid technology changes in the area of control and instrumentation (C&I) resulting in obsolescence of the items. Upgrades are needed in this area after about ten years of operation.

Rajasthan Station is having design of 60’s. Instrumentation control loops were of 10-50 mA. Fuelling machine control cards were designed with germanium transistors. Electromagnetic relays were used for logic control. All these are obsolete and spares are difficult to obtain. These areas are suitably upgraded.

2.5. Plant performance

Issues related with ‘maintenance’ and ‘systems affecting plant performance’ are compiled over the period of time. These are also addressed during PLIM.

3. MAJOR UPGRADATION DONE FOR RAPS-2

En-masse coolant channel replacement and other life extension work have been carried out in Rajasthan Unit-2 (RAPS-2) during 1996 to 1998. Some of the important jobs carried out are briefly described below.

3.1. En-Masse Coolant Channel Replacement

All 306 Zircalloy-2 coolant tubes have been replaced with Zr-2.5% Nb coolant tubes. The whole activity was carried out by developing indigenous tooling.

The material of new coolant tubes was changed to Zr-2.5% Nb alloy in view of its higher strength, low corrosion susceptibility, less hydrogen/deuterium pick up and higher creep resistance. The tubes were subjected to stringent eddy current and ultrasonic examination.

The existing design with only 'two' loose fitting garter springs between the coolant tube and calandria tube was modified to 'four' numbers of tight fitting garter springs.

The end-fitting assembly design was modified by increasing length of journal ring to accommodate higher axial creep. The rolled joint bores were designed and machined for `Zero clearance' joint at one end and `Low clearance' joint at the other end for the coolant tube end-fitting rolled joints.

A large number of tools were developed for remote operation to minimize the radiation dose. Special tools were required for removal of anti-torque collar assembly, opening of Graylocks, holding the feeders, chip-less cutting of the coolant tubes, rolling of liner tubes, rolling of coolant tubes, induction heating of the end-fitting etc. Tools were also developed for insertion and location of garter springs and guiding and insertion of coolant tube into end fittings in-situ.

In order to rule out any possibility of scratches or damage caused to calandria tubes while removing the coolant tubes, all calandria tubes were subjected to inspection. Special tools were developed for visual inspection by mini camera and for eddy current examination of the calandria tubes. Sag measurement of calandria tubes was carried out by optical means in RAPS-2 and by special slope measuring tool in MAPS-2. Special probe and tools were developed for quickly checking the location of garter springs over the coolant

All tools were designed and developed indigenously and were qualified for the operation by conducting extensive mock-ups.

3.2. Safety Upgradation

3.2.1. Retrofitting of high pressure emergency core cooling system

The existing emergency core cooling system (ECCS) in RAPS-2 consisted of low-pressure moderator water injection/recirculation into PHT system in case of a loss of coolant accident (LOCA). This system could adequately take care of the large breaks in PHT system. However, for small/medium breaks, a longer time is expected for the PHT system pressure to fall down to a value sufficient to enable the moderator injection. During this period void formation may take place in the channel, leading to high fuel temperature. Hence, the retrofitting of the ECCS system has been carried out to mitigate this situation.

The retrofitted ECCS consists of high-pressure heavy water (D2O) injection followed by modified low-pressure long-term moderator water injection / re-circulation at low pressure.

3.2.2. Augmentation of small leak handling system

The small leak handling system is modified such that it now takes water from ECCS tank and is no longer dependent on moderator system. Thus moderator water with high tritium activity will not get mixed up with PHT water.

3.2.3. Inspection and repair of feeder elbows for wall thinning

Thickness measurement of all feeder elbows was carried out on all the 612 feeders in RAPS-2. The results of ultrasonic thickness gauging indicated that feeders, mostly outlet feeders at the first elbow downstream of Grayloc coupling, had undergone reduction in thickness. The wall thinning of feeder elbows may be due to corrosion or flow-induced erosion. The repair work using weld deposit method was carried out where found necessary.

3.2.4 Light Water Dousing System Modification

Dousing system is designed to limit the containment peak pressure within design limits by condensing the steam released, in case of break in the primary pressure boundary. Prior to modification, it consisted of flow modulating feature where dousing flow varies in proportion to the velocity of steam air mixture flowing in the vicinity of dousing curtain. However, it was observed that as dousing initiates, it increases the air velocity in the vicinity which in turn gives signal to increase dousing flow thus leading to a divergent situation such that any initiation of dousing resulting in maximum rate of dousing, making modulating feature ineffective.

It was observed that lower flow rates for a longer duration are more effective to limit containment peak pressure. Studies were carried out and it was found that dousing flow rate of 30% of existing maximum could limit the containment peak pressure within design limits for all primary heat transport (PHT) break sizes.

To take care of the above, the system was modified as follows:

1) Instead of high dousing and low dousing schemes, fixed flow scheme is selected which takes care of all break sizes of PHT system piping.
2) Dousing flow is limited to 30% of earlier maximum capacity. For this purpose, three out of ten dousing lines are retained. Balance seven lines are blocked. Control of dousing valves is changed from modulating type to simple on-off type.

3.2.5. Additional DG Set for handling flood condition

The class-III supply was provided with 2 numbers of diesel generators (DG) sets each for RAPS-1 and RAPS-2 with no interconnection between class–III buses of the two units. The continuous rating of each DG set is 1200 kW. Interconnection is provided between class-III buses of RAPS-1 & RAPS-2 for better availability.

In the event of severe flood condition the above DGs may become unavailable due to submergence. In order to meet the requirement of safe shutdown and core cooling under the flood condition, when the load requirement would be small, an additional DG set of 625 kW, which is air cooled, is provided at a higher elevation.

3.2.6. Segregation of power & control supplies and cables and installation of fire barriers

The segregation for power supplies and cables was improved to avoid common cause failure. Following methodology was adopted to achieve the objective of segregation.

1. All the power & control supply panels were divided into two groups and identified as Group-1 & Group-2.
2. All the safety related loads along with their associated support system/valves were reviewed such that they are also divided group-wise.
3. Scheme for relocation of some of the panels feeding safety related loads was implemented to avoid mix-up of group-1 and group-2 panels and a wall constructed between Group-1 area and Group-2 areas to effect physical separation.
4. Totally diverse routes for cables from Group-1 area to TB and RB-2 and from Group-2 area to TB and RB-2 were identified to avoid criss-cross of safety related cables belonging to two groups.
5. Control cables from field junction boxes /marshalling boxes to control distribution frame (CDF) were segregated group-wise.

3.2.7. Dedicated instrument air supply to essential loads

For essential loads required to be operated after Loss of coolant accident (LOCA) a dedicated arrangement of air supply is made to increase reliability. With this provision non essential loads can be isolated for limiting reactor building pressure rise due to in-leakages of instrument air.

3.2.8. Instrumentation for detecting pressure tube leak

Unlike stations at Narora, Kakrapar, etc, the annulus gap between pressure tube and calandria tube in RAPS-2 is open to calandria vault atmosphere. Hence, it was not feasible to provide an annulus gas monitoring system for sensing any leak from pressure tube.

As an alternative, measuring of dew point of calandria vault atmosphere was reviewed in detail under different configurations of vault dryer in operation and not in operation. It was found that it will take about 4 hours to indicate a serious leak with calandria vault dryer not in operation. When the dryer is in operation, it will take about 8 hours to indicate a serious leak in the calandria vault. This scheme has been implemented at RAPS-2.

3.3. Replacement of heavy water heat exchangers

The heavy water heat exchangers such as moderator heat exchanger, Shutdown cooler, Bleed cooler and Pre-cooler etc undergo periodic eddy current testing (ECT) to determine healthiness of the tubes and degraded tubes of the heat exchangers are plugged. In RAPS-2 all the heavy water heat exchangers have been replaced with new ones. Improvements in the design and selection of materials are carried out as required.

3.4. Control and Instrumentation

3.4.1. Supplementary control room

As part of upgradation, Supplementary Control Room (SCR) was introduced, to bring the station in line with current safety practices. SCR is provided for the purpose of achieving certain critical safety control functions when there is a loss of ability to accomplish these functions from the main control room due to loss of habitability of the main control room.

SCR is located on the first floor of a separate building. It has a supplementary control panel, control power supply and distribution panel and uninterrupted power supply system along with batteries.

Supplementary Control Panel has the following provisions.

1. Manually tripping of the reactor.
2. Opening of 4 nos. of large steam discharge valves.
3. Monitoring of all essential plant parameters like PHT system pressure, temperature, boiler pressure, radiation field in boiler room, boiler room pressure, calandria level, calandria outlet temperature and display of boiler level low/high conditions and reactor building isolation damper positions.
4. Recording of Neutronic parameters, Log N and Log Rate N.
5. Display of plant parameters through a Cathode Rate Tube (CRT) along with a CPU provided in SCR. It receives input from Plant Information System.

Independent sensors are used, for sensing critical plant parameters, like PHT pressure, calandria level etc. In the case of neutronic parameters, it was not feasible to provide separate Ion Chambers and hence isolated signal was generated, using buffer amplifiers. Provision has been made to feed power supply to Ion Chamber and buffer amplifier from SCR, in case of main supply failure.

To achieve independence from main control room, separate power supplies and cables were used for all sensors. For sensing PHT system temperature, it was decided not to make new penetration for installing RTDs since it involves puncturing of pressure boundary. Existing RTDs are wired to temperature transmitters with dual isolated outputs, one for catering to main control room and other for Supplementary Control Room.

415 VAC, Class-III main power supply was taken from different sources to enhance availability. To further improve reliability of power supply, an un-interruptible power supply system (UPS) with battery back up was also provided.

3.4.2. Upgradation of channel temperature monitoring system

Channel Temperature Monitoring (CTM) system of RAPS-2 used moving coil type Indicating Alarm Meters (IAMs) for indication of channel temperature and generation of channel temperature High & Very High Alarms. Each installation of CTM has 306 such IAMs custom made product. The alarm generation mechanism of these IAMs uses Light & Vane principle. The models are obsolete. The maintenance effort required to maintain the large number of meters is high. Hence, it was planned to replace the IAMs of one installation by solid state comparator, called Temperature Alarm Units (TAUs).

Salient features of the upgradation of CTM system are as follows:

- Use of advanced (NAPS type) comparators,
- Replacement of temperature alarm indicating meter by TAU is one to one.

- Minimum disturbance to the original wiring. Original IAM connector retained. Additional wiring through additional independent connector.
- Redundant power supplies, with power supply fail alarm.
- Highly stable reference generator with failure alarm.
- Buffered input to comparators and buffered output for CTM computer.

The Upgradation work has lead to higher reliability and minimum maintenance effort.

3.4.3. Computer based plant information system

RAPS-2 was provided with a Data Logger built with discrete components. This was rudimentary as per current technology. This was replaced by a PC based Plant Information System (PIS) developed within the department.

It is a PC based distributed data acquisition system with colour graphic terminals in Control Room as user interface. The system was designed to integrate the two other major data acquisition systems of Unit-2 i.e. Channel Temperature Monitoring System & Process Disturbance Analyser. Presently the system has 12 nodes networked through an ETHERNET (10 MBPS) LAN.

It has graphic display of mimics, bar chart, trending, and data in tabular form etc. Also it has facility of alarm, disturbance analyser and event sequence recording functions.

Here are some of the highlights of the PIS.

- Physically distributed panels.
- Flexible configuration of RTD inputs – 100 Ohms/200 Ohms software selectable. Lead resistance correction through software.
- Selection of type of thermocouple through software.
- Universal card for RTD & T/C.
- Digital inputs with opto-isolator.
- Provision of digital outputs through relay.
- Multiple 20” PC based colour graphic terminal as user interface.
- Separate PC based terminal for maintenance activities.
- Separate colour graphic terminal in SCR with full capability of control room terminal.
- Information of CTM & PDA available throughout the network.
- Data presentation in current standard.
- A Large number of new inputs have been connected for example radiation monitoring system, turbine supervisory system etc.
- Expandable system to meet the future need.
- Time synchronised with station master clock.

3.4.4. Upgradation of Controllers, Electronic transmitters

In RAPS-2 all single loop electronic controllers used discrete components, which are out of production range of Foxboro for many years. The performance of these controllers was very satisfactory, in past. But due to deterioration of various key components, non-availability of spares due to obsolescence their long term maintainability could not be ensured further. Hence as a part of upgradation, all these controllers were replaced by indigenously developed Digital Programmable Controllers.

While replacing, the basic functional requirements of the control loop have been retained including controller settings viz. proportional band, reset and rate time. In case of simple control loops, the old controller was replaced by new controller after appropriate programming. For complex loops full benefit of powerful features of these controllers was taken and the secondary instruments like high selector, low selector, summing amplifier, square root extractor, remote set station etc. were eliminated by incorporating their functions in the controllers.

The new controllers are having 4-20 mA outputs. Hence electro-pneumatic converter and other related hardware was also changed.

3.4.5. Augmentation of fire detection and alarm system

In view of various fire incidences in RAPS and other plants, particularly NAPS, there was review of adequacy of the fire detection and alarm (FDAS) system. The enhanced number of types of detector recommended by the Review Committee could not be supported by the existing system because of the obsolescence, incompatibility and lack of expandability.

In view of the above constraints, it was decided to install a new centralized FDAS with a total number of approximately 209 different types of detectors covering Reactor Building 2, Turbine Building-2, Service Building, ECCS Building, Fifth DG Building and Switchyard. Various types of detectors were used to suit the type of combustible materials and optimum coverage of the area. The newly installed ionisation, optical and thermal detectors and manual call points are analogue addressable type facilitating quick identification of the area under fire. Non-addressable type detectors like beam detectors and flame detectors also have been used in Boiler and Moderator room.

For better reliability the FDAS has been divided into two independent control panels called Group-A and Group-B panels. Wiring of the detectors has been done to maintain this independence. Class A-type (ring main) wiring with adequate number of fault isolators in each loop has been adopted for detector looping. The incoming and outgoing cables of the loops are routed to physically diverse routes. Each fire zone is covered by detectors of both the groups. These control panels have alphanumeric display for quick identification of fire. A common fire / trouble audio-visual indication has been provided in main control room. A user-friendly supervisory computer with a printer for event logging has been provided in CTM computer room. A remote audio-visual alarm indicator has also been provided in the fire station.

3.4.6. Upgradation of fuel handling controls

a) Logic Cards Upgradation:

Fuel Handling System is the lifeline of PHWRs. The Fuel Handling Electrical Control System of RAPS-2 was designed in the late sixties and consisted of logic cards built with Germanium transistor circuits. Due to obsolescence, increased rate of failure due to ageing and also due to non-availability of equivalent spares, the control system needed renovation. Accordingly, new logic cards were designed using CMOS ICs keeping the same terminal and functional specifications as well as providing value addition features (i.e. improvement in noise immunity, sharp changeover of logic states, more access for maintenance, on board testability, on board I/O isolation and forcing, improved indications through LEDs etc). All this was done keeping the size of the cards identical to the original, since these cards were to be retrofitted within the available space. The job involved development of new prototype cards, manufacture (nearly 1800 cards of 10 different varieties), shop testing, installation and re-commissioning of the entire Fuel Handling Control System at site using these new cards.

Site engineers were inducted from day one when the conceptual designs were discussed in details and finalised. This approach reduced the delays in communication, review and freezing of the design documents. This upgradation activity was successfully completed 3 months ahead of schedule, in just 19 calendar months.

b) Remote Viewing System:

To aid remote viewing of high radiation Fuel Handling areas such as Fuelling Machine Vaults, a Remote Viewing System was provided in the early days. Due to non-availability of spares, the system was not available for a long time. In view of distinct advantages of this system, it was replaced with a new system. While installing the new system, additional critical areas like Fuel Transfer Room, Spent Fuel Inspection Bay etc. were also provided with remote control cameras enabling viewing of these areas also from the main control room.

c) Spent Fuel Inspection Bay Panel:

Spent Fuel bundles discharged from the reactor are received at Spent Fuel Receiving Bay and after confirming (as required) that the bundles are in healthy condition, they are transported to Spent Fuel Storage Bay for long term storage. All the related movements of the actuators at the receiving bay are controlled manually using Spent Fuel Inspection Bay panel, which is common for both the units of reactor. The panel is redesigned with new switches and replaced.

4. MAJOR UPGRADATION CURRENTLY BEING DONE FOR MAPS-2

In Rajasthan-2 it took about 2 years for the EMCCR work. Now with past experience and better tooling the period is being curtailed to about 15 months. During this period other safety upgrades and renewals are also carried out.

Madras Station is located at a coastal site. While reviewing the requirements for upgrades to Madras Unit-2 in view of upgrades done for Rajasthan Unit-2, certain differences in the environmental conditions were noted necessitating in different requirements of upgrades. Like sea water ingress in condenser leads to chloride attack in steam generator. Corrosive coastal environment also leads to faster degradation to the exposed surfaces.

The following upgrades, similar to the ones for RAPS-2, are being done in MAPS-2.

1. En-masse coolant channel replacement (EMCCR)
2. Replacement of thermal shield cooling system heat exchangers
3. Segregation of power and control supplies and cables, and installation of fire barriers
4. Retrofitting of high pressure emergency core cooling system

5. Augmentation of small leak handling system
6. Inspection and repair of feeder elbows for wall thinning
7. Supplementary control room
8. Augmentation of fire detection alarm system
9. Upgradation of channel temperature monitoring system
10. Instrumentation for detecting pressure tube leak
11. Computer based plant information system
12. Upgradation of fuel handling controls
13. Dedicated instrumentation supply to essential loads
14. Replacement of recorders
15. Inspection and replacement of tubes in DNM system
16. Master Clock

The following additional jobs are also being implemented in MAPS-2

1. *Installation of sparger channels:* – This is replacement of moderator inlet diffuser that had failed earlier. With this moderator flow will be normalized. Station will be up-rated back to its 220 MWe capacity.
2. *Replacement of steam generators:* –5 hairpins had tube failures. It is proposed to replace all steam generator hairpins.
3. *Continuous blow down for boilers:* - for better chemistry control of boilers
4. *Boiler steam water sampling system, panel modification:* - old system was difficult to maintain.
5. *Replacement of MG set with 600 KVA UPS:* - Motor generator set was difficult to maintain and also the rating of Class-II supply has been increased.
6. *Replacement of CL-II and III ACBs / switchgear:* - for better maintainability and segregation.
7. *Invertors replacement with 30KVA invertors & ACVR/250 V modification:* -for better maintenance and segregation
8. *Replacement of protective system IAMs with comparator:* - this will replace alarm function of moving coil type indicating alarm meters with solid state comparators to improve reliability
9. *Microprocessor based beetle monitoring system:* - for ease of monitoring and providing better diagnostic features
10. *Replacement of generator exciter with static one:* - to overcome maintenance problems.
11. *Refurbishment of oil centrifuge:* - to overcome maintenance problems.
12. *Introduction of Debris Filter for condenser cooling water:* - to limit damage to condenser tubes.
13. *Replacement & relocation of Diesel operated Fire Water Pumps 1&3 by Vertical Turbine type diesel engine operated pumps:* - to relocate them above flood level
14. *Revamping of Cathodic protection system*: - To improve performance.

5. GENERAL ASPECTS DURING UPGRADATION

5.1. Upgraded systems are designed with current standards requirement

Current standards are applied for the upgradation work to the extent possible wherever feasible keeping in view the space limitation and constraints of interfaces. For example coolant tubes and C&I systems are of new design. But replacement steam generators in MAPS-2 are of existing design.

5.2. Man-rem reduction

Reduction of man-rem is of high priority. This is achieved through use of improved remote operated tools and better training.

Mechanized platform with sliding shield towards reactor face is used in MAPS-2 to reduce radiation exposure.

Chemical decontamination of primary heat transport system is carried out prior to commencement of EMCCR work

Full scale mock up

Extensive mock-ups are done and operators are trained and qualified before starting actual jobs for all the activities.

5.3. Training and re-qualification

The operator retraining for the upgraded systems and equipment was an essential part of upgradation program. All operators were re-qualified.

5.4. Document updating

All the affected documents such as safety report, design manuals and drawings, operating manuals and flow sheets are revised to reflect the changes carried out with revision control procedure.

5.5. Regulatory approach

Safety clearance is required to be obtained from Atomic Energy Regulatory Board (AERB) for all safety related changes. For this a list of the intended changes is submitted and discussed. Design details are submitted and approved by AERB before carrying out the job in field. Usually a specialist task force/committee is also assigned to follow up the upgradation activities in field to suggest any improvements for safety requirements.

All the Safety applications undergo the review process involving the following four stages.

- Station Operation Review Committee (SORC) – (Station Committee)
- NPC-Safety Review Committee (NPC-SRC) – (Utility Headquarter Committee)
- RAPS-MAPS Safety Committee (RMSC) – (AERB Technical Committee)
- Safety Review Committee for Operating Plants (SARCOP) - (AERB Senior Committee)

AERB Board is the licensing authority.

6. FUTURE PLAN

Experience of earlier work is taken into account to improve upon PLIM program for the subsequent plants. With data collected through ISI program the life of equipment can be predicted better.

RAPS-1 and MAPS-1

RAPS-1 has completed 6.5 EFPY and is shutdown for addressing safety issues. The upgrades will be carried out in line with those done for RAPS-2.

In MAPS-1 unit extensive inspection and repair work has already been carried out. Full scale upgrades are envisaged after 10.5 EFPY

TAPS (BWR)

Design basis and Safety Analysis review for TAPS-1&2 have been recently carried out. Modifications are being planned accordingly.

Narora unit-1 onwards

Planning is continuous process. In NAPS already Computerized Operator Information System, Fuelling Machine Control computer, adjusters rod servos have been replaced. Also replacements of CTM computer, moderator heat exchangers and standby coolers have been planned.

Methodology

In RAPS-2 EMCCR work was done in house. For MAPS-2 the field work is being done through contractor. In later plants, as the in-house activities will be increasing due to more number of plants, there may be a need for contracting theoretical and analysis work also outside.

7. SUPPORT AT HEADQUARTERS FOR PLIM PROGRAM

Starting from RAPS-2 upgradation a dedicated group at headquarters has been established to extend design and procurement support for the PLIM program.

The group's role is:

1. Review operational feedback from Operating Stations; identify areas for improvements and priorities in consultation with Operations Group at headquarters and Stations.
2. Engineer suitable design modifications for enhancing the operating performance, safety, including radiological safety, and the reliability of the stations
3. Review, redesign and carry out re-engineering related to upgradation of components, equipments and systems during major shutdowns.
4. Extend support to stations for regulatory clearances
5. Procure critical items

8. CONCLUSION

Nuclear Power Corporation of India Limited is committed to continuous improvement of safety, reliability and operability of all its operating plants. It has already carried out major upgradation in RAPS-2 and has taken up upgradation of MAPS-2. Upgradation of subsequent plants is envisaged on similar line.

Table 1. Performance of Nuclear Plants

S No.	Reactor Unit	Rated Capacity	Type	Commercial operation	Capacity factor 2001-02	Million Units generated up to July 2002 since commercial operation
1	Tarapur-1	160 MWe	BWR	Oct-1969	85	30608 MUs
2	Tarapur-2	160 MWe	BWR	Oct-1969	94	30513 MUs
3	Rajasthan-1	150 MWe	PHWR	Dec-1973	21	11445 MUs
4	Rajasthan-2	200 MWe	PHWR	Apr-1981	86	21032 MUs
5	Madras-1	170 MWe *	PHWR	Jan-1984	85	18479 MUs
6	Madras-2	170 MWe *	PHWR	Mar-1986	76	16192 MUs
7	Narora-1	220 MWe	PHWR	Jan-1991	92	13133 MUs
8	Narora-2	220 MWe	PHWR	Jul-1992	81	13028 MUs
9	Kakrapar-1	220 MWe	PHWR	May-1993	89	11835 MUs
10	Kakrapar-2	220 MWe	PHWR	Sep-1995	96	10973 MUs
11	Kaiga-1	220 MWe	PHWR	Nov-2000	76	2608 MUs
12	Kaiga-2	220 MWe	PHWR	Mar-2000	80	2608 MUs
13	Rajasthan-3	220 MWe	PHWR	Jun-2000	74	3154 MUs
14	Rajasthan-4	220 MWe	PHWR	Dec-2000	84	2462 MUs

* Note: MAPS-1 & 2 Derated to 170 MWe from 220 MWe w.e.f. 8 April 96 due to moderator diffuser breakage in both units. Capacity is planned to be regained after installing sparger system.

Table 2. Nuclear Plants under Construction

S No.	Reactor Unit	Rated Capacity	Type	First Pour of Concrete	Expected commencement of commercial operation
1	Tarapur-3	540 MWe	PHWR	May-2000	Jan-2007
2	Tarapur-4	540 MWe	PHWR	Mar-2000	Apr-2006
3	Kudankulam-1	1000 MWe	VVER	Mar-2002	Dec-2007
4	Kudankulam-2	1000 MWe	VVER	Mar-2002	Dec-2008
5	Kaiga-3	220 MWe	PHWR	Mar-2002	Mar-2007
6	Kaiga-4	220 MWe	PHWR	May-2002	Sep-2007
7	Rajasthan-5	220 MWe	PHWR	Oct-2002	Aug-2007
8	Rajasthan-6	220 MWe	PHWR	Oct-2002	Feb-2008

Abbreviations used:

C&I	Control and Instrumentation
CDF	Control Distribution Frame
CTM	Channel temperature Monitoring
DG	Diesel Generator
DNM	Delayed Neutron Monitoring
EFPY	Effective Full Power Years
EMCCR	En-Masse coolant Channel Replacement
FDAS	Fire detection and Alarm System
IAM	Indicating Alarm Meter
ISI	In-Service Inspection
KAPS	Kakrapar Atomic Power station
LAN	Local Area Network
MAPS	Madras Atomic Power Station
NAPS	Narora Atomic Power Station
NPCIL	Nuclear Power Corporation of India Limited
PC	Personal Computer
PHT	Primary Heat Transport
PHWR	Pressurized Heavy Water Reactor
PIS	Plant Information System
PLC	Programmable Logic Controller
PLIM	Plant Life Management
RAPS	Rajasthan Atomic Power Station
RTD	Resistance Temperature Detector
SCR	Supplementary Control room
SG	Steam Generator
T/C	Thermocouple
TAU	Temperature Alarm Unit

CURRENT APPROACHES TO NUCLEAR POWER PLANT LIFE MANAGEMENT IN JAPAN

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Abstract

More than 30 years have passed since the first commercial operation of a nuclear power plant in Japan. Concerns regarding the progress in the aging of such nuclear power plants have increased, making the provision of countermeasures against such aging a matter of immediate importance. Technical evaluation of aged nuclear power plants shall be carried out in a “Periodic safety review” some time prior to the plant reaching 30 years of operation from its turn over date. Long-term maintenance program shall be carried out in accordance with the long-term maintenance plan established from the aforementioned technical evaluation. Re-evaluation of the long-term maintenance plan shall also be carried out in a “Periodic safety review” every 10 years.

1. INTRODUCTION

In April 1996, the Agency of Natural Resources and Energy of the Ministry of International Trade and Industry (MITI: now the Nuclear and Industrial Safety Agency (NISA) of the Ministry of Economy, Trade and Industry (METI))(1) issued a report entitled "Basic Policy on Aged Nuclear Power Plants". The report stated that the electric utility companies shall conduct technical evaluations prior to plant's reaching 30 years operation from its turn over date, and should establish maintenance plans appropriate for the followed operation (i.e., long-term maintenance plan). It is also important to enhance inspections and to develop technology for aged nuclear power plants.

(Note 1) In 2001, the Ministry of International Trade and Industry (MITI) was reorganized into the Ministry of Economy, Trade and Industry (METI). The Nuclear and Industrial Safety Agency (NISA) of the METI was established in 2001

In November 1998, the Nuclear Safety Commission (NSC) accepted the MITI's report. The Commission also recommended effectively evaluating and including the technical evaluations into the periodic safety reviews in order to conduct the followed activities for the aged nuclear power plant systematically and periodically.

In February 1999, the MITI reported that the technical evaluations and long-term maintenance plans carried out by the utilities are properly evaluated for the three aged power plants (Tsuruga-1, Mihama-1 and Fukushima Daiichi-1), and issued the report entitled "Evaluation on Countermeasures for Aged Nuclear Power Plants Carried out by Electric Utility Companies and Concrete Plan for Aged Nuclear Power Plants". At the same time, the MITI addressed to incorporate the technical evaluations of aged nuclear power plants into their periodical safety reviews from then on.

In June 2001, the NISA of the METI deemed the reports appropriate for the technical evaluations and long-term maintenance plans carried out by the utilities as the periodic safety reviews for the two nuclear power plants (Mihama-2 and Fukushima Daiichi-2).

The utilities are now carrying out the technical evaluations and formulating long-term maintenance plans for their four plants (Takahama-1&2, Shimane-1, Genkai-1).

2. OUTLINE OF UTILITIES' EVALUATION

The utilities shall carry out 'technical evaluations on aging phenomena' and shall establish 'long-term maintenance plans (identified additional maintenance measures)' for all of components and structures with safety functions before 30 years of operation from the turn-over date.

2.1 Contents of Technical Evaluation

Assuming 60 years of commercial operation, the integrity of the components and structures important to safety shall be evaluated, taking aging phenomena into account (thinning, cracking, material degradation, insulation deterioration, etc.).

The evaluation reports prepared by the utilities confirmed that by continuing the current maintenance practices for the reactor pressure vessels and other components, no technical issues would be presented if the plants continued to operate up to 60 years.

2.2 Contents of Long-term Maintenance Plan

Considering their current maintenance practices (such as operation monitoring, daily patrols and inspections, periodical tests and examinations, refurbishment and replacement as preventive maintenance, and countermeasures for prevention of reoccurrence of non-conforming examples), the electric utility companies shall identify items to be enforced or enhanced, and shall establish a plan which they should then implement.

For example, the items for concrete structures and cables are identified for enhancement of inspections. These items are to be incorporated into the maintenance plan from now on and implemented as a new maintenance program.

3. NISA's EVALUATION

The NISA of the METI receives to the opinions and the advice of academics and specialists, and then evaluates the results of the technical evaluation and the long-term maintenance plans carried out by the utilities. The NISA then reports the results of these evaluations to the NSC.

The NISA will re-evaluate aging countermeasures of the nuclear power plant performed by the utilities in a "Periodic safety review" every 10 years.

The NISA of the METI will evaluate the results of the technical evaluation and the long-term maintenance plan performed by the utilities from the following point of view.

- To be performed analysis of aging phenomena and aging effects using the latest information and knowledge.

  (Example of technology: Application of the test and research results such as the environmental fatigue test, under simulated operating conditions, and the latest information and knowledge obtained from non-conformities of operating experiences.)

- To be evaluated: necessity of additional maintenance countermeasures and efficiency of the current maintenance activities based on the above analysis results.

- To be endeavored to develop technology related to aging by the utilities (identification of technical issues to be developed and adequate application of development results to the maintenance plan).

- To be incorporated immediately: into the maintenance activities to improve reliability from now on, when valuable R&D results, information and knowledge for aging phenomena and aging effects are obtained from domestic and oversea plants.

- To be re-evaluated: the aging countermeasures when the periodic safety review is to be carried out every 10 years.

4. RESEARCH AND DEVELOPMENT

The report, "Basic Policy on Aged Nuclear Power Plants" issued April 1996 indicates that the technical issues to be developed are inspection and monitoring, evaluation methods for aging, and preventive maintenance and refurbishment technology.

Recognizing the importance of this research and development, the NISA of the METI carried out the following research:

- Inspection and monitoring
- Reconstitution technology of the surveillance test sample of reactor pressure vessels
- Evaluation methodologies for aging
- Verification of allowable flaw criterion for vessels and piping to failure despite of propagation
- Enhancement of evaluation methods for environmental fatigue under reactor water conditions
- Enhancement of evaluation methods for neutron irradiation embrittlement of low alloy steel at the upper shelf energy region and thermal embrittlement of

duplex stainless steel

- Establishment of assessment method regarding the neutron irradiation embrittlement of low alloy steel in the upper shelf region and thermal embrittlement of duplex stainless steel
- Research on initiation and propagation of irradiation assisted stress corrosion cracking
- Research on propagation of nickel based alloy stress corrosion cracking
- Research on aging evaluation technology for cable
- Preventive maintenance and refurbishment technology
- Surface improvement technology for reactor internal structures
- Repair welding technology for irradiated reactor components

FIG. 1 Background for approach of the aged nuclear power plants

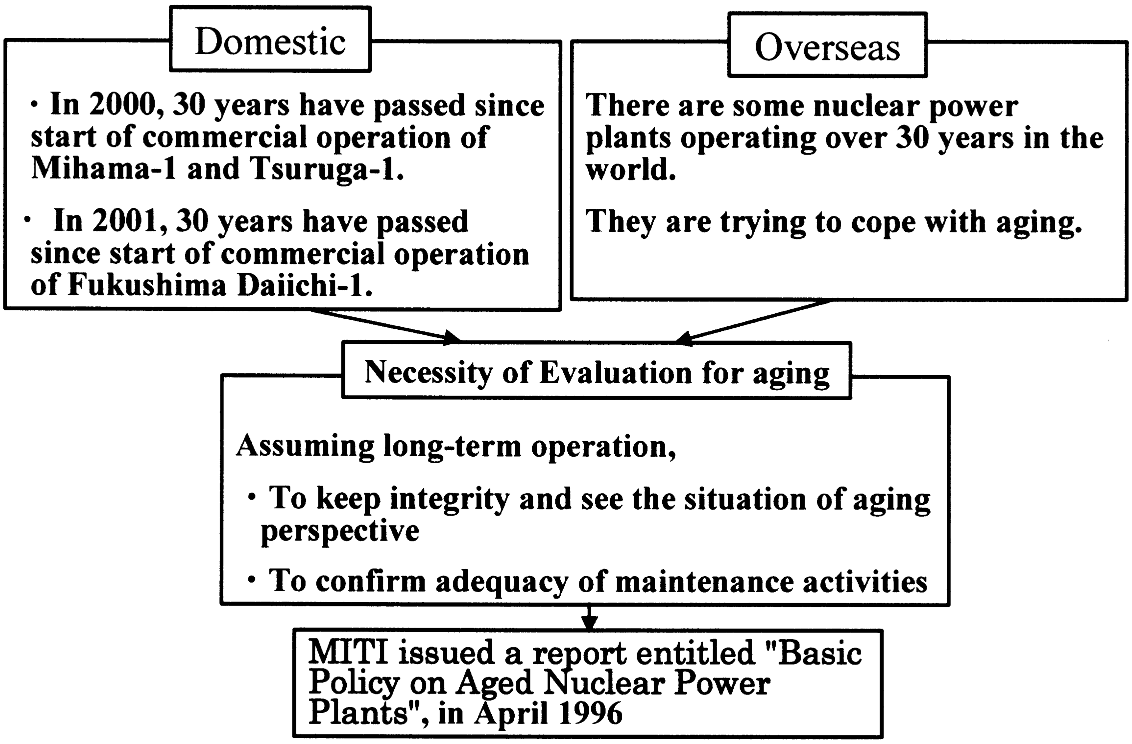
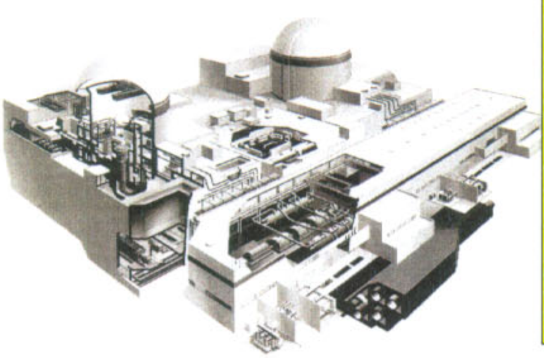


FIG. 2 Components and Structures subject to Evaluation
(Example of PWR)

Components and structures, having safety functions, in the "Review Guideline relating to the Significance Category of the Safety Functions for Light Water Cooled Power Reactor Facilities" (Directive of the Nuclear Safety Commission dated August 30, 1990) are defined as the items subject to evaluation.



1. Pump (Example:Primary Coolant Pump)
2. Heat Exchanger (Example:Steam Generator)
3. **Pump motors (Example: Seawater pump motor)**
4. **Vessels (Example:Reactor vessel)**
5. **Piping (Example: Primary Coolant Piping)**
6. **Valves (Example: Steam Control Valve)**
7. **Reactor internals**
8. **Cables (Example: Power cables)**
9. **Electric equipment (Example: Generator)**
10. **Turbine (Example:High pressure turbine)**
11. **Concrete and steel structures**
12. **Instrumentation and control system**
13. **Air conditioning system (Example: CV Circulation Fan)**
14. **Mechanical equipment (Example: Fuel Handling Facility)**
15. **Electric power source (Example: Emergency DG)**
16. **Other systems (Example: Liquid Waste Evaporator)**

FIG. 3 Flow for Evaluation of Aging Countermeasures

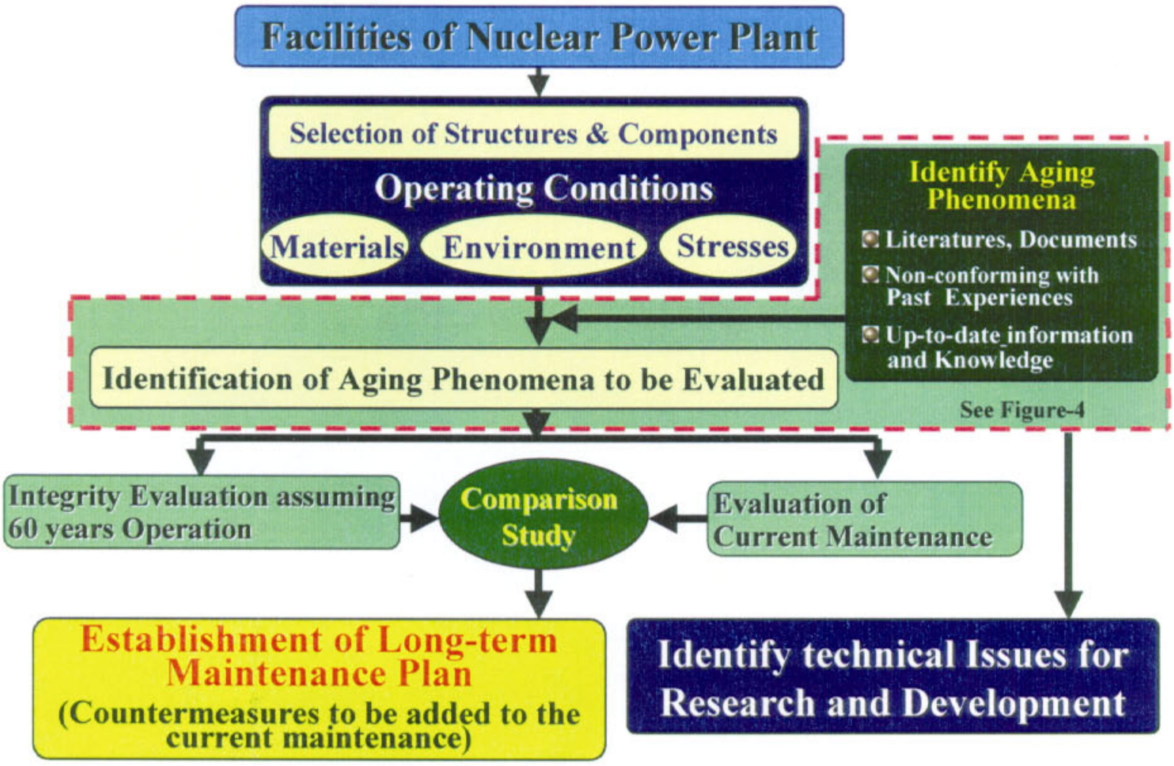


FIG. 4 Identification of Aging Phenomena to be Evaluated

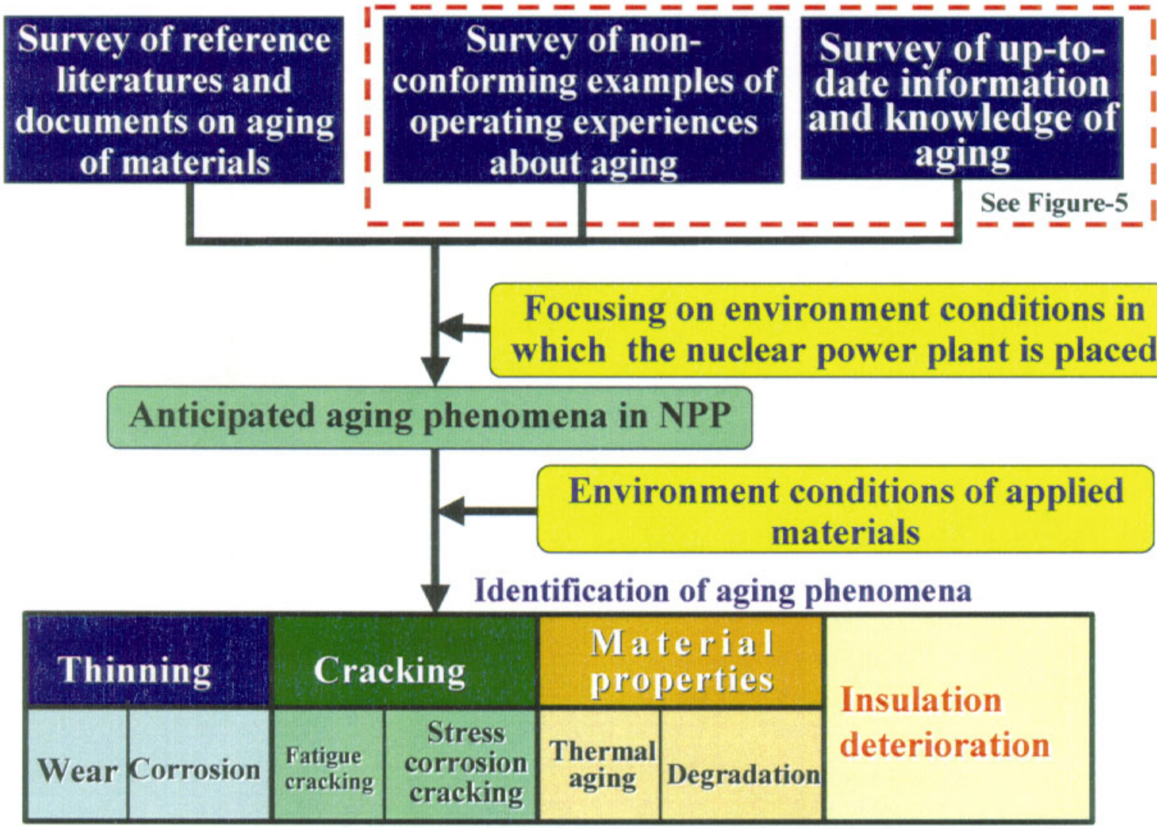


FIG. 5 Mihama-2 evaluation of operating experiences and application of up-to-date information and knowledge for aging

Evaluation of aged nuclear power plants is carried out, being considered lessons learned from operating experiences (excluding design and fabrication issues) and the latest information and knowledge obtained from domestic and oversea plants for aging. For evaluation of Mihama-2, major latest knowledge obtained after evaluation of Mihama-1 is as follows;

Incidents and up-to-date information and knowledge	Mihama-2 evaluation of the aging described in the report	Countermeasures for maintenance plan
Mihama #2 Leak from excess water let-down piping (April, 1999)	**Provided that main flow in the reactor coolant loop piping is hotter than a branch pipe that branches off downward, hot water penetrates into the branch pipe down to the elbow where it locates at about 9D and corresponds to occur thermal stratification. Thermal stresses fluctuation and fatigue crack then occurred. The branch pipe was replaced with 6D length to the elbow. No hot water penetration or thermal fluctuation by temperature measurement was then confirmed.**	**It is believed to prevent damage due to thermal fatigue caused by mixing of hot and cold water. This pipe shall be inspected periodically by hydraulic test.**
Tsuruga #2 Leak from regenerative heat exchanger connecting pipe (July, 1999)	**This incident caused high cycle thermal fatigue due to mixing hot bypass flow outside the inner shell and main cold flow at the junction. However, the component in the Mihama #2 case has a low possibility of high cycle fatigue damage because there is no mixing portion and it has no inner shell. UT also confirmed no indication of flaw.**	**It is believed there is a low possibility of high cycle fatigue damage, because the regenerative heat exchanger has no inner shell and no mixing portion as in Mihama #2 case.**
Tsuruga #1 Stress corrosion crack in shroud support (January, 2000)	**This incident caused stress corrosion crack in the weld of the shroud support. This crack propagated during long-term operation. However, there is no such facility in the case of Mihama #2.**	**Based on the event of Tsuruga 1 and a viewpoint of direct inspection, UT at the inserted portion of high-pressure turbine blades shall be performed in addition to visual inspection of the blade-ring bolts prudently.**
V.C. Summer Crack in the welding joint of reactor vessel nozzle (October, 2000)	**It is revealed tat this incident caused stress corrosion crack due to repeated weld repairs during fabrication. In case of Mihama-2, they evaluated material of Inconel 600 alloy. They replaced the vessel head, and water-jet peening on the in-core flux monitor guide tube as a preventive maintenance. They plan to inspect them periodically.**	**As to the RPV inlet and outlet nozzle, the utilities will continue the current maintenance practices and perform UT periodically and then establish a plan such as repair or preventive maintenance based on the test results.**
Application of guideline to environmental effects of fatigue life	**Tentative evaluation equations shall be applied for components formed from carbon steel, low alloy steel and austenitic stainless steel for light water reactors. This is based on interim results of the research conducted by JAPEIC under the auspices of the NISA. The evaluation confirmed it is within the criteria.**	**Confirmation of a number of experienced operating cycles is planned to monitor and evaluate fatigue life for long-term operation.**

FIG. 6 Major countermeasures for aging (Mihama-2)

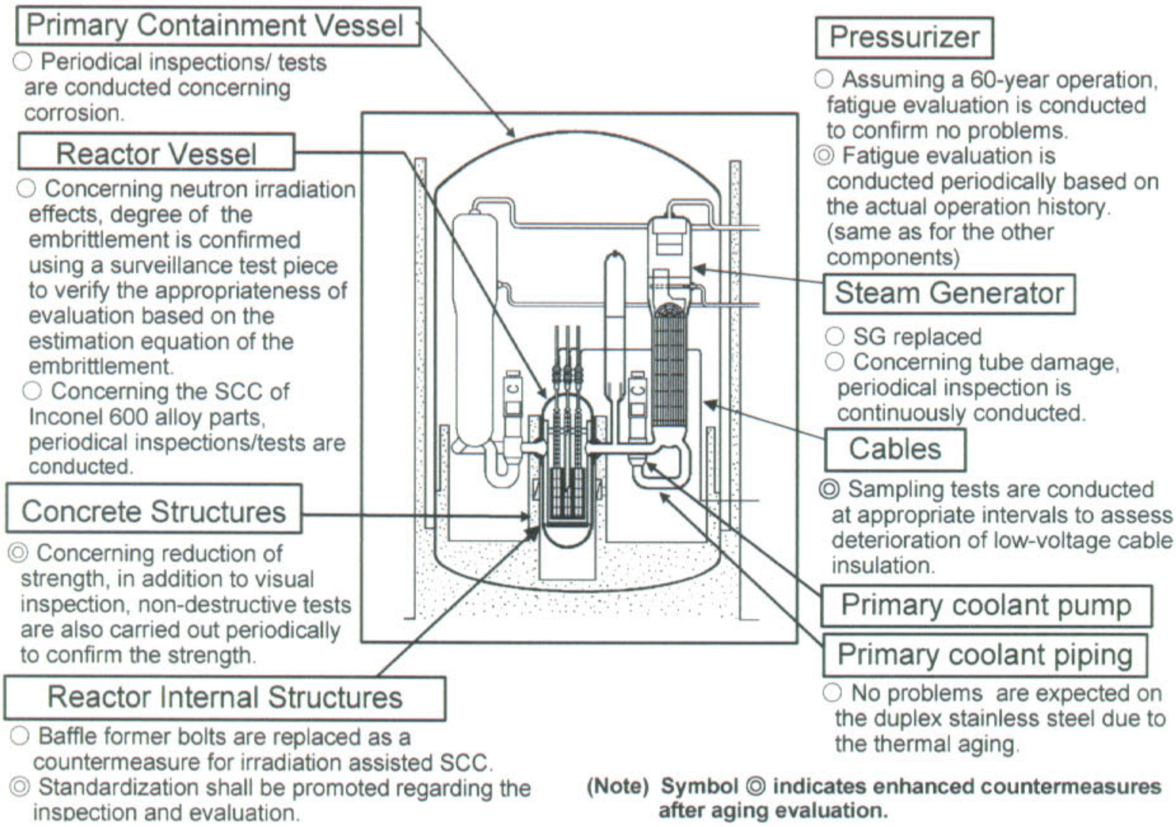


FIG. 7 Relationship between 'Periodical Safety Review' and 'Evaluation of Countermeasures for Aged Nuclear Power Plants'

- **As for countermeasures for the aged nuclear power plants of Mihama-2 and Fukushima Daiichi-2, technical evaluations and establishment of long-term maintenance plan based on the evaluations were carried out as a part of the Periodical Safety Review.**
- **A new maintenance measures (long-term maintenance plan) based on the technical evaluations shall be implemented systematically during the planned annual inspection.**
- **The evaluation and establishment of long-term maintenance plan shall be re-evaluated periodically (approximately every 10 years) as Periodical Safety Review from now on.**

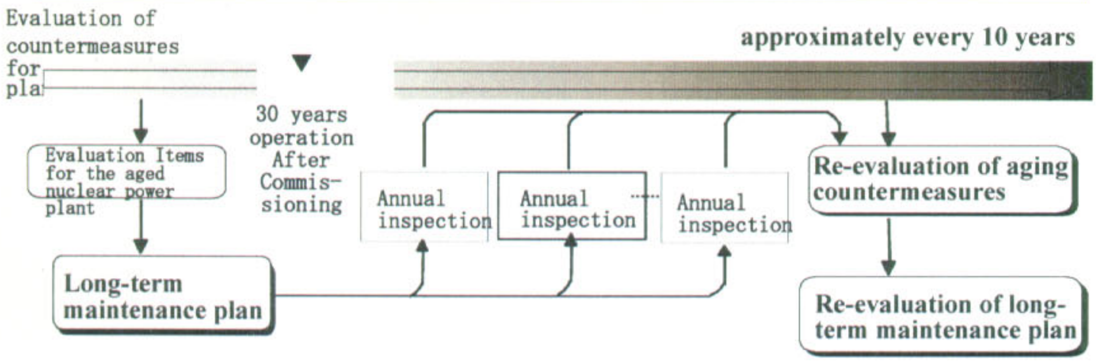


FIG. 8 Status of Technical Development relating to Aging (National Project)

Project Name	Outline
Technical Surveys related to Aging Countermeasures	· Developments of assessment technology of irradiation-induced stress corrosion cracking, assessment method for SCC, propagation of nickel-base alloy and assessment technology for aging affect for cables are performed.
Nuclear Power Plant Integrated Safety Management Technology	· Planning of enhancement of integrity assessment technology regarding neutron irradiation embrittlement of reactor vessel as well as development of toughness surveillance technology as the aging surveillance technology of aged plants. Integrity assessment technology for thermal embrittlement of primary system boundary piping is to develop as well.
Environmental Fatigue Test	· Developing fatigue assessment method in reactor operating environment by environmental fatigue tests under simulated operating conditions, thus survey fatigue behavior of component and piping under actual environment of operating nuclear power plants.
Integrity Assessment of Flaws in Structural Discontinuities	· Establish residual stress assessment and crack propagation assessment methods for complicated configuration part of nuclear power plants during in-service period and verify integrity of components and piping by model verification tests.
Structural Assessment of Flawed Equipments	· Fracture mechanical tests and analysis were performed on the behavior of small defects assumed in the structures of nuclear power plants and verified that the integrities of components and piping could be kept during in-service period even though there exist small defects.
Development on Standards and Guides for Up-graded Inspection System on Nuclear Power Plant	· Enhancement of non-destructive test technologies, such as ultrasonic examination, which is significant inspection technique for confirming the safety of nuclear power plants in Japan, will be attempted.
Repair Welding Technology of Irradiated Materials	· Remedy technology is established by finding out appropriate welding method, welding conditions, threshold of neutron irradiation upon welding work to the materials for reactor vessel and internals irradiated by neutrons simulating long-term operation.
Nuclear Power Plant Maintenance Technology	· To prevent occurrence of SCC of primary enhanced part, performance test is conducted using test piece system components for PWR and reactor internals for BWR, effective surface performance enhancement technology is selected and work integrity and enhanced part performance test is conducted using test piece simulating actual component/part (including neutron irradiation test piece).
Plant Maintenance Technology reliability Verification Operation	· Replacement work methods for ICM housing, shroud, CRD housing/stub tube for BWR and reactor internals, in-core instrumentation guide tube for PWR are verified by full-scale mockup test, etc.

Note: As of July 2001

FIG. 9 Ongoing Aging Related R&D (Joint Researches of Utilities)

Project Name	Outline
Research on austenite stainless steel piping O_2 SCC evaluation (1998-2001)	* Acquisition and evaluation of basic data for stainless steel SCC at high temperature and high dissolved oxygen in piping (stagnant portion, etc.) under PWR environment conditions. The parameters are materials, water quality, testing temperature and welding methods.
Research on non-destructive aging diagnostic technology for cable (2001-2002)	* Development and application of non-destructive diagnostic technology to evaluate cable integrity in place of the current sampling method. (diagnosis by insulator indenter modulus or specific compressive elasticity)
Research on aging related concrete integrity in nuclear power plants (2001-2004)	* Acquisition of basic data for the effects of irradiation on toughness and shielding performance of the concrete structures.
Research on evaluation of high cycle thermal fatigue at the portions mixing with hot and cold water (2000-2001)	* Accumulation of data regarding thermal fatigue at the portion mixing with hot and cold water (effects of temperature differences, frequency of cycle, etc.) and research on evaluation methods. (Conducted as a feedback from the heat exchanger problem at The Japan Atomic Power Co., Ltd.
Research on integrity evaluation for electrical appliances and instrumentation at nuclear power plants (2000-2001)	* The integrity of electric equipment and instrumentation employed at nuclear power plants for a long time is evaluated with prediction of future integrity.
In the planning stage. (BWR)	* An equation to predict neutron irradiation embrittlement in pressure vessels is studied in the USA on the basis of the conventional empirical equation, considering the effects of neutron fluence on embrittlement mechanism. The appropriateness of their equation is, therefore, verified.

Note: As of July 2001

PLIM PROGRAMS OF KOREAN NUCLEAR POWER PLANTS

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Abstract

Twenty-three years have passed since Kori Unit 1, the oldest nuclear power plant in Korea, started to operate in 1978. To mitigate aging phenomena of nuclear power plants (NPPs) operated for a long time, nuclear power plant lifetime management (PLiM) programs have comprehensively assessed current physical condition of operating plants by the ways of technical evaluations, performance monitoring, and field tests.

The first phase of the program, PLiM (I), had concluded that it was technically and economically feasible to operate Kori Unit 1 continuously beyond the design life 30 years. Based on this result, the subsequent program, PLiM (II), had proceeded with focusing on evaluating in-detail integrity of major structures and components (SCs) aged, developing diagnosis techniques, and establishing aging management programs to ensure plant safety and reliability for the latter half of and beyond the design life.

To operate the plant beyond the design life Kori Unit 1 is to apply the aging management programs recommended from the technical assessments of the PLiM. No matter how well-planned PLiM programs are implemented for the continued operation of aged NPPs, both structural integrity and safety of the plant have to be enough to meet the level of safety and performance in cost-effective way.

Systematic periodic safety review (PSR) can be used to verify the plant operating safety incorporating with the PLiM program. Korean nuclear utility deals physical aging with the plant lifetime management and non-physical aging with the periodic safety review. PSR can supplement the PLiM program in verifying the plant safety. This paper introduces overall process of nuclear PLiM programs, its relationship with PSR in terms of purpose, scope, and depth, and summarizes the experiences and on-going status of Korean nuclear PLiM programs that is supposed as a good tool of continued operation.

1. INTRODUCTION

Seventeen nuclear power plants (NPPs) are currently operating, three units under construction, and four units scheduled to be built in Korea. There are four units of pressurized heavy water reactors (PHWRs) and others of pressurized water reactor (PWR). Kori Unit 1, the oldest PWR plant in Korea, has been operated to date for twenty-four years while

Wolsong Unit 1, the oldest PHWR plant, for eighteen years. Although the operating NPPs have recorded good performance with high capacity factors of each plant, introduction of deregulation and the subsequent competition in the electricity industry have raised two issues to the nuclear power. First, tougher competition is expected bituminous coal plants. Second, the siting and financing with of low interest rate for new NPPs have become quite difficult relatively to the past. Therefore, it is natural to make efforts to keep operating older NPPs beyond their design lives [1].

Several programs for the plant lifetime management have been performed to cope with the aging of operating nuclear power pants in Korea [2,3,4,5]. The first phase of Kori Unit 1 lifetime management program had shown that the continued operation beyond its design life was technically and economically feasible. The second phase had done with the result of detail life evaluation and provisions of aging management programs. Periodic safety review (PSR) program has been also adopted in Korea per recommendation of the International Atomic Energy Agency, which will be a very useful to assess and ensure the safety of operating plants and contribute to plant lifetime management (PLiM) [6].

Nuclear PLiM is all technical activities related to maintaining the safety and performance of operating plants up to the economically optimum life. Nuclear plant safety should not be affected and degraded by aging phenomena during the whole operating period. NPP aging can be taken account into two parts; the physical aging resulting in deterioration of physical and performance characteristics of systems, structures, and components (SSCs) and the non-physical aging resulting in obsolescence of the plant systems in comparison with current safety concepts, standards and technologies [7]. This paper introduces overall process of nuclear PLiM program rather than technical details, its relationship with PSR and summarizes the experiences and on-going status of Korean nuclear PLiM program that is supposed to be a good tool of continued operation.

2. NUCLEAR PLIM AND PSR

2.1. Safety

Plant safety is the most important factor in the operation and lifetime management of NPP. For safe operation and preventing plants from failures of their functions, utility implements comprehensive programs, such as preventive maintenance programs, refurbishment of degraded structures and components (SCs), and quality assurance programs of safety systems. In order to maintain the safety level acceptable, plant staffs carry out periodic inspections, in-service performance tests of operating components, and monthly safety diagnosis according to the technical specification of the plant and regulatory guidelines.

Nevertheless the plant safety can be affected by the status of system performance that could be dependent on the structural integrity and degradations of the SCs belong to the system. To solve this issue IAEA has recommended member states to implement PSR as a tool of ensuring a high level of safety throughout plant service life. Reviewing plant safety in every 10 years, PSR can deal with the cumulative effects of plant aging, modification, operating experience, and technology evolutions.

2.2. Lifetime management and PSR

A proper lifetime management program implemented in a plant can keep the plant safety in the initial good condition set at the beginning of operation. PLiM can control, within acceptable limits, aging degradations of the SSCs so that there remain adequate integrity, functions and safety margins in excess of their normal operating requirement. The result of PLiM program can be used as an input to "management of aging," one of the eleven safety factors of PSR. Other safety factors of PSR in addition to the management of aging are actual physical condition of NPP, safety analysis, equipment qualification, safety performance, use of experience from other plants and research findings, procedures, organization and administration, human factors, emergency planning, environmental impact [8].

By performing PSR adequately, utility can periodically confirm the plant safety through the whole service life of nuclear plants. It is possible to review comprehensively the current activities for the safety enhancement and to show public utility's efforts to upgrade the safety. Incorporating the results of PLiM program, PSR finally concludes the compliance of current safety level of the plant to the safety requirements of regulatory body. Current safety practices of the field can be improved, by repeating PSR every ten years, for reflecting the experiences and lessons learned from precedent failures, incidents, malfunctions, mitigation, and safety upgrades of other NPPs in the world.

2.3. Depth and scope

All of the plant SSCs, both the passive and the active, is dealt in PSR but PLiM basically focuses on the long lived passive components which are relatively hard to replace and refurbish during normal operation. Therefore it can be said that the scope of PSR is wider than that of PLiM. However, technical depth of PLiM is deeper than PSR because it includes engineering evaluations, such as quantitative time limited aging analysis (TLAA), residual life estimations, field tests and examinations, diagnosis and monitoring, and aging management programs (AMPs), etc. Figure 1 simultaneously tells the scope and depth of evaluating the SCs of PLiM from ones of PSR.

Because most of the SCs excluded from the PLiM are usually short-lived active components, they are scoped into the aging management of PSR, and the engineering level of life evaluation is not complicate and deep as much as that of PLiM. PSR reviews the current physical status and records of maintenance and inspection done to the components in the past. Comparing the review results with current safety standards and practical experiences on- and off-shore in terms of aging and maintenance, utility revises the technical procedures and plans how to improve the system safety and slow down the degradation of SCs for the next 10 years. So the depth of PSR engineering evaluation is shallower but the scope is wider than that of the PLiM.

3. NUCLEAR PLIM PROGRAMS

3.1. Strategy

PLiM program can be a powerful tool not only to solve the plant aging and maintenance obsolescence but also to provide the vision for continued operation beyond the plant design life. The primary goal of PLiM is to operate nuclear plants safely and economically during the design life of plants. When the primary goal is achieved, the operation of NPPs beyond the original design life will be carried out to operate plants up to their optimum life. In order to continue the operation of older NPPs, overall safety and integrity of the plant have to be justified technically and approved by a regulatory body.

Nuclear PLiM program of Korean utility usually consists of three phases as tabulated in Table 1. In the first phase, a feasibility of the continued operation is evaluated in terms of technical, economical and regulatory aspects to support top manager's decision making whether continuing to operate the plant. Once the policy is determined to operate the plant continuously on the basis of the feasibility study, the second phase program works out to evaluate detailed lives of SSCs and to establish aging management programs together with field walk downs, tests, diagnosis and aging inspections. If the license for the continued operation is endorsed by the regulatory body through PSR, the aging management programs are implemented to the field in the third phase by replacing aged components, installing new performance monitoring systems, and improving obsolescent systems in the following outages as scheduled.

3.2. Status

Kori Unit 1 and Wolsung Unit 1 are the leading plants of Korean PLiM program in developing technologies and field applications to the PWR and PHWR plants respectively. PLiM phase I of Kori Unit 1 to see the feasibility of continued operation had been done in 1996 and phase II program for detailed life evaluation and integrity assessment of SSCs to provide AMPs during remaining service life was done in 2001. While performing PLiM study of Kori Unit 1, pressurized thermal shock evaluation for the reactor pressure vessel (RPV) was completed in 1998 to verify the integrity of the aged RPV which could be the most life limiting component [9].

In 2000 KEPRI started to develop PLiM program of Wolsung Unit 1 (PHWR) to check the feasibility of continued operation. Some generic technologies used in PLiM of PWR can be used in PHWR but some technical methodologies should be developed to deal with the inherent degradation mechanisms unique to the PHWR SSCs. Because PLiM evaluation should address structural integrity and functional performance of the overall plant, all the SCs of PHWR systems should be technically evaluated plant-specifically using currently verified techniques.

Figure 2 depicts the generic process of nuclear PLiM technical evaluation that is being used in the Korean utility PLiM program. When the feasibility of the continued operation is verified based on the results of the phase I study, utility management makes phase II study be started in accordance with the recommendations from the previous one. If any technical issue from the recommendations is turned out to be so critical that impacts the whole PLiM and PSR program such as the PTS issue in Kori Unit 1 reactor pressure vessel, the issue shall be resolved by a separate program. Phase III includes the implementation of the AMPs developed in the phase II study to the plant with the schedule of PSR corrective actions. This general approach of nuclear PLiM program is applicable to both PWR and PHWR typed nuclear power plants.

3.3. Technical Approach

3.3.1. Screening of critical structures and components

Identification of critical components for aging evaluation is an important step of the PLiM phase I study, because it is necessary at the beginning of the PLiM program to concentrate the efforts and to properly allocate resources. The screening process to identify the critical components adapts safety-related criteria based upon the US license renewal rule (LR) and the maintenance rule (MR), described in 10CFR54 and 10CFR50.65 respectively.

Power production (PP) criteria showing the importance of a SC in power generation regarding plant availability and other safety requirements are also applied in the screening process. After screening critical SCs as shown in Figure 3, they are to be identified and prioritized to determine their relative importance in the PLiM. Critical components were prioritized utilizing ten attributes selected to assess the impact of component replacement or refurbishment on continued operation like replacement or refurbishment cost, impact on plant availability, radiation dose, etc.

Most of screened SCs would be long-lived passive SCs that are costly and technically difficult to resolve degradation because it is hard and expensive to replace or there are, sometimes, no precedent experiences. Other long-lived passive components discriminated from the PLiM and active ones of the plant that are relatively easy to replace or refurbish are to be assessed in the viewpoint of preventive maintenance and PSR. PLiM can be thought divided as an area of aging management for the long lived passive SCs and another area of maintenance for active ones.

Nuclear PLiM reviews the aging effects of long-lived passive components and establishes aging management program to sustain the integrity and safety of plant components throughout lifetime. Because the integrity of the active components can be maintained by casual or preventive maintenance, they are excluded from the scope of the aging management but have to be properly kept up good performance by optimum preventive maintenance programs. If an active component is considered important for power production and maintenance by field staffs, it could be included in the scope of PLiM program.

3.3.2. Engineering Evaluations

3.3.2.1. Sub-components

In order to ensure the feasibility of continued operation, it should be demonstrated that critical sub-components of a SC be maintained in a reliable service condition for the component through the whole service life. A detailed list of all the sub-components contributing to the intended functions of a SC is provided. Based on the risks and damage histories of the sub-components, critical sub-components affecting the integrity and functions of the SC are selected as items that need aging evaluation.

Screening sub-components is processed through several activities. The first step is to derive the list of all sub-components with the references of FSAR, the equipment specifications, and so on. The second is to characterize the sub-components considering function, geometry and design features. The third selects critical sub-components using screening criteria and groups them, if necessary, to make engineering evaluation convenient. Safety-related, long-lived passive, failure history, availability of repair or simplicity of replacement, and others could be regarded as points of engineering judgement in screening. By the result of taking these steps, the sub-components that need aging assessment and management are to be listed with technical rationales why they are screened as SCs important to the PLiM.

3.3.2.2. Aging mechanisms

After screening the sub-components, identification of age-related degradation mechanisms (ARDMs) should be performed. The first step is to determine ARDMs that might, generically, affect the integrity of the selected sub-components. Numerous industrial and regulatory technical documents, reports, and nuclear PLiM experiences are reviewed to provide a technical procedure and evaluation methodology in the process of Figure 3. Identified potential ARDMs with evaluation methods are compiled and written in a list. The list includes a discussion for various aging stressors that cause or exacerbate the ARDMs.

Then the next step is to identify those ARDMs that are applicable to the specific sub-components being evaluated. A matrix of ARDMs and sub-components is tabulated to assess aging effects of ARDM on the components and to find the relationships of them. At this point, material characteristics, operating environments, maintenance histories, etc. about specific sub-components are considered in tabulating the matrix. Potential ARDMs will be chosen with technical rationales and be used to manage the aging of each SCs.

3.3.2.3. Life evaluations

The main purpose of the life evaluation is to assess whether the integrity of sub-components are maintained, whether the aging phenomena occur or advance, and whether there is a sufficient margin for the integrity of component during operating period. The procedure of the life evaluation is divided into three steps; pre-evaluation, quantitative/qualitative evaluations in detail, and integrated life evaluation. The pre-evaluation usually done in PLiM phase I is summarized as followings; (1) review of design documents including design stress reports, design specifications, certificate of material test report (CMTR), geometric drawings, etc., (2) review of failure history and maintenance, and (3) determine whether the review results satisfy design acceptance criteria.

For the results of life evaluations are normally reviewed by the objective regulatory bodies to justify continued operation of a plant, aging mechanisms not applicable to a component should be justified why they are not addressed in the life assessment of it. The quantitative detailed evaluations are usually performed when the results of pre-evaluations do not meet the acceptance criteria of structural integrity. The qualitative evaluations which are most part of PLiM phase II life assessment are performed by investigating the following activities; review of operating condition and maintenance history, evaluation of plant specific aging effects, and determination of whether the result of aging evaluation satisfies operating acceptance criteria.

In the integrated life evaluation, residual lives of SCs are determined by reflecting the results of site walk down, examinations, monitoring or tests as well as detailed evaluation performed. These are integrated into the activities of establishing aging management programs (AMPs) of SCs to keep plant in the required level of safety and integrity up to the optimum life of plant. Figure 4 shows the process of detailed life evaluation in the PLiM program

3.3.2.4. Aging management programs

Aging management programs can provide an utility timely detection and mitigation of significant aging effects of plant to ensure their integrity and functional capability and to contribute to the safe and reliable plant continued operation. From the integrated life evaluation, long-lived passive components may need to have AMPs to sustain their intended functions against aging. The most suitable AMP to a component will be determined based on the result of sub-component aging evaluation and understanding current aging status of the component.

Existing AMPs usually well mitigate the aging effects occurring in operating plants. When, however, a new degradation mechanism is revealed or existing programs at present are not sufficient to control the aging mechanisms, additional AMPs would be implemented to the field as one of the PSR corrective action items. To develop proper AMPs, existing ones in the fields and other experiences of domestic and offshore NPPs should be reviewed. The literature survey includes in-service inspections and surveillance test experiences as well as operations, technical support, and external R&D programs.

Unless a proper AMP is searches for an aged SC, R&D plan would be required to plan to resolve, control, or monitor degradations caused by specific aging phenomena. While planning AMPs of a nuclear plant, it is important to consider an economic aspect of decisions on the type and implementing time of the aging management actions for the competitiveness of nuclear power operation.

3.3.3. Performance Monitoring and Field Tests

3.3.3.1. System performance monitoring

Even though the PLiM program assures the integrity and safety of SSCs, aging progress and current status of aged NPPs should be monitored so that any symptom of failure in system function be periodically checked up during normal operation to prevent the plants from being degraded. Field walk down, aging monitoring, system performance diagnosis and in-situ tests are necessary to verify that the plant is well maintained to the required level of safety. These activities support the technical life evaluation of the major SCs in understanding correct current physical aging, and also can be used as AMPs of continued operations.

Currently available monitoring systems for each degradation mechanism were identified by reviewing not only the monitoring systems of domestic NPPs but also additional new systems, such as radiation embrittlement monitoring of reactor pressure vessel material, fatigue and transient monitoring, and a cable condition monitoring system. Monitoring systems and tools developed for the aging management of Kori Unit 1 are listed in Table 2.

3.3.3.2. Field test and inspections

Field data earned by inspections, examinations, or tests are sometimes input to the PLiM technical evaluation. For example, it is well known that the ferrite content of material is a dominant factor in thermal embrittlement of reactor coolant system (RCS) piping made of cast austenitic stainless steel, that is, the higher ferrite content, the higher susceptibility of thermal embrittlement. As the ferrite content data shown in the CMTR of RCS piping of Kori Unit 1 was high enough to be susceptible to thermal embrittlement. Therefore the actual ferrite contents of the pipe obtained in field test during an outage of the plant s found to be low enough, Based on the field data it was concluded that Kori Unit 1 was not susceptible to thermal embrittlement.

Another typical field inspection in nuclear PLiM is an aging diagnosis of the buried commodities and structures, which seem to be low priority in normal plant operations and maintenance. To diagnose aging of the buried commodities and structures field walk down procedures was developed. Steel and concrete structures and buried commodities were visually inspected during the field walk down. PLiM usually recommends to perform field tests and aging management with others of cable aging diagnosis, field test of vibration and thinning integrity for main steam and feed water piping, thermal stratification test of pressurizer surge line, and leak detection of branch lines connected to RCS piping.

3.4. PLiM Database

In order to effectively carry out nuclear PLiM program it is very important to collect and keep the records of component design, manufacturing, operation, maintenance, replacement, and betterment histories. In case of old plants it is also additionally required to survey the status of back-fit implementation, experience of technical procedure changes, and safety upgrades to evaluate non-physical aging. As a prerequisite to the evaluation of the plant aging status, a voluminous amount of plant design and field data, accumulated since plant construction, should be surveyed and reviewed. This is an essential process of PLiM and performed on the plant-specific basis, which would require tremendous amount of resources for reproducing useful data from the raw.

Followings are examples of necessary data for PLiM technical evaluations. They could be general methodology and technical references, engineering procedures, operating transient history and experiences, component design specification and manufacturing data, maintenance practices and experiences, in-service inspection and surveillance data, results and procedures of life evaluation, aging management program and its implementation schedule, and the contents of related R&D's.

All the information and data produced by life assessment and economic evaluation in PLiM program have to be kept in PLiM database, and well organized so that parties related to aging management can share and use it to fairly verify plant integrity. The database should be updated periodically by a PLiM coordinator and be input with newly produced data throughout plant service lifetime. PLiMDB was developed so as to be operating on Web site for easy access of PLiM-related personnel or organizations and even public in the future.

4. CONCLUSIONS

Continued operation of a nuclear power plant beyond its design life with a proper lifetime management can be a effective way to resolve the increasing demand for elecricity avoiding huge investment for new constructions. The nuclear PLiM is at present one of the most important tasks in Korean nuclear programs as Kori Unit 1 becomes aged. PSR supplements the PLiM engineering evaluations of structural integrity and system functions with reviewing current status of operating NPPs in comparison with current safety standards. This paper discussed PLiM strategy, on-going status, and technical approach, and the relationship between PLiM and PSR in terms of purpose, scope, and depth.

The technical approach dealt with overall process and items to be considered in nuclear PLiM based on lessons learned and experiences of Kori Unit 1 and Wolsung Unit 1 PLiM programs. The general sequence of technical evaluation in nuclear PLiM was in turn explained. That is to screen SSCs important to PLiM, to perform engineering evaluations, and to do system performance monitoring and field tests. The engineering evaluation consists of screening sub-components, identifying aging mechanisms, life evaluations, and establishing aging management programs for aged SCs. All the data collected and results from PLiM program has to be stored and kept in systematic database so that parties related to aging management can share and use it to fairly verify plant integrity.

Korean utility is ready to apply PSR to operating domestic NPPs because nuclear PLiM program has been carried out last decade. It is believed that well-harmonized PLiM program with implementing PSR is so useful for utility to diagnosis system performance and integrity of SCs and to keep plant safety at the level of regulatory requirements. Another contribution of PLiM to nuclear utility is expected to reduce the financial burden of new plant construction and to increase the competitiveness of nuclear power production. Furthermore, continued operation of NPPs by PLiM evaluation shall be one of countermeasures to mitigate greenhouse effect of carbon dioxide.

REFERENCES

[1] Chung, K. N., June 2000. Reorganization of KEPCO and Lifetime Management Status in Korea, Int. Workshop on Nuclear Power Plant Life Management in a Changing World, OECD/NEA, Washington DC, USA

[2] Hong, S. Y., Jeong, I. S. and Jin, T. E., 1997. Recent Strategy and Progress in Korean Nuclear PLIM Program, Nuclear Engineering and Design 174, 25-32

[3] Jeong, I. S., May 1999. Lifetime Management Study of Kori Unit 1, IAEA IWG-LMNPP TCM, Vienna, Austria

[4] Jeong, I. S., Kim, J. S. and Hong, S. Y., October 2000. Nuclear Plant Lifetime Management and Periodic Safety Review, 12^{th} Pacific Basin Nuclear Conference, Seoul, Korea

[5] Jeong, I. S., Jang, C. and Hong, S. Y., 2001. Lessons Learned from the Plant-specific Pressurized Thermal Shock Integrity Analysis on an Embrittled Reactor Pressure Vessel, Int. J. of Pressure Vessels and Piping(2001) to be published

[6] Pachner, J., August 1999, Systematic Aging Management Process: A Key Element for Long Term Safety, Reliability and Economy of Nuclear Power Plants, Tran. of the 15th Int. Conf. on SMiRT, Seoul, Korea

[7] Park, Y. O., Lee, J. I. and Eun, Y. S., April 1999. Regulation Technology for Nuclear Plant Lifetime Management in Korea, The 14^{th} KAIF/KNS Annual Meeting, Seoul, Korea

[8] IAEA, November 1994. PSR of Operating Nuclear Power Plants, Safety Series No. 50-SG-012, IAEA, Vienna, Austria

[9] KAIST, July 1999. Lifetime Management and Performance Enhancement Technologies for NPPS, Industry-School Co-operation Seminar, KAIST, Taejon, Korea

Table 1 Phases of nuclear PLiM program

Phases	Period	Contents
Phase I	2-3 years	- Feasibility study of the continued operation - Evaluation the feasibility in terms of technology, economy and license
Phase II	3-4 years	- Detail life evaluation and provision of aging management program - Comprehensive safety evaluation and license acquirement
Phase III	7-8 years	- Implementation of the aging management program and components replacements for the continued operation

Table 2 Monitoring Systems developed in the PLiM of Kori Unit 1.

Monitoring Systems	Aging Mechanisms	Components or Systems
Cable Aging Tester (CAT)	Jacket Hardening	Cables
Environment Temperature Monitor (TEM)	Environment Temperature	
Transient Auto Counting System (TACS)	Fatigue	Pressure Boundary Components
Monitoring Inherent System Performance (MISP)	Intended Function	General Systems
Component Performance Analysis for IST (CoPAIST),	Operating Function	IST related Components
Ex-vessel Neutron Dosimeter (END)	Neutron Embrittlement	Reactor Pressure Vessel
Detection for Isolation Valves Internal Leak (DIVIL)	Thermal Stratification by Internal Leak	Pressure Boundary Valves

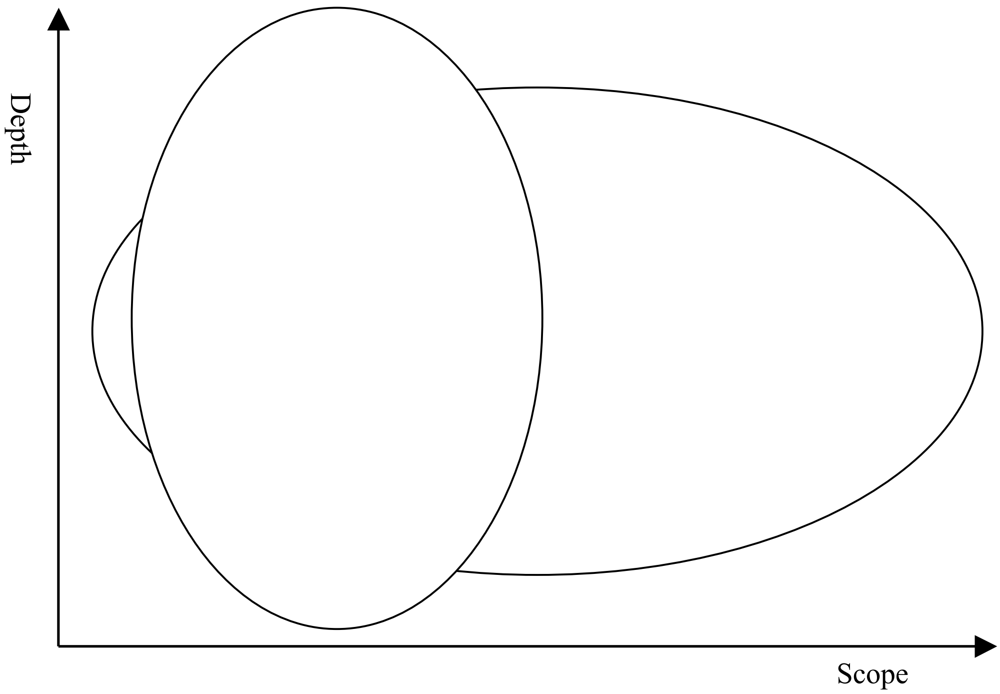


FIG. 1. The Scope and Depth of PLiM and PSR in Korea

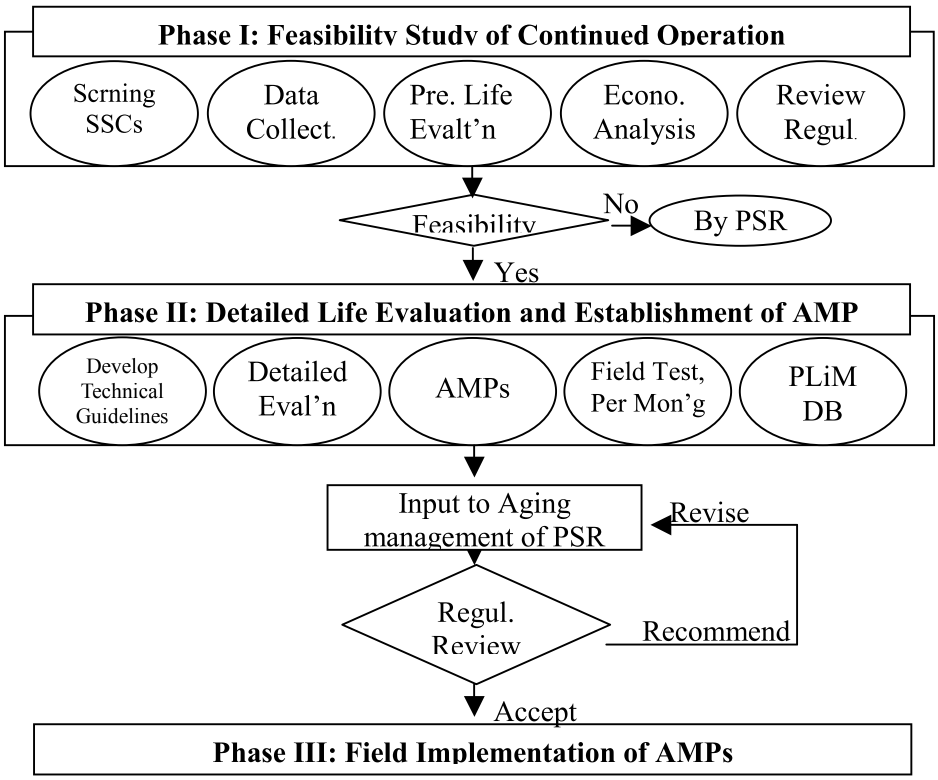
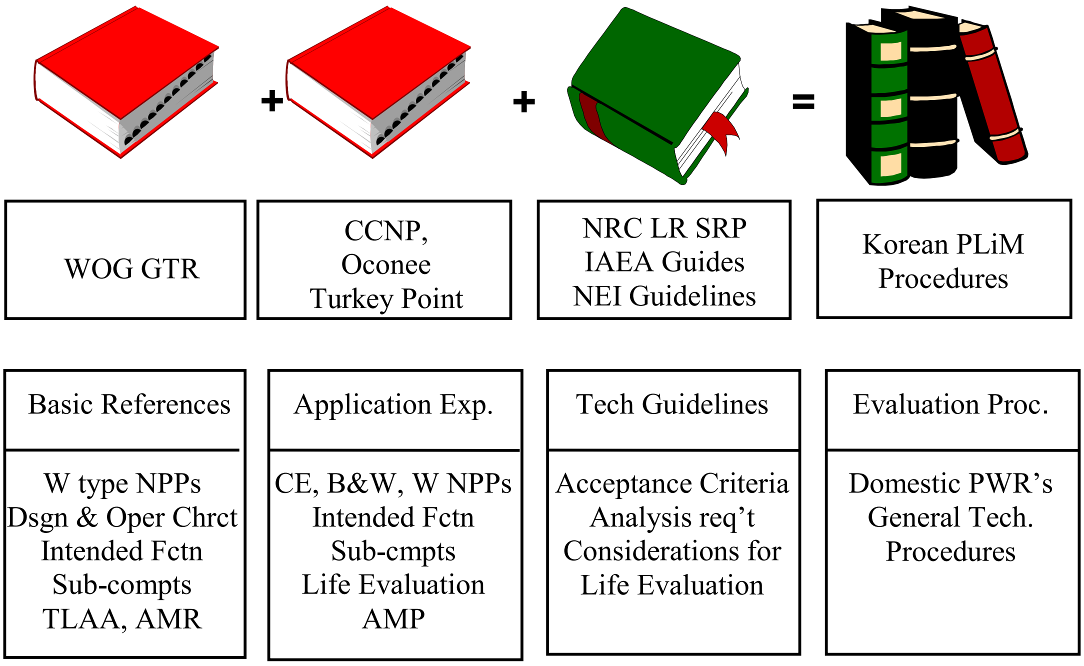
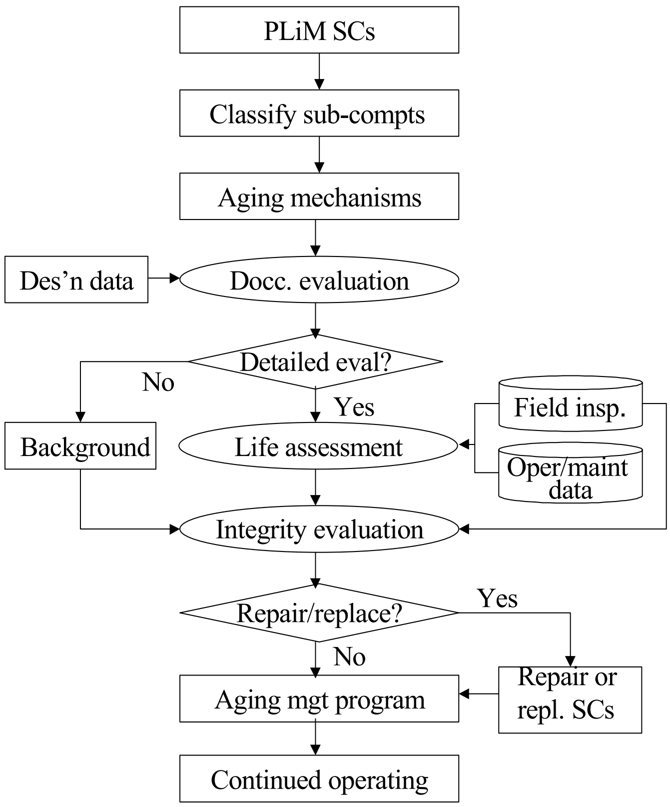


FIG. 2. Generic Process of Nuclear PLiM Program



FG.3. Developing Technical Procedure and Evaluation Methodology



FG.4. Detailed Life Evaluation of SCs

Session 5

CURRENT NATIONAL APPROACHES TO PLIM

Sub-Session 5.3

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MANAGING AGEING OF KARACHI NUCLEAR POWER PLANT

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Abstract

Karachi Nuclear power Plant (KANUPP), a 137 MWe CANDU plant, was built on a turn-key basis by the Canadian General Electric (CGE) in the late sixties. The plant with a design life of thirty years went into commercial operation in October, 1972. After nearly three decades of operation, KANUPP, like all other plants, has faced the problem of equipment ageing and obsolescence. KANUPP has been striving hard to combat these problems with the assistance from IAEA, COG and WANO. During early eighties IAEA expert missions were called at KANUPP on various safety issues and, on their recommendations, various projects such as Safe Operation of KANUPP, Technical Upgradation Project, Balancing, Modernization and Rehabilitation Project, were initiated to combat equipment ageing and obsolescence. KANUPP has made substantial progress in the implementation of the tasks under these projects and the operating life of the plant is expected to be extended by 15 years. Three IAEA expert missions were invited to KANUPP during 1999-2000 to carry out assessment of the ongoing activities related to plant ageing management. Based on their recommendations an Ageing Management Programme has been formally established at KANUPP to manage age-related degradation of plant systems, structures and components important to safety and to ensure that the required safety functions are available for the extended life of the plant. KANUPP has applied to Pakistan Nuclear Regulatory Authority for renewal of operating license. This paper briefly describes the activities related to ageing management of KANUPP.

1. INTRODUCTION

Karachi Nuclear Power Plant (KANUPP) is a 137 MWe CANDU Pressurized Heavy Water Reactor (PHWR). It is situated on the Arabian Sea coast, 20 km from the centre of Metropolitan City Karachi. Built in the late sixties, it is among the first few CANDU nuclear generating stations and is certainly the pioneer Canadian off-shore plant. The plant commissioned in 1972 has completed its nominal design life of 30 years and has generated more than 10 billion units of electricity so far.

Ageing is a natural phenomenon and ignoring its effects is likely to result in degraded plant performance. It results from such diverse phenomena as wear and tear, crack development, fatigue, increase in vibration, corrosion, erosion, deposition, oxidation, changes in material properties, etc. In addition to routine preventive maintenance, a number of projects for safety improvements, surveillance and technological upgradation are in progress for combating the effects of equipment ageing and obsolescence. Recently a proactive type ageing management programme utilizing a systematic ageing management process has been started to provide an umbrella to all the relevant operations, maintenance and engineering programmes.

2. NEED FOR CONTINUED OPERATION OF KANUPP

Karachi Nuclear Power Plant during its operation so far, has been operating mostly below 100 MWe, with the 30 years operation being equivalent to nearly only 10 Effective Full Power Years (EFPYs). The lifetime availability factor of the plant has been around 56%.

KANUPP has so far operated safely with no incident that would cause any concern to the safety of the people of the city of Karachi or to the people living in the areas closer to the plant. Further, there have been no abnormal release of radioactivity during the normal plant operation and KANUPP has at all times met the radiation release limits while still working on the principle of ALARA.

3. MEASURES TO COMBAT AGEING

Based on the performance of the plant and the operating experience from other CANDU plants the following projects were initiated in late eighties and early nineties to combat ageing/obsolescence and for safety upgrades of the plant:

3.1. Safe Operation of KANUPP (SOK) Project.

KANUPP is an old plant, which was designed to safety standards of early sixties. Like all other old plants, it has also encountered usual age-related problems, equipment obsolescence and safety issues accruing from earlier safety standards, which differ substantially from the current standards. This is primarily due to the fact that the safety has continuously been evolved as a result of research, technological advances and operating experience. It is not true that older plants are unsafe. The older plants can maintain and demonstrate acceptable level of safety because of conservative design, in-service inspections, ageing management and safety backfits.

KANUPP has initiated a systematic programme for determining the adequacy of major systems particularly those that provide the basic safety of the plant. IAEA Operational Safety Review Team (OSART) Mission was invited first in 1985 and then in 1989 to review the plant's operational safety practices and give recommendations to improve its performance. An Assessment of Safety Significant Events Team (ASSET) Mission from the IAEA was also invited in 1989 to conduct an in-depth analysis of reactor components specially sagging of one of the fuel channels. In November, 1990 IAEA/CANDU Owners Group (COG) expert team also reviewed the safety features of KANUPP against modern CANDU safety standards and practices.

KANUPP has greatly benefited from the IAEA missions and based on the recommendations of OSARTS (85, 89), ASSET (1989), IAEA/ COG expert team and the work done at KANUPP, an Integrated Safety Review Master Plan (ISARMAP) was prepared for the plant for overall improvement in its performance and safety..

Based on ISARMAP, a project "Safe Operation of KANUPP (SOK)" was prepared, and got approved by IAEA. This project has now been renamed as "Improve Safety Features (ISF)" of KANUPP. IAEA has set up a Steering Committee consisting of IAEA experts from Canada and other CANDU operating countries to guide, prioritize, adjust and approve the implementation plan of the project. Significant progress has been achieved in the implementation of the various tasks of this project, and this has made great contribution towards improving safe operation of KANUPP. Some of the major activities of SOK/ISF project have been covered under the following sections.

a) Project Management: To arrange and coordinate the resources required, and prioritize/schedule the execution of the various types of activities to optimize the safety return at all times.
b) Ageing: To assess and rehabilitate safety-significant equipment affected by ageing.
c) Obsolescence: To replace and upgrade the safety-significant equipment affected by obsolescence.
d) Operational Safety: To improve and modernize the operational safety practices and procedures.
e) Design Safety Improvements: To implement design improvements in the safety systems based on an updated safety analysis, while adopting the feasible obvious improvements towards modern design practices.

3.2. Technological Upgradation Project.

This large project has been undertaken to combat obsolescence in C&I Equipment / Systems. Some of the major jobs already completed are listed below:

- Up-gradation of T/G Machine Monitoring System
- Plant Communication System Improvements
- Replacement of Plant Radiation Monitoring Equipment
- Power for Karachi Nuclear Power Complex by the Extension of KANUPP 132 KV Switchyard

The backfitting of Plant Computers, Control & Instrumentation is in hand. Engineering of CC&I backfitting has already been completed and its installation is in progress.

- Under this project following replacements are being carried out:
- Measurement Loops (Pressure, Temperature, Flow, level) 76
- Closed Control Loops (Pressure, Temperature, Flow, Level) 41
- Safety Loops relating to Reactor Protection & Engineered Safeguards 68
- Digital Computers performing the functions of Reactor Regulation, 02
- Alarms Annunciation, Logging, Special Computations.

The project is expected to be completed by the mid 2003.

3.3. Balancing Modernization & Rehabilitation Program

For ageing management of KANUPP's conventional island, a comprehensive project namely "Balancing, Modernization and Rehabilitation (BMR)" is under way. Some of the major jobs being undertaken under this project are as under:

Plant Chillers	One Chiller replaced, another will be replaced soon.
Service Air Compressors	One compressor replaced.
Boiler Feed Water Motors	One motor replaced.
Rotating Seal Faces for PH-Pumps	Fabrication Completed, Replacement when required.
Condenser Tubes Replacement	To be undertaken in early 2003.
Desalination Plant	Installed, operational.

Oil Circuit Breakers	One main output breaker replaced with SF6 type circuit breaker.
Digital Radio Communication link	Replaced.
Fire Alarm System	Installed, operational.
Generator Automatic Voltage regulator	Planned

4. IAEA ASSISTANCE FOR AGEING MANAGEMENT

Various IAEA missions were invited to KANUPP to carryout assessment of the ongoing ageing management activities as given below:

a) IAEA Ageing Management Team (AMAT Mission) - November 1999
   - To assist in establishing a systematic Ageing Management Programme (AMP).

b) IAEA expert mission for preview of Ageing Management Programme for I&C Cables - November 2000.
   - To provide guidance for establishing and maintaining AMP for cables.

c) IAEA expert mission on Ageing Management Programme for MOVs - November 2000.
   - Recommended overhauling and refurbishment of safety related MOVs and actuators.

d) National Workshop on Easy Fixes Program for Seismic Upgrading of NPP – November 2000.
   - To guide for implementation of seismic fixes, complete geophysics study of off-shore area.

Based on the recommendations given by the IAEA expert missions, during their visit to KANUPP, a proactive type ageing management programme has been started to effectively manage the plant ageing. The programme can provide the timely detection and mitigation of ageing degradation in order to ensure that the required safety margins (i.e. the integrity and functional capability) of the SSCs are maintained. Such programme include maintenance, in-service inspection and surveillance, as well as operations, technical support and external programmes such as R&D. Plant operations, inspection and maintenance supported by trending and assessment of condition and functional indicators are the primary means of managing ageing in nuclear power plants.

5. AGEING MANAGEMENT PROGRAMME (AMP)

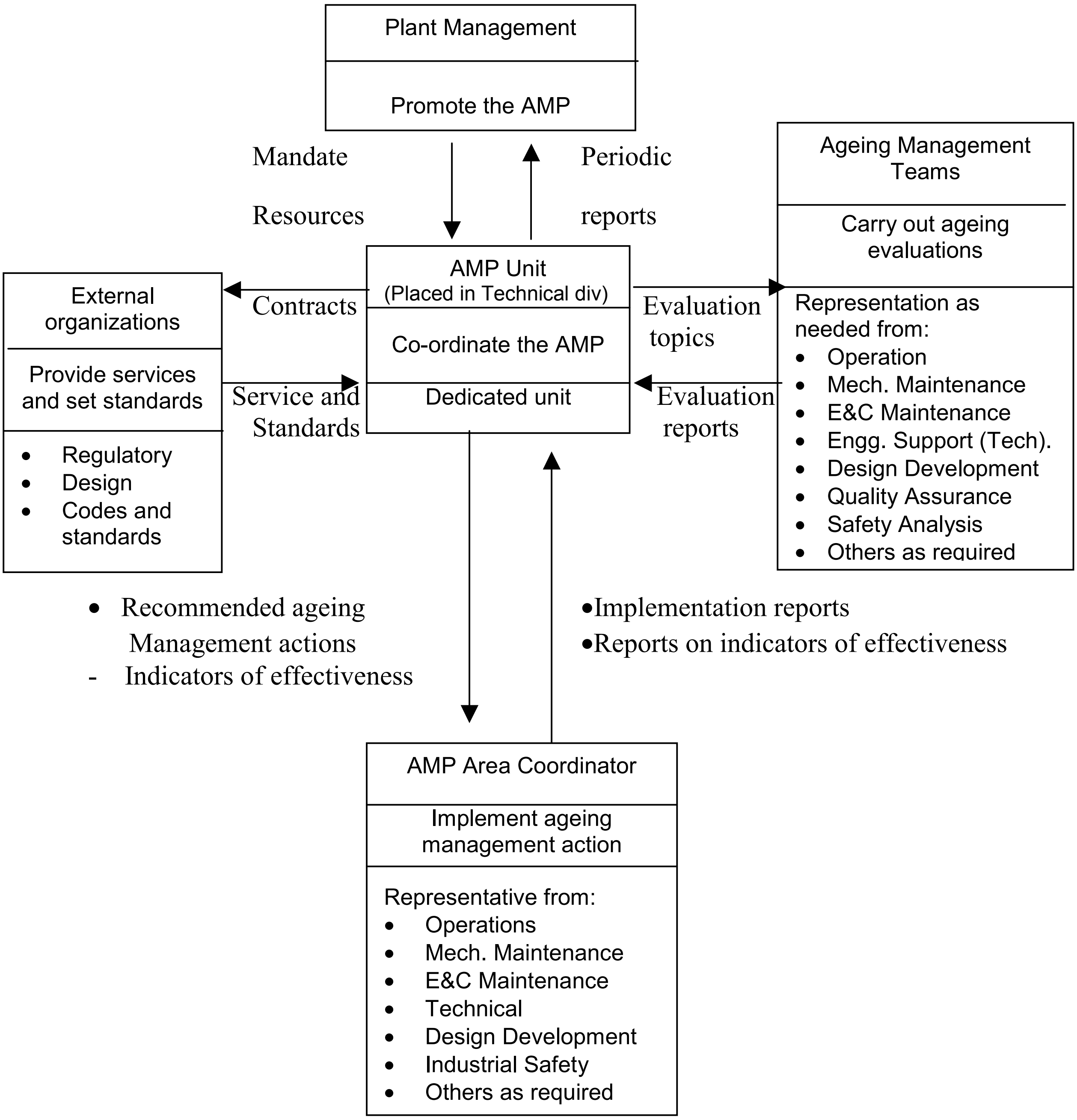
A work plan of Ageing Management Programme has been prepared keeping in view the recommendations of IAEA expert missions. The salient features of AMP are as under:

5.1. Station Policy.

Prior to establishing an Ageing Management Programme it is essential to define a clear cut station policy. Keeping this in mind a policy has been established on AMP defining the scope and goal of AMP, AMP organization setup, and role and responsibilities of AMP unit.

5.2. Ageing Management Organization Set-up

The implementation of a systematic ageing management process requires an organization which systematically co-ordinates all relevant existing programmes and activities. The organizational set-up for AMP at KANUPP is illustrated as below:



5.3. Management of Material Ageing

Managing material ageing involves screening SSCs, performing ageing evaluations for the selected SSCs, and implementing appropriate ageing management actions.

5.3.1. Screening Components

Many of the components that would be identified as important are relatively short lived or are consumables (e.g. gaskets, lubricants and some I&C equipment) and are already covered by existing replacement or preventive maintenance programmes. These components do not require further ageing evaluations. The AMP will cover the long-lived passive components, which are not routinely inspected or maintained (e.g. inaccessible pipes, structures, vessels, valve bodies or cables). A manual on “Identification of SSCs” describing the methodology for screening of SSCs, based primarily upon identification of the active and passive functions, has been prepared.

5.3.2. Performing Ageing Evaluations

Evaluations of the selected SSCs will be carried out to systematically assess age related degradation mechanisms and their effects on their ability to perform the design functions. Effective ageing management actions will be developed to detect and mitigate the ageing effects before the integrity or functional capability of the SSCs is compromised. A technical manual has been prepared to establish the instructions and requirements for performing ageing management review. The review process evaluates ageing, identifies ageing management practices and performs an assessment of the component’s condition.

5.3.3. Ageing Management Condition Assessment

The ageing management condition assessment is performed in two parts: a system ageing management review and periodic condition assessment. The first part compiles and evaluates ageing management practices from a system performance perspective to determine the need for additional ageing management or performance monitoring. The second part periodically evaluates the system condition and the effectiveness of ageing management practices and implementation plan.

5.3.4. Implementing Ageing Management Actions

Implementation of the recommendation, evolved as a result of ageing management evaluations, will be carried out by the division/ units responsible for inspection, maintenance, and operations. These divisions/ units however, will first review the recommendations and decide on their cost effectiveness and long-term safety and reliability benefits.

5.3.5. In-Service Inspection as an Ageing management tool

In-Service Inspection (ISI) as the name implies is inspection of System, Structure or a Component during its useful life. An effective ISI program thereby contributes to the management of ageing of NPPs ensuring its safe operation. In an Ageing Management Process inspection and surveillance play a vital role in managing ageing. Inspection of any kind helps in assessing the condition of that part while being exposed to degradational parameters like temperature, pressure, radiation and of-course the material ageing under the influence of these parameters. Some sort of inspection had been going-on at KANUPP since its inception during the maintenance works until a regular ISI facility with the assistance of IAEA, was established in late 1980s. KANUPP has benefited from the numerous IAEA based expert missions that visited KANUPP from 1989 onwards in establishing a self-sustained ISI facility. KANUPP ISI programs are in conformance with the ASME and CSA inspection requirements. These ISI programs are referred to as In-service Inspection Program (ISIP) and

Periodic Inspection Program (PIP) respectively. An inspection manual is in place for all ISI activities including Reactor Components, Main steam Condenser, Turbine-Generator, Balance of Plant piping and are mostly based on vendor recommendations and the 'Good Practices' credited in various NPPs in the world.

KANUPP had been regularly performing inspection of its PHT (Piping and components), Steam generator (vessel and tubes), Main steam condenser (tubes and structure), Balance of Plant (BOP) piping and supports, and several other components and piping. Consequently a watch on their degradation is thus maintained for operational and ageing aspects. So far the degradation assessment has been successfully achieved on SGs, Condenser and BOP areas. This can be inferred from the following facts:

- The condenser tubes that had been causing concerns of tube leakages were plugged on prediction basis and the plant has since then operated smoothly.
- The only SG tube leak that occurred in 1990 is a single event after which a comprehensive inspection was done and some tubes are being pro-actively plugged.
- Most of the PHT and BOP mechanical structure seems to be in good condition to operate the plant for another 10 to 15 years.

5.4. AMP Pilot Projects.

Implementation of an AMP is started with a few pilot projects dealing with a limited number of representative SSCs in order to establish, define and adapt the AMP processes and interfaces at the plant. Pilot SSCs are chosen for the purpose of demonstrating AMP processes and assessing the availability of data rather than on the basis of their importance for plant safety or reliability. Once the processes have been demonstrated, implementing procedures written and a working relationship established, the AMP will be extended to cover the remaining important SSCs.

Following four SSCs have been selected for AMP at KANUPP as pilot projects:

i) Fuel Channel Integrity Assessment (FCIA)

- Assessment was carried out in 1993
- One channel was removed and 7 others were inspected
- Condition of the channels was found quite good.
- Next assessment will be carried out by early 2003.

ii) Sludge Removal of Steam Generator by Water Lancing

- Soft sludge above tube sheet removed.
- Inspection of SG tubes has been carried out.
- Constriction mapping, plugging of the constricted tubes and strengthening by stabilizer bars.
- Removal of hard sludge and SGs chemical cleaning have been planned.

iii)AMP for Electrical Cables

- Recording of environmental condition data and listing of safety related cables is in progress.
- Radiation monitors installed at various locations of the cables in the reactor building about an year ago for dose measurement are being sent to the IAEA experts for assessment of cables service lifetime.

- Plan for replacement of the cables will be made after getting report from the IAEA experts.

iv) AMP for MOV / SOV

- One actuator of the MOV has been replaced with a new one.
- The old actuator is being refurbished.
- The refurbished actuator will replace another old MOV actuator and so on. until all the actuators of the MOVs are refurbished.

5.5. Backfitting of Electrical Equipment

i) Design Review of Electrical Distribution System

An engineering study of KANUPP distribution system and coordination of protective relays has been carried out to get the documents, related to system design and technical specification, updated in accordance with the current standards.

ii) Oil Circuit Breakers

In 1998 main generator output circuit breaker (bulk oil type) was destroyed after a fire incident that took place in its control panel due to ageing of its components. The breaker has been replaced with a new SF6 type circuit breaker. It has been planned to replace the remaining two oil circuit breakers of the transmission lines by the end of 2003.

iii)Protective Relays of Transmission Lines

Problem of spurious operations of ground directional overcurrent protective relays of the transmission lines at KANUPP end is being faced frequently. The reason is increase in the fault level of the Karachi Electric Supply grid resulting in invalidity of the relay settings. These relays cannot be adjusted according to the new fault level due to their limitation. Moreover, there is no technical support from the original vendor. Hence, it has been planned to replace the transmission line protective relays by the end of 2003.

iv)Low Voltage Switchgear

Low voltage circuit breakers supplied by the Canadian General Electric were originally equipped with electromechanical type direct-acting overcurrent protective devices (known as EC unit). With the passage of time these EC units have become problematic and often cause spurious tripping of breakers and need replacement. About 30% EC units have been replaced with microprocessor based over-current trip devices known as Micro Versa Trip (MVT). The rest of the units are planned to be replaced soon.

v) Boiler Feed Water (BFW) Pump Motor

The current shorting bars (made of the aluminum alloy) of the rotor of Boiler Feed Water Pump motor have been found broken at their welded joints due to ageing. According to an expert from the original vendor General Electric, repairs of such type of damage have not been found successful. The motor has been replaced with a new one.

5.6. Inspection of Civil Supports and Structures

Physical inspection of the plant building have been carried out in consultation with Civil Engineering experts. To assess the life of the reactor building core sampling is planned to be carried out during long shutdown. Tendon inspection of reactor building is also planned to be carried out in collaboration with the experts.

6. CONCLUSION

The various safety reviews and investigations carried out over the years by IAEA and WANO has indicated that KANUPP has maintained a good safety record and nothing is wrong with its critical components such as reactor, its auxiliaries and turbine-generator, etc.

Existing programmes such as preventive maintenance, in-service inspection and surveillance contribute to combat the ageing of plant systems, structures and components. Different plant groups belonging to Operation, Maintenance, Technical and Development divisions are carrying out these programmes. All such programmes have now been focussed under the Ageing Management Programme.

KANUPP has amply demonstrated that it is possible to continue its operation safely for the next 10-15 years utilizing the indigenous support. However, the sustained excellence could be better achieved with the support and input from the vendor and international assistance from IAEA, WANO and COG.

REFERENCES

[1] International Atomic Energy Agency, Implementation and Review of Nuclear Power Plant Ageing Management Programme, Safety Report Series No. 15, Vienna (1999).

[2] International Atomic Energy Agency, Methodology for the Management of Ageing of Nuclear Power Components Important to Safety, Technical Report Series No. 338, Vienna (1992).

[3] International Atomic Energy Agency, Assessment and Management of Ageing of major Nuclear Power Plant Components Important to Safety IAEA. TECDOC – 1197, Vienna (2001).

[4] Siddiqui, Z. H., Mian, A. G., Upgrading the Safety to Extend the Life of KANUPP, WANO-TC/PAEC Inter-regional Seminar, 14-18 Nov, 1995, Islamabad, Pakistan.

[5] Hoda, M. Q., Tariq, M., Safety Culture at KANUPP – Good Practices and Self Assessment.

[6] Rahman, J., Alam, M., Maintenance Experience at Karachi Nuclear Power Plant – Problems and Remedial Actions (presentation, Korea, 1996).

CONSIDERATIONS RELATED TO PLANT LIFE MANAGEMENT FOR CERNAVODA – 1

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Abstract

Cernavoda-1 NPP, the first CANDU in Europe, is one of the original five CANDU 6 plants and the first CANDU 6 producing over 700 MWe. While the first series of CANDU 6 plants (which entered service in the early 1980's) have now reached the middle portion of their 30 years design life, the Cernavoda-1 was put into service on 2 December 1996. However, the Plant Life Management (PLiM) Program for Cernavoda-1 should be an increasingly important program to Utility ("CNE – Prod") in order to protect the investment and the continued success of plant operation. The goal of the paper is to document and prepare a preliminary study on the concept of PLiM Program applicable to Cernavoda-1 NPP.

1. INTRODUCTION

The Plant Life Management Program, known as ***PLiM Program*** is concerned with the analysis of technical limits of the safe operation - from the point of view of nuclear safety - in NPP units, aiming at attaining the planned 30 years life duration and its extension to 40 or even 50 years of safe and economical operation. For the CANDU-type the so-called PLiM and PLEX (Plant Life Extension) Programs are just applied [1 – 11]. These are applied research programs that approach with priority the current practices for assessing the capability of safe operation within the limits of nuclear safety (***fitness-for-service assessment***). These programs also approach inspection, monitoring and prevention of degrading due to the ageing of critical systems, structures and components (CSSC). The systems, structures and components (SSC) that influence decisively the Nuclear Power Plant (NPP) reliability are considered as critical. Also, for the accident conditions, the SSC, which have a major influence to the system availability / operability, are considered as critical.

All SSCs are subjected to degradation and ageing effects, with reactor operation and passage of time. These effects must be recognized at the design stage to provide sufficient margins and must be effectively managed during operation to ensure that the required safety margins are maintained throughout the plant life. This field is becoming more widely recognized as the world inventory of nuclear power plants ads operating experience and as the average age of the nuclear fleet increases. For many nuclear plant operators, plant design life is no longer a limiting factor. The plant economic viability can be extended beyond the original design life provided that the degradation and ageing effects are understood and provisions have been made in the plant design and maintenance programs for inspection and refurbishment. Plant life extension beyond the original design life is becoming an attractive economic consideration in the nuclear industry.

Life management requires an in depth understanding of the behaviours of materials in an operating environment over extended periods of time. Ageing and degradation during the plant-operating phase is dependent on system conditions, maintenance and operating environment. Through proper monitoring and mitigation programs, ageing of major plant components are managed to ensure that the design life is achieved or exceeded, [12].

As each nuclear plant is somewhat different in its components and systems, materials composition, procurement, construction, and operational history, directed research and development programs into materials behaviour, monitoring techniques, and methods to mitigate ageing are required to support the life management program. Over the past three years, ICN (Institute for Nuclear Research – Romania) has been working with AECL – Canada on R&D Programs to support a comprehensive and integrated Cernavoda-1 Plant Life Management (PLiM) program that will see the Cernavoda-1 NPP successfully and reliably through to design life and beyond.

2. CERNAVODA-1 THE 5th ORIGINAL CANDU 6

The first commercial CANDU units have already reached the planned 30 years life-duration, as is the case with Units 1,2 and 3 at Pickering A, while the four CANDU 6 plants, considered by B.A. Shalaby in [10] and [9] as original for the CANDU 6 project (i.e. Point Lepreau, Gentilly-2, Wolsong-1 and Embalse) operate for 19 and 18 years, respectively. To-date there is 8 CANDU 6 units in operation and 3 under construction. If in Canada [13] and Korea [14] programs regarding planned life management are already implemented, how urgent could the problem be in Romania [15], where Cernavoda-1 Unit (the 5th in the original CANDU 6 series) was commissioned on December 2nd 1996 (the first criticality being attained on April 16, 1996), although the works on the site began in 1980 [16].

While from the standpoint of technology and construction (design documentation, procurement and construction) the Cernavoda-1 is almost identical with Wolsong-1, from the standpoint of commissioning and operation is almost identical with Point Lepreau. From the point of view of safety objectives (including the licensing conditions) it is identical with the first four CANDU 6 NPPs. This is why we consider that the original CANDU 6 series encompasses 5 (five) nuclear plants: Point Lepreau, Gentilly-2, Wolsong-1, Embalse and Cernavoda-1. The Table 1 displays the evolution of the CANDU 6 reactors.

If we also add the 16 years completion period (the largest until now for a CANDU 6 unit) there are enough arguments in favour of our opinion, that the Plant Life Management program applied to Cernavoda-1 is not only actual, but also important to protect the investment and to successfully carry on the plant operation. As regards the performances in operation we can mention the following:

- On July 1st 1997 the Cernavoda-1 operation was put under the direct management of Romanian operators, authorized by CNCAN. The first year of operation ended with a capacity factor of 87.3%.
- On June 30, 1999 the Cernavoda-1 came 10th in the world (among nuclear reactors of more than 150 MWe power) with a capacity factor of 99.0%[1].
- In the year 2000 Cernavoda-1 produced 5,053,335 MWh and delivered into the National Power System 4,950,334 MWh, which is about 10% of the total output of energy in Romania. The operation performances in 2000 were outstanding: Cernavoda-1 recorded the best gross capacity factor since commissioning, i.e. 88.3% thus classifying among the first CANDU-type units in the world [18].

[1] "Nuclear Engineering International – Load factors to end of June 1999"

Table 1. Evolution of CANDU 6 Reactors

CANDU 6 Reactors	Unit	Location	Gross Output	Project Cod	In-Service Date	Age (Years)
Original Generation	Point Lepreau	Canada	680 MWe	87	Feb.01, 1983	19
	Wolsong-1	Korea	679 MWe	59	Apr.22, 1983	19
	Gentilly-2	Canada	675 MWe		Oct.01, 1983	19
	Embalse	Argentina	648 MWe	18	Jan. 20, 1984	18
	Cernavoda-1	**Romania**	**706 MWe**	**79**	**Dec.02, 1996**	**6**
Current Generation	*Cernavoda-2*	*Romania*	*706 MWe*	*82* [17]	*2005*	-
	Wolsong-2	Korea	715 MWe	86	July 01, 1997	5
	Wolsong-3	Korea	715 MWe	86	July 01, 1998	4
	Wolsong-4	Korea	715 MWe	86	Oct. 01, 1999	3
Advanced Generation	*Qinshan-1*	*China*	*728 MWe*	*98*	*2003*	-
	Qinshan-2	*China*	*728 MWe*	*98*	*2003*	-

On April 30, 2002 the Cernavoda-1 NPP performances for the entire period of operation were:

- Total generated power (gross): 29,965,948 MWh
- Total power to grid (net): 27,634,387 MWh
- Gross capacity factor since in service: 86.96%

A first and important step in applying the Program of "Plant Life Management" in Cernavoda-1 was taken in 1999 by the Inaugural Inspection of pressure tubes (Baseline Fuel Channel Inspection) [19]. On this occasion 14 pressure tubes selected according to well-defined criteria have been inspected, out of the 380 pressure tubes of the core [20]. During the inspection the CAN/CSA N285.4-94 standard was applied ("Periodic Inspection of CANDU NPP Components", published in December 1994 and in force at the inspection date).

In defining and applying a PLiM Program at Cernavoda-1 we start from the premise that Cernavoda-1 NPP, the first CANDU-type nuclear plant in Europe, is one of the five original CANDU 6 (original generation) and the first one of this type in the world that turns out more than 700 MWe.

3. CERNAVODA-1 PLIM PROGRAM APPROACH

The main objectives of the PLiM program applicable to Cernavoda-1 are [10], [8]:

a) To maintain the long-term reliability and safety of the Cernavoda-1 during the design life (life assurance).
b) To maintain the long-term availability and capacity factors of the plant with controlled and reasonable generating costs during the nominal design life of (life assurance).
c) To "avoid surprises" through identification of potential ageing issues, ahead of its occurrence and provide means for monitoring and mitigation to ensure reliable component performance.
d) To preserve the option of extending the life of Cernavoda-1 with good safety and availability at reasonable cost, beyond the nominal design of 30 years, up to 50 years (life extension).

The strategy adopted in preparing the concept of a PLiM program applicable to Cernavoda-1 involves the following steps:

a) Identify critical components.
b) Undertake Ageing assessment studies of such critical components.
c) Implement Life Management Programs aiming to maximise component life, ensures good performances and monitor plant conditions.
d) Plan, scope and implement required programs attain the original design life.
e) Prepare economic case studies for rehabilitation and life extension (PLEX programs).
f) Implement rehabilitation and operate beyond the nominal design life.

Following the above strategy, the multiphase approach of a PLiM Program applicable to Cernavoda-1 includes three phases in its structure:

- **Phase 1**: Detailed studies for the identification of critical SSCs, evaluation of ageing and definition of credible mechanisms of critical components degrading;
- **Phase 2**: Definition, planning and implementation of detailed programs of ageing management (AMP), [21 – 25], in view of attaining the planned life duration;
- **Phase 3**: Upgrading, replacement and maintenance of critical components in order to ensure extension of the plant life duration.

The key elements of the PLiM program are detailed in Fig. 1 while the targets of each phase are summarized in Table 2.

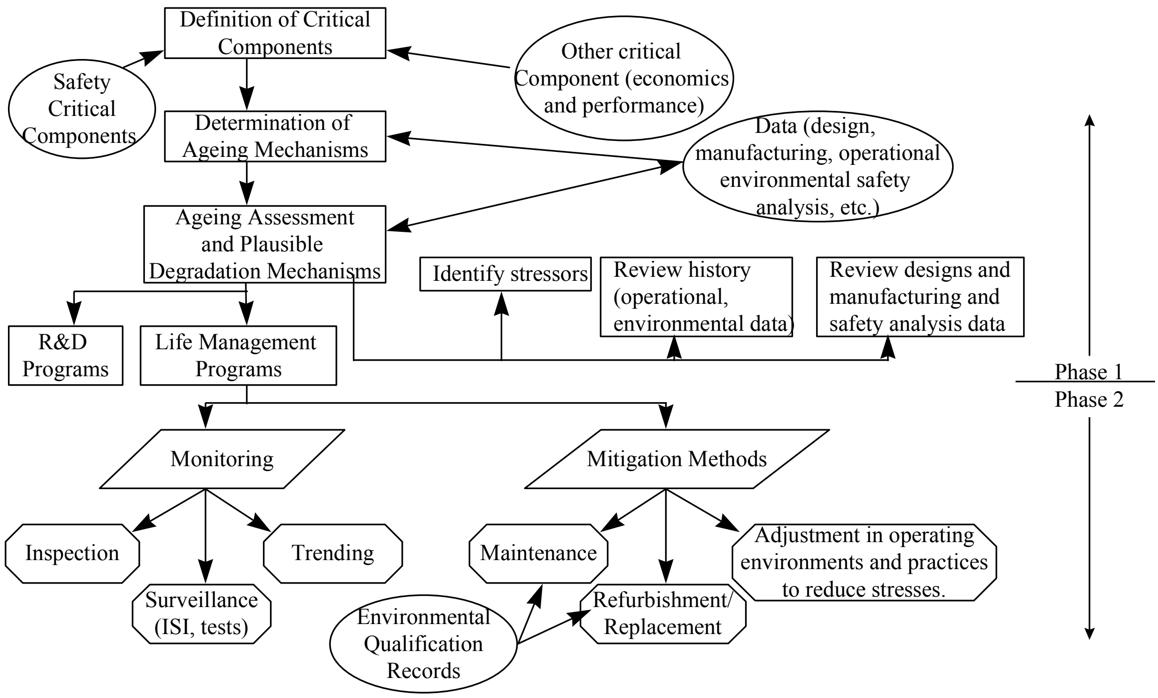

FIG. 1 Elements of CANDU 6/Cernavoda-1 Plant Life Management

Table 2. The CANDU 6/Cernavoda-1 PLiM Multiphase Approach

Phases	Scope
<u>*Phase 1*</u> PLiM Assessment and Recommendation Program	• Identification of major CSSCs • Ageing assessment studies & R&D of critical components • Systems maintenance optimization studies • Technology Watch planning • Advanced technology development
<u>*Phase 2*</u> PLiM Attainment Implementation Program	• CANDU 6 plant specific detailed inspection and residual life assessment of key components • Implementation plant monitoring and surveillance life management programs • Enhancement of plant inspection and maintenance • Technology Watch implementation
<u>*Phase 3*</u> PLiM Extension Program	• Replacement component strategies and planning • Assessment of regulatory and safety related design changes for life extension • Rehabilitation / Replacement programs for components identified in CSSC studies or from inspection in plant life attainment program

The PLiM Program suggested to be applied at Cernavoda-1 NPP would prevent premature ageing of the plant, ensuring reliable operation, with the observation of nuclear safety requirements for the entire designed lifetime duration and will allow even its extension to more than 50 years. The proposed PLiM Program considers nuclear safety, operation performances and economical requirements.

4. R&D SUPPORT TO PLIM PROGRAM

The multiphase program, proposed to be applied at Cernavoda-1 NPP, is supported both by the experience of CANDU 6 owners and by the results of research conducted within ICN. Thus, the first step of *Phase 1* has been covered, referring to the studies on the assessment of CSSCs operation, encompassing the methodology related to the definition of critical SSCs [15]. The works have been performed between 2000 – 2002, within the ICN R&D Program on ***"Process Systems and Equipment"*** [26], [27]. This programme deals with the increase of performances of NPP systems and components; their upgrading based on the evaluation of their operation behaviour. Another objective of this programme is the evaluation of "ageing" of some critical components in the structure of NPP systems (i.e. heat exchangers, valves).

In order to attain all the objectives of *Phase 1*, ICN has been initiated other four R&D programs for the evaluation of the capability to carry on safe operation within the limits of nuclear safety ("*fitness-for-service-assessment*") of the key critical components in the Cernavoda-1 plant, such as:

- ***"Fuel Channel"***
- ***"Steam Generator"***
- ***"Chemistry, Chemical Control"***
- ***"Instrumentation and Control"***

The ICN R&D programs in support of the first phase of the PLiM Program are focused on:

- Understanding operating environment and degradation mechanisms, and developing models.
- Developing and applying inspection and monitoring technology.
- Applying models to field data to predict component behaviour and recommend maintenance and management activities, and/or develop and qualify improved components or systems.

Until now considerable progress has been made in the understanding degradation mechanisms and developing Ageing Management Programs for the major critical components, i.e. pressure tubes, reactor assemblies, steam generators and feeder assemblies, [28].

Phase 2 of the Cernavoda-1 PLiM program started with the Inaugural Inspection of the pressure tubes, is concerned with the maintenance and inspection requirements for the beginning, mid-life period and the last years of operation. At Cernavoda-1 a special program for technological surveillance has been initiated [29]. The objectives are focused on the early identification of potential ageing phenomena and of defect modes that could affect the plant performances, together with the inspection and maintenance required monitoring and evaluating the ageing effects. In parallel, the ICN initiated the R&D program: **"*Analysis of NPP Operation Events, Ageing, Environment Qualification and Increase of NPP Lifetime*"**. This program includes the acquiring of international experience related to the categories of events occurring in NPP operation and to the actions taken to eliminate their causes, accomplishment and implementation at Cernavoda-1 the PLiM Program.

5. TIME SCHEDULING OF THE CERNAVODA-1 PLIM PROGRAM

For a better timing of the Cernavoda-1 PLiM program there were considered both the framing of Cernavoda-1 in series zero of the CANDU 6 type NPPs and the main instances of its completion and operation. The main milestones in the history of the Cernavoda-1 completion are:

- 1982: The containment concrete was poured (reactor building base slag).
- 1989: Mounting of the 380 pressure tubes.
- December 1989: Romanian revolution; The Cernavoda-1 is 45% complete.
- 1995 May-June: The fuel loading of the Cernavoda-1.
- 1996, April 16: The first criticality of the Unit 1 reactor.
- 1996, July 11: The first synchronization to the grid of the Cernavoda-1 NPP.
- 1996, December 2: The Cernavoda-1 is declared in commercial operation.

The following moments related to operation have also been considered:

- 1998, May-June: The first inspection of turbo-generator [30];
- 1999, November: Completion of the first inspection of the pressure tubes [19].

Besides, it is estimated that the Utility – "CNE-Prod" will benefit from the ICN research and development works and from the COG/IAEA experience in the application of PLiM program. Another prediction is that a period of 15 years, from the plant commissioning until the replacement of the first pressure tubes, is an optimistic one for Cernavoda-1 in comparison to the situation of pressure tubes at Wolsong-1 [31].

Another worth-mentioning issue in the case of Cernavoda-1, whose completion took more than 16 years, is the moral wear (*obsolescence*) of components/equipment. This aspect implies three levels of evaluation [32]:

- An item or system is suspected of becoming *obsolete* at the time it cannot longer be procured by normal means.
- Obsolescence begins when a manufacturer removes it from his marketed product range.
- Absolute obsolescence occurs when a supplier withdraws all forms of support or service from an item.

As a confirmation of these considerations, Fig. 2 suggests a time scheduling of the three phases of the PLiM program applicable to Cernavoda-1 NPP.

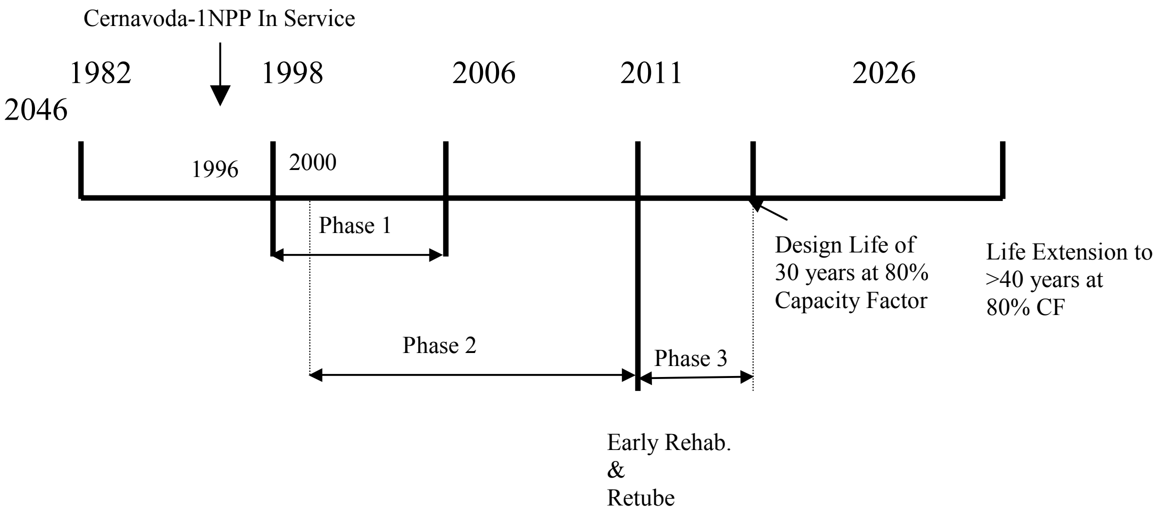

FIG. 2 Time scheduling of the PLiM Program applicable to Cernavoda – 1 NPP.

6. CONCLUSIONS

A comprehensive and integrated Plant Life Management (PLiM) program applicable to Cernavoda-1 has been proposed. A time scheduling of the multiphase Cernavoda-1 PLiM program has been suggested. This program should be helped the Utility–“CNE-Prod” to achieve goals for safe, economic and reliable life attainment and to preserve the option for extended operation.

The ICN/Utility PLiM program interaction should be provided the utility with in-depth assessment and promising life prognosis for the key critical components, structures and systems in the plant. The proposed ICN/utility interaction also demonstrates the synergy of the ICN R&D programs with the PLiM Program applicable to Cernavoda-1.

REFERENCES

[1] J.H. Nickerson, et al, "*An Overview of Plant Life Management and Life Extension Plan for CANDU™ 6*", 16th International Conference on Structural Mechanics in Reactor Technology, Washington DC, U.S.A. (2001).

[2] J.H. Nickerson, et al "*Plant Life Management Considerations Related to Life Extension of CANDU™"*, 2001 KAIF Conference, (2001).

[3] J.H. Nickerson, et al. "*CANDU™ Plant Performance Improvements and Life Management*", 2000 KAIF Conference, Seoul, Korea, (2000).

[4] J.H. Nickerson, et al "*Reliability Enhancement and Life Extension Plan for CANDU™"*, 2000 KAIF Conference, Seoul, Korea, (2000).

[5] J.H. Nickerson, et al "*Designing for Longer Life and Plant Life Attainment Programs"*, 1999 KAIF/KNS Conference, Seoul, Korea, (1999).

[6] J.R. Hopkins, et al "*CANDU Plant Life Management Programs*", 1999 SIEN Conference, Bucharest, Romania, (1999).

[7] R.L. Tapping, R.A. Holt, *"R&D in Support of CANDU Plant Life Management"*, 1999 SIEN Conference, Bucharest, Romania, (1999).

[8] P. Charlebois, et al "*CANDU Plant Life Management – An Integrated Approach*", KAIF/IAEA Conference, Vienna, Austria, (1998).

[9] P. Charlebois, B.A. Shalaby, "*CANDU Plant Life Management – Safeguarding the Investment*", 1998 PNBC Conference, Banff, Canada, (1998).

[10] B.A. Shalaby, "*Key Programs on Lifetime Management of CANDU NPP",* IAEA PLiM – PLEX 97 Conference, Prague, Czech Republic, (1997).

[11] B.A. Shalaby, et al *"Current Activities in Support of CANDU Plant Life Management: An Industry Perspective",* IAEA – doc.

[12] J. Pachner, "*IAEA Guidance on Managing Safety Aspects of NPP Ageing",* IAEA Workshop "Ageing Management for Cernavoda NPP", ROM/4/027 9001 01, Vienna, Austria, (2001).

[13] R.F. Dam, et al "*The CANDU Experience with Reliability Centred Maintenance and Plant Life Management",* IAEA – Specialists Meeting on Optimization of NPP Maintenance Programmes, Charlotte, NC, U.S.A., (2001).

[14] Ill-Seok Jeong, Sung-Yull Hong, *"Nuclear Plant Lifetime Management Program of Korean Utility",* 16th International Conference on Structural Mechanics in Reactor Technology, Washington DC, U.S.A., (2001).

[15] G. Florescu, M. Cojan, "*Maintenance Optimization Methods for NPP Critical Systems, Structures and Components",* IAEA – Specialists Meeting on Optimization of NPP Maintenance Programs, Charlotte, NC, U.S.A., (2001).

[16] M. Cojan, et al, IRNE Report – Pitesti, Romania, (1979).

[17] G. Lipunov, A.F. Parsons "*Identification and Review of the Reference Safety Objectives*" NNC Task 2-Report, Appendix G: "List of Changes to be Applied at Cernavoda Unit 2", Phare Project "Cernavoda 2 – Nuclear Safety Expert Project", NNC Ltd. UK, Ref: C6214/0002/TR/001, (2001).

[18] I. Rotaru, "*Cernavoda NPP – Performance and Development Activities* ", paper presented at ICN Conference "Nuclear Power – Present and Future", Pitesti, Romania, (2001).

[19] *** "*New Fuel Channel Inspection System at Cernavoda Unit 1"*, NEXUS – AECL News, Volume 2, Issue 4, (1999).

[20] M. Soare, "*Baseline Inspection of the Fuel Channels at Cernavoda NPP Unit 1*", Technical Specification, 79-03641-TS-001, Rev.0, Cernavoda, Romania (1998).

[21] IAEA, "*Assessment and Management of Ageing of Major NPP Components Important to Safety: Steam Generators"*, IAEA – TECDOC – 981, Vienna, (1997).

[22] IAEA, "*Assessment and Management of Ageing of Major NPP Components Important to Safety: Concrete Containment Buildings"*, IAEA – TECDOC – 1025, Vienna, (1998).
[23] IAEA, "*Assessment and Management of Ageing of Major NPP Components Important to Safety: CANDU Pressure Tubes"*, IAEA – TECDOC – 1037, Vienna, (1998).
[24] IAEA, "*Assessment and Management of Ageing of Major NPP Components Important to Safety: In-containment instrumentation and control cables"*, IAEA – TECDOC – 1188, Vienna, (2000).
[25] IAEA, "*Assessment and Management of Ageing of Major NPP Components Important to Safety: CANDU reactor assemblies"*, IAEA – TECDOC – 1197, Vienna, (2001).
[26] M. Cojan, Information Report *"2000 R&D Program: Process Systems & Equipment",* ICN – RI 5962, Pitesti, Romania, (2001).
[27] M. Cojan, Information Report *"2001 R&D Program: Process Systems & Equipment",* ICN – RI 6228, Pitesti, Romania, (2002).
[28] M. Cojan, et al, "Understanding ageing mechanisms of critical components", ICN – RI 6323, Pitesti, Romania, (2002).
[29] T. L. Corlan, "*Optimized Integrated Maintenance Program at Cernavoda NPP",* IAEA – Specialists Meeting on Optimization of NPP Maintenance Programs, Charlotte, NC, U.S.A., (2001).
[30] *** *"Cernavoda NPP – Unit 1"*, COG Report "CANDU Station Performance Newsletter", (1998).
[31] Y.W. Park, S.S. Kang, B.S. Han, *"Structural integrity assessment of pressure tubes for Wolsong Unit 1 based on operational experiences",* Nuclear Engineering and Design, 212, Nos.1 – 3 (2002) 41 – 48.
[32] Ki-Sig Kang, "*Management of Ageing of I&C Equipment in NPP",* IAEA Workshop "Ageing Management for Cernavoda NPP", ROM/4/027 9001 01, Vienna, Austria, (2001).

PLANT LIFE MANAGEMENT IN THE SLOVAK REPUBLIC

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Abstract

At present, there are under operation two units of WWER 440/230 and four units of WWER 440/213 reactor type in Slovakia. The construction of two more units in Mochovce NPP has been interrupted and nowadays they are under conservation regime. Whereas, it was decided by the Government to decommision two older units in Bohunice NPP earlier at two years against the the original design lifetime, there is the licensee aim to extend the operation of WWER 440/213 type units beyond the design lifetime. This paper presents information about current Slovakian legal basis regulating the ageing management and plant life management. The Safety Guideline on Nuclear Power Plants Ageing Management issued by the Nuclear Regulatory Authority supplements generally binding legal regulations with methodology for implementation and utilization of Ageing Management Programme.

1. INTRODUCTION

The Nuclear Regulatory Authority of the Slovak Republic (NRA SR) was established on January1, 1993, after splitting of the former Czechoslovak Republic. All legal documents valid in Czechoslovakia were valid also in the Slovak Republic. Nevertheless the process for of the development of the new legal basis documents started.

In 1998, new “Atomic Act” has been issued, which replaced former “federal” atomic act. On the basis of this act new regulations have been developed in order to prescribe in more detail the atomic act provisions.

The third level of Nuclear Regulatory Authority documentation represents safety guidelines, which discuss in details the provisions of the atomic act and corresponding regulations.

There are two groups of safety guidelines. The first group represents those that are obligatory to licensees and which are based on the international treaties on peaceful use of nuclear energy. The second group is aimed to elaborate in more detail provisions of generally biding legal regulations.

In the frame of the nuclear power plant life management, the Nuclear Regulatory Authority of the Slovak Republic has elaborated and issued the Safety Guideline on Nuclear Power Plant Ageing Management.

The current situation in Nuclear Power Plant Ageing Management. (AMP) and Nuclear Power Plant Life Management (PLIM) is presented in this paper.

The aim of this paper is to describe the current situation in following areas:

- legal basis for AMP and PLIM

- description of NRA SR Safety Guideline on AMP
- description of current situation in NPPs
- conclusions and lessons learned

Legal Basis for AMP and PLIM.

The atomic act regulates the peaceful use of nuclear energy.

One of the areas regulated by the atomic act is “an extension of nuclear power installation lifetime”. In a general meaning, the act regulates the life extension of any nuclear facility – including nuclear power plants.

In the paragraph 16 of the atomic act, there are declared three basic statements.

1.) Regulatory authority may extend the validity of the operational licence based on and actual condition (state) of the nuclear installation and also on the supplementary safety documentation. The extension of operational licence validity, according to the atomic Act, may be done against the licensee’s application.

2.) The supplementary safety documentation supplements the safety documentation submitted to regulatory authority prior commissioning and operation licence application. Supplementary safety documentation is a required enclosure to the application for the nuclear installation lifetime extension.

3.) The third paragraph of the Article16 of the Atomic Act says that the contents and the scope of supplementary safety documentation will be elaborated by the regulation issued by the regulatory authority.

In 2002, the Nuclear Regulatory Authority of the Slovak Republic developed and issued the Regulation No.318/2002 Coll. on Safety Documentation of Nuclear Installation, which covers all phases of nuclear power plant lifetime, including the construction, commissioning, operation, lifetime extension and the decommissioning.

As far as concerns the nuclear power plant life extension, the Article 27 of this regulation defines the scope of supplementary safety documentation to be submitted to the regulatory authority within the application for the nuclear installation lifetime extension.

The main components of this documentation is:

- an overall evaluation of condition of equipment, especially the physical state, the evaluation of the safety criteria fulfilment, trending and prognosis
- evaluation of operational phase, namely, usage factor calculations, limiting number of operational regimes, violation of technical specifications, failure and accident events evaluation
- evaluation of ageing management programme as one of the precondition enabling the lifetime extension.
- modification of the operational procedures (both emergency and for normal operation as well) which are necessary to be modified for the extended period of operation
- design modifications required for the extended operation beyond design lifetime. These may cover both, the obsolesce of SSCs and also requirements of the up-dated safety criteria

- safety assessment of proposed modifications – it is required to elaborate and to submit to the regulatory authority the appendix to the operational safety analysis report reflecting the adopted and the planned safety modifications.

Nuclear Regulatory Authority has developed and issued a safety guideline entitled as BNS II.1.9.2/2001 on "Ageing Management of NPPs".

The purpose and the main aims of these guidelines are:

1) To provide the operating and technical support organisations with the methodology for elaboration, implementation and utilisation of ageing management programme of the nuclear power plants.
2) To elaborate and develop in more detail the provisions of generally binding legal regulations – i.e. the Article 16 of the act No. 130/1998 Coll. and the Article 27 of the Regulation No 318/2002 Coll.

The main features of this guideline are as follows:

- The guidelines are fully based on IAEA documents, e.g. IAEA TECDOC, IAEA Guidelines, etc.
- The guideline is not obligatory but the Nuclear Regulatory Authority of the Slovak Republic in specific cases may require the operator to follow these guidelines and to act in accordance with them.
- The requirements for the ageing management programme according to this guideline are understood as minimal ones and the operator can extend them in an appropriate manner.
- Despite the fact that the title of the guidelines concerns the nuclear power plants, the guideline can be use correspondingly for all types of nuclear installations.
- The safety guideline is preliminary valid for two years. After this period, based on the comments and recommendations of the und users the guideline will be revised.

The guideline in detail defines basic areas of the concern as follows:

1) Field of application

a) defines goals of the guide, i.e. to ensure the safety and reliable long term operation of an NPP through:
   - understanding of degradation mechanisms
   - monitoring and trending of degradation
   - minimisation of expected degradation
   - implementation of appropriate corrective measures
   - evaluation of AMP effectiveness

b) Interface with other programmes. In the nuclear power plant, there are in force the programmes, dealing with maintenance, in-service inspection, in-service testing, research and development programmes and others. The AMP should have a feedback from these programmes and the clearly defined interface.

c) Main attributes and the scope of AMP. The guideline clearly defines specific areas to be included into AMP and defines required scope of associated documents.

d) Operation, maintenance and in-service inspection feedback. The ageing management programme shall have a clearly defined procedure for the utilisation of operational, maintenance, ISI and IST experience feedback.

2) Criteria for the SSCs selection

The criteria for the selection of SSCs shall be defined as the first step in the ageing management programme. The criteria shall consider following factors:

- Importance to nuclear safety – the safety classification shall be done according to internationally recognised criteria.
- Possibility of the replacement of component (components considered to be unreplaceable, replaceable components, bud the cost is too high, and components easily replaceable).
- Expected failure modes (e.g. violation of integrity, failure to start/to stop, etc.):
- Dose rate of personnel – radiation situation in connection with the operation, monitoring, maintenance, ISI, etc.
- Operational experience feedback (operating regimes, low cycle fatigue, chemistry...).
- Availability of monitoring techniques – implementation and utilisation of condition monitoring systems.

3) Organization of ageing management programme.

It is required that the control and the co-ordination of all activities within the ageing management programme shall be made by, for this purpose assigned, organisational unit within the licensee organisation.

This unit should be composed of an appropriate staff, to be able to cover all technical issues of the AMP.

The interface with other organisational units within the licensee organisational structure shall be clearly defined.

One of the duties of this organisational unit should be focussed on the co-ordination of the research and development activities in the field of ageing management.

Very important portion of activities of AMP organisational unit is the regularly assessment of the effectiveness of the AMP utilisation. For this purpose the criteria, which are to be used for the „effectiveness measurement“, should be established in order the corrective measures and the recommendations could be easily identified.

4) Database requirements

Generally, two groups of databases can be identified:

- general data described individual SSCs (or family of SSCs) in a general manner, as e.g. : drawings, main (design) dimensions, material characteristics (standard requirements), etc
- specific data of SSCs characterising each individual SSCs in term of deviation from the design and standard requirements. Such data e.g. are: dimensional deviations, real material testing data, description of flaws identified during manufacturing of SSCs, etc.

Ageing management database is a living document, which has to be kept in up-to-date state. Therefore, it is necessary to manage an effective interface with other plant databases.

5) Requirements on documentation

The safety guideline lists documents, which compose mandatory components of the ageing management programme.

There is a strong requirement on the documentation, which shall comply with the nuclear power plant quality assurance systems.

The emphasis should be given to the good administration and record keeping practice.

6) Assessment of AMP implementation

The implemented ageing management programme to be effective shall be assessed in an appropriate periodicity. It is recommended to evaluate the utilisation of AMP in one and four year intervals (e.g. in-service inspection periodicity is 4 years for WWER NPPs) and within the periodic safety review, which is done in ten-year intervals. The guidelines require the results of the AMP assessment to be submitted to the regulatory authority for information.

7) Responsibilities

The guideline defines the responsibilities and duties of all parties involved in the ageing management programme, i.e.:

- operational organisations
- technical support organization
- regulatory authority

Current situation in Slovakia

At present, in the Slovak Republic, there are 6 NPP units in operation, 2 units are under construction, though the construction has been frozen.

The age profile of Slovakian NPPs is following:

plant/type	unit	commissioned	remark
Bohunice V-1/WWER 440/230	1	1978	EOL 2006
	2	1980	EOL 2008
Bohunice V-2/WWER 440/213	3	1984	PLEX
	4	1985	PLEX
Mochovce/ WWER 440/213	1	1998	PLEX
	2	1999	PLEX
construction frozen	3		
construction frozen	4		

Comments to the ageing profile table:

Bohunice V-1 NPP

During 1991-1993 the first stage of the safety upgrading took place at the Bohunice V-1 NPP - so called "small reconstruction". During this period very important safety upgrading and mitigative actions have been done. The most important actions were:

- annealing of the core belt weld of the reactor pressure vessel of both units
- sampling of the RPVs material to precise a chemical composition of the base and weld metal of RPVs
- seismic upgrading
- fire protection upgrading, etc.

In the second phase, the gradual reconstruction has been implemented in the period 1994-2000. The major activities were:

- ECCS improvement
- sprinkler system improvement
- I & C system refurbishment
- service water system reconstruction, etc.

It can be stated that the safety upgrading measures identified in the IAEA - TECDOC – 640 "Ranking of Safety Issues for WWER-440 Model 230 Nuclear Power Plants" have been fully implemented. The result of both stages of reconstruction is the increasing of DBA, originally LOCA of ID 32 mm to new one of LOCA with DEGB of ID 200 mm and at the best estimate condition even with DEGB of ID 500 mm.

End of life of these two units is now determined by the decision of the Slovak government to 2006 and 2008, respectively. The design lifetime is shortened by 2 years.

Bohunice V-2 NPP

At present the modernisation and the safety-upgrading programme is under way. This programme is scheduled up to 2008 and all issues determined in she IAEA-EBP-WWER-03 publication on “Safety Issues and Their Ranking for WWER-440 Model 213 Nuclear Power Plant” will be implemented (lot of them were already implemented within the construction and the refuelling outages).

Within this modernisation programme, one issue is devoted to the plant life optimisation. This issue will be implemented as the research and development project. The main aim this project is to utilize of an integrated plant life management programme and the plant lifetime management.

Mochovce NPP

Two units (unit 1 and unit 3) are in operation, and unit 3 and 4 are under construction. The construction has been frozen. At present, the unit 3 and 4 are in the conservation regime. For the conservation and the preservation of equipment the conservation programme has been developed and approved by the regulatory authority. The conservation is preliminary scheduled for five years.

The ageing management programme is implemented in each nuclear power plant and covers following issues:

- RPVs embrittlement (surveillance programmes)
- fatigue damage evaluation (including monitoring)
- corrosion programme of primary circuit components materials
- corrosion-erosion programme for secondary circuit
- confinement (tightness, cladding degradation)
- cables

It is expected that ageing management programmes will be revised in the light of the results of research and development programme, raised for Bohunice V-2 nuclear power plant.

2. CONCLUSIONS

1) Implementation of ageing management programme is precondition for NPPs life extension.

2) Existing generally binding legal regulations assume a potential plant life extension.

3) Safety guideline on “Nuclear Power Plant ageing Management” provides methodology for implementation and utilization of AMP in the Slovak Republic.

REFERENCES

[1] AMAT guidelines, IAEA Service Series No. 4, IAEA, Vienna, 1999.
[2] PDRP-3 Regulation of the life cycle of nuclear installations, IAEA Vienna, 1999.
[3] Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Steam Generators, TECDOC 981, IAEA, Vienna, 1997.
[4] Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: PWR Pressure Vessel, TECDOC 1120, IAEA Vienna, 1999.
[5] Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Concrete Containment Building, TECDOC, 1025, IAEA Vienna, 1998.
[6] Assessment and Management of Ageing of Major Nuclear power Plants Components Important to Safety: PWR Vessel Internals, TECDOC 1199, IAEA Vienna, 1999.
[7] Glossary of Nuclear Power Plant Ageing, OECD-NEA.

NUCLEAR POWER PLANT LIFETIME MANAGEMENT STRATEGIES - THE SPANISH APPROACH

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Abstract

Spanish nuclear power plant (NPP) owners are increasingly concerned with optimising Plant Life Management. As a result of this interest, they are setting up Lifetime Management Programmes with the strategic objective to operate the NPPs beyond the 40 years design life (long term operation). In addition, a close link exists between life management work and improved maintenance practices concerning safety, strategic and reliability aspects of NPPs. As result of these important considerations, utilities will increase and globalise programmes for monitoring and mitigation of ageing. Spanish utilities have carried out, under UNESA coordination, a common methodology for NPP lifetime management that is developed through a specific Life Management Programme for each plant. The methodology consists in the following main steps: definition of priorities and scope (SCC selection); identification and analysis of ageing mechanisms; evaluation of current maintenance practices; and, review of operation and maintenance programmes. Also described in the paper is the current Spanish regulatory framework related to the NPP life management. With no defined time limit for the NPP operation, the plant operating license is based on periodical renewals associated with Periodic Safety Reviews (PSR) done in general every ten years. Actually, the Spanish Nuclear Regulatory Authority is working in determining specific aspects of the licensing process to be required for the NPP long term operation.

1. INTRODUCTION

During the last years, Spanish utilities that own nuclear power plants (NPP) are increasingly concerned with optimising plant life management. As a result of this interest, Life Management Programmes have been set up with the strategic objective to operate the NPPs as long as they are considered safe and reliable. In parallel with this initiative, both utilities and the Spanish Nuclear Regulatory Authority (CSN) are working to identify specific aspects of the licensing process that could be necessary for long term operation of NPP.

This paper presents a summary of the present Spanish Regulatory framework that could be applied to long term operation of NPP and the methodology developed jointly by all the Spanish utilities, under UNESA coordination, for setting Life Management Programmes.

2. SPANISH REGULATORY FRAMEWORK

The Spanish NPPs in operation are:

NPP	Type	Capacity [gross MWe] (as of June 2002)	Commercial Operation
Jose Cabrera (Zorita)	PWR	160	1969
Sta. Mª Garoña	BWR	466	1971
Almaraz 1	PWR	974	1983
Almaraz 2	PWR	983	1984
Ascó 1	PWR	1,028	1984
Ascó 2	PWR	1,027	1986
Cofrentes	BWR	1,080	1985
Vandellos II	PWR	1,087	1988
Trillo	PWR	1,066	1988

The current Spanish nuclear regulatory framework is based on periodical renewal of the NPP operating licence, with no specific time limit for operation. These renewals are done usually every ten years, although this period can be shorter if the CSN considers the NPP does not meet all the requirements to operate safely through this period.

The current regulatory process is based on three main actions:

- Continuous safety evaluations of the NPP by the CSN through surveillance, periodic inspections, etc.
- Preparation and submittal to the CSN of an annual report with updated results of the plant Life Management Programme.
- Performance of Periodic Safety Reviews (PSR) by the NPP for the licence renewal.

The main objectives of the PSR are:

- Analyse internal and external operating experience.
- Guarantee the correct application of operating experience analysis process, including the modifications global revision.

– Analyse the plant behaviour during a long operating period, including the results of surveillance requirements and equipment maintenance, in order to verify the safety level has not decrease and to guarantee the safety operation during then next period.
– Evaluate the plant safety level, considering the new national and international codes and standards, in particular those applied in the technology origin country to the similar NPPs.
– Update the safety evaluation and improvement programmes.

From the point of view of life management, the PSR have to include the analysis of the critical component behaviour, identifying the ageing or degradation mechanisms and the current corrective measures adopted by the plant to control and mitigate them, as well as updating the safety evaluation and improvement programmes.

PSR results have to be reported in a specific document produced by the NPP, including the analysis of all areas under the scope of the PSR, and identifying the safety improvement actions and its implementation schedule.

PSR requirements are stated in a specific guide (CSN Safety Guide No. 1.10) published by the CSN in December 1995. This Safety Guide has been prepared following international practices and recommendations, including IAEA Safety Guide "Periodic Safety Review of Operational NPP".

Although currently under discussion with the CSN, the Spanish utilities do not foresee that the long-term operation of NPP will introduce significant changes in the regulatory process for licence renewal.

3. SPANISH UTILITY APPROACH FOR NPP LIFETIME MANAGEMENT

In order to facilitate the application of CSN Safety Guide 1.10 "Periodic Safety Reviews of Nuclear Power Plants", the Spanish utilities association (UNESA) has developed a specific guide, titled "Guideline for the development of Periodic Safety Review".

The areas considered within the UNESA guide are the following:

– Operating experience
– Experience related to the radiological impact
– Changes in the regulations and laws
– Equipment behaviour
– Installations modifications
– Probabilistic safety assessment
– Updating of safety evaluation and improvement programmes

The two areas related to ageing and life management are equipment behaviour and updating of safety evaluation and improvement programmes.

The approach to life management has also been standardised by UNESA by the development of a common methodology. This methodology has the following phases (see Fig. 1):

1. Selection of systems, structures and components.
2. Identification and study of ageing mechanisms.
3. Maintenance evaluation.
4. Review of operation and maintenance programmes.

For an NPP long term operation (beyond 40 years design life) it is foreseeable that the mentioned methodology be reviewed in order to adapt it to new requirements, according to the currently discussions between CSN and Spanish utilities.

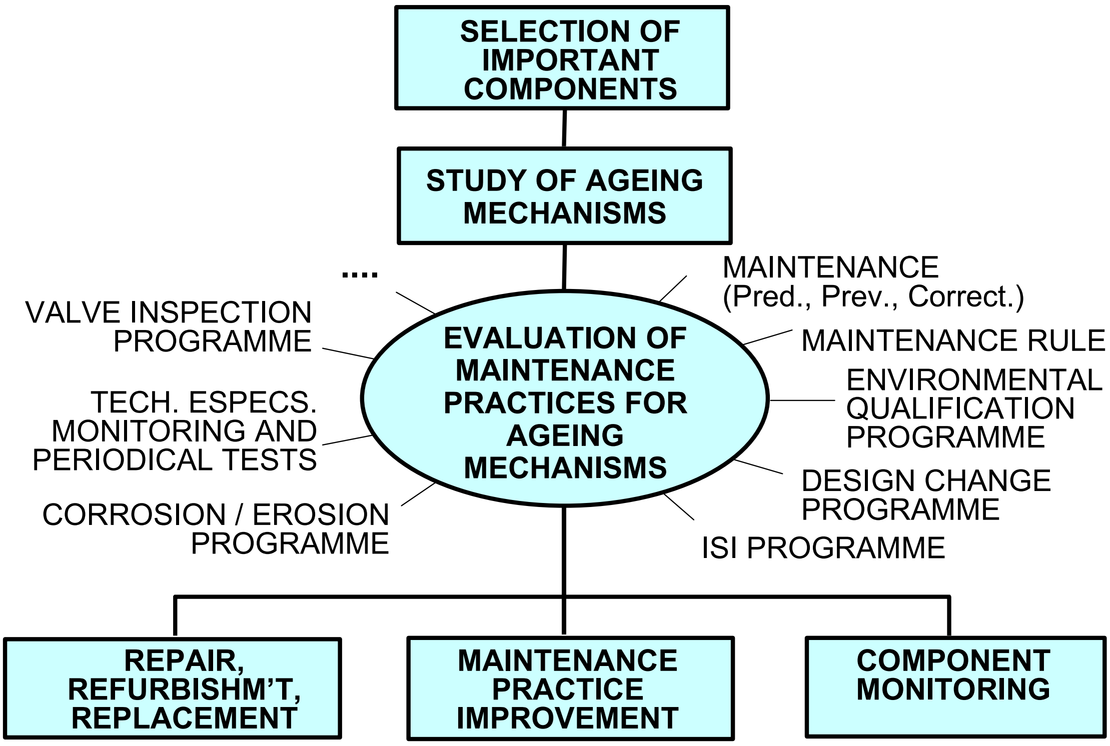

FIG. 1 Spanish NPP Life Management Methodology

3.1. Selection of systems, structures and components

As basic criteria Life Management Programmes will focus in components having the greatest sensitivity to ageing and higher operating and maintenance (O&M) costs. These components will be the object of further investigations and research to identify the parameters that affect their life cycle.

The selection of important components for lifetime management is based on two steps (see Fig. 2): Selection of systems and components and Grouping of similar components.

The selection of systems is done considering safety, availability and cost criteria. The selection of components within these systems is performed based on a wide range of criteria such as:

– Significant impact on safety level of component failure.
– Component is safety-related or required for safe shut-down
– Important component in licensing process
– More aggressive than design operating and environmental conditions.

– Component maintenance is not effective for ageing control and mitigation
– High cost or long period needed for component replacement.

If wanted, the components selected can be ranked following several weighted criteria such as: regulatory factors, reliability considerations, maintenance programme effectiveness, service conditions and history, etc.

Finally the establishment of grouping criteria allows inclusion in the same class of components with similar surveillance parameters and techniques. This allows for a more effective and efficient ageing surveillance and management of these grouped components, as they require similar parameters and techniques.

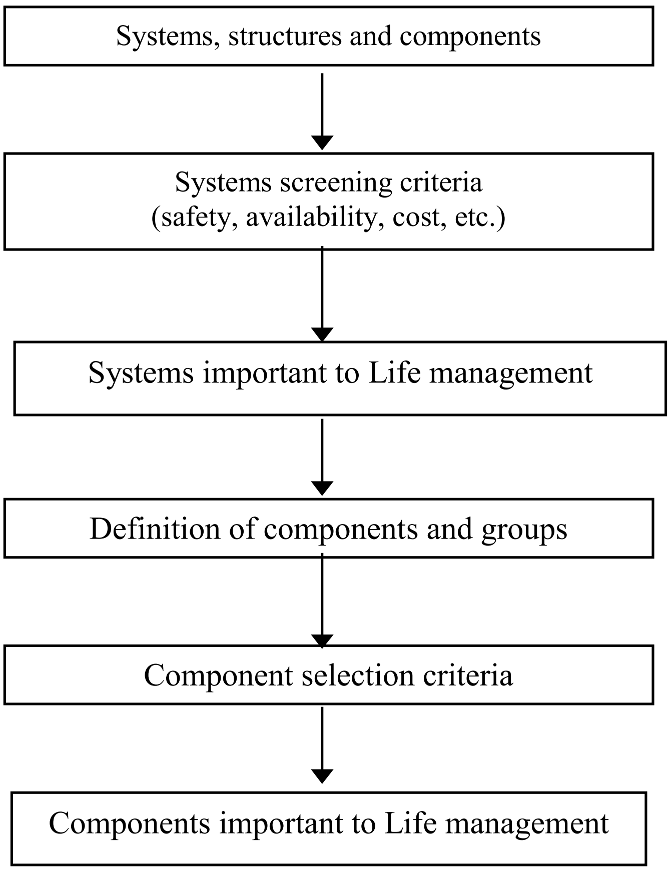

FIG. 2. Components selection

As an example, in Table 1 a list of the first 15 components for each kind of reactor is shown.

Table 1. List of more significant components selected and their prioritisation

	Component/ Group/Structure
	PWR Plant
1	Reactor Pressure Vessel
2	Steam Generators
3	Reactor Building
4	Reactor Vessel Pedestal
5	Metal Primary Containment
6	Auxiliary Building
7	Reactor Coolant Pumps Foundation
8	Pressurizer Pedestal
9	Steam Generator Pedestal
10	Essential Service Water Piping
11	Primary Containment Mechanical Penetration Assemblies
12	Pressurizer
13	Intake Structure
14	Primary System Equipment Supports
15	Electrical Building
	BWR Plant
1	Reactor Pressure Vessel
2	Reactor Vessel Pedestal
3	Drywell Metal Shell Foundation
4	Reactor Building Basement
5	Suppression Chamber including support
6	Plant Control Centre
7	Biological Shield
8	Fuel Pool Slabs and Walls
9	Drywell Metal Shell
10	Snubbers
11	Sacrificial Shield Wall
12	Reactor Recirculation Piping
13	Control Rod Driver
14	Reactor Core Shroud
15	Reactor Vessel Nozzle Safe Ends

3.2. Identification and study of ageing mechanisms

This programme phase begins with an initial condition evaluation, which serves as the basis for establishing the main preventive/corrective maintenance and monitoring actions, and for preparing the first cost/benefit analyses for Life Management. The Programme continues to progress with periodic re-evaluation of condition to confirm the corrective measures are the right ones and to adopt new measures, if necessary, as a result of the monitoring established.

The initial conditional evaluation of each component is performed considering the following information:

– Component (or group of components) design, manufacturing and operational data, including process and service conditions, stress values, etc.
– Potential degradation mechanisms and the level of harshness of selected components is determined based on previous collected data.
– The degradation mechanisms analysis is complemented by history of the operation and maintenance, and the results of diagnosis and monitoring, to detect incidents that might have affected the condition of the plant, or for evidence of degradations.
– Uncertainty about the severity of some of these ageing effects may require extra inspections or tests, to provide more precise data.

Condition evaluation requires collection and ordering of the documentation and records of manufacturer, operation and maintenance that contain information needed for the analysis. This collection requires application of procedures that establish the data and records, with the periodicity of their acquisition clearly identified for successive re-evaluations, and the screening requirements for easier collection and analysis.

The result of this analysis is an Evaluation Report (Fig. 3) for each component or group of similar components. In addition, each evaluation report includes information about the following age related matters for the target component(s):

– Component detailed description, including design and service information, potential degradation mechanisms and main variables controlling ageing process.
– Techniques allowing to detect, survey and monitor the component ageing mechanisms, including inspection, surveillance and periodic testing.
– Lifetime prediction methodology, including life consumption assessment algorithms, degradation status determination and evolution calculations and acceptance criteria for ageing prediction.
– Recommendation for ageing mitigation: different theoretical approach to mitigate ageing effects are proposed in evaluation reports, including new research and development activities to solve existing problems in mitigation techniques.

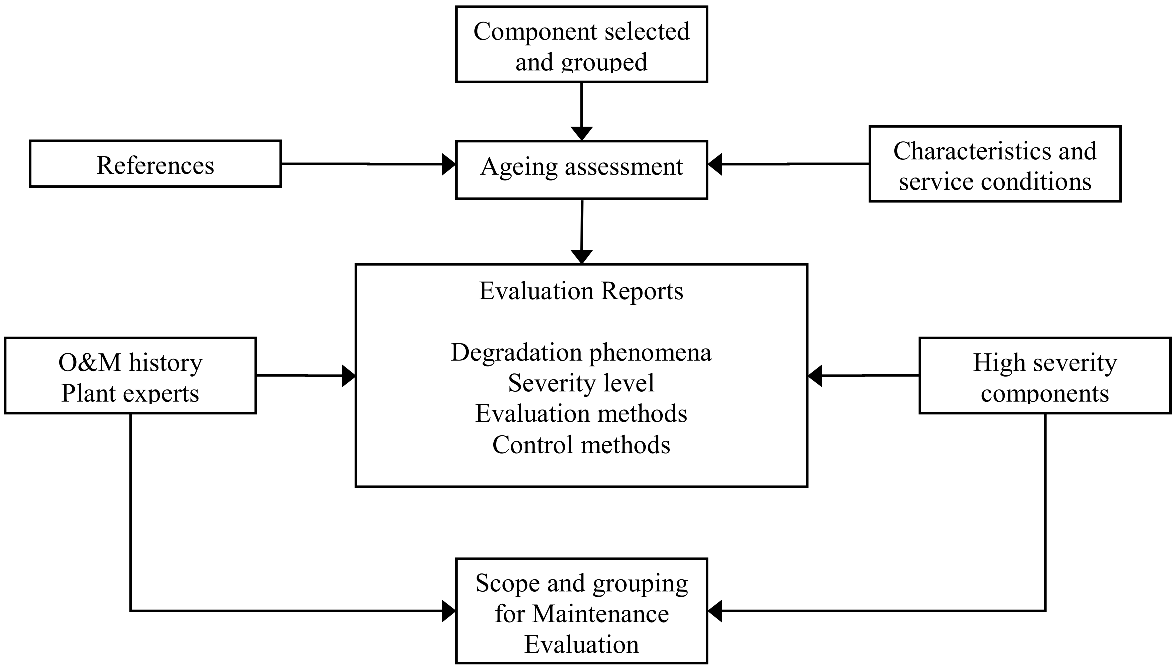

FIG. 3. Ageing evaluation process

Examples of component–degradation mechanisms identified for Spanish NPPs are included in Table 2.

Table 2. Examples of component - degradation mechanism pairs identified

Component	*Degradation Mechanisms*
PWR Plant	
Reactor pressure vessel	Neutron embrittlement, fatigue, IGSCC, Wear
Steam generators	Fretting, SCC, Fatigue; Fouling
Pressurizer	Fatigue, IGSCC, Electrical ageing
RPV Internals	Wear, IGSCC, IASCC
Containment mechanical penetration assemblies	Galvanic corrosion
RPV support	Corrosion, Fatigue, SCC
Emergency diesel generator (engine)	Wear, Fatigue, Corrosion, Stress relaxation, Fouling, Erosion, MIC.
BWR Plant	
Reactor Pressure Vessel	Neutron embrittlement, fatigue, IGSCC
RPV Internals	Wear, IGSCC, IASCC, Fatigue
Containment mechanical penetration assemblies	Galvanic corrosion
Metal containment	Corrosion, Galvanic corrosion
Drywell vent lines including bellows	Fatigue
Reactor building crane	Overload

3.3. Maintenance evaluation

In addition to the improvements in operation and service conditions, a substantial part of the causes and effects of ageing mechanisms have to be mitigated by maintenance work. The nature of these long-term ageing mechanisms has meant that, in certain cases, current maintenance practices do not prevent them. This requires these practices to be evaluated and modified where necessary to improve their efficiency in conservation and the mitigation of degradation.

The maintenance evaluation methodology requires the following activities:

- Detection of the component-degradation mechanisms pairs to evaluate.
- Determination of the programmes, practices and procedures that affect each component/degradation mechanism pair
- Evaluation of the possible deficiencies in control of ageing of the maintenance of each components and propose, when necessary maintenance improvements.

These activities are materialised in the following procedure:

- Production of Component Degradation Data Sheets (CDDS): The data to be filled out on the CDDS are component description; functions; design parameters; operating experience; degradation mechanisms; and the part of the component affected by ageing.

- Generation of Maintenance Practices Data Sheets (MPDS): For each programme, practice and procedure that affect each component-degradation mechanism pair, a MPDS is prepared with: practice limitations; time when corrective action is taken; data needs; mitigation, detection and monitoring actions; and criteria, as well as any other experience or comments from practices application experience.
- Creation of Maintenance Evaluation Checklist (MEC). Each component CDDS and all the MPDS that affect the component are compared to obtain the MEC of the component. MEC shows the possibly deficiencies in ageing control of each component maintenance. When required, maintenance improvements are proposed. The most extended actions are changes in service condition, fluid chemistry control and environmental conditions, as well as, improvement or recovery of material characteristics and modification in operation modes.

3.4. Review of operation and maintenance programmes

Based on the conclusions of the previous activity, several actions can be taken in order to review the operation and maintenance programmes. These actions could be the following:

- Repairs, replacement or modifications of the components most severely affected, if availability or performance improvement justify the investment.
- Modification of operating procedures and in service conditions to make them less harsh.
- Improve maintenance practices to achieve full efficiency for safe and economically viable life extension.
- Implement additional monitoring to improve condition evaluation and trends, especially for component–degradation mechanism combinations with more uncertainties. This will reduce the effort required for collection and analysis of information, and allow the use of realistic criteria for ageing parameters in the life management decision-making process.

Examples of maintenance programmes optimisation are the extensive effort in older plants to replace old cable and instrumentation, which were considered age-sensitive. An example of the mitigation activities is the core optimisation, which reduces reactor pressure vessel neutron embrittlement.

4. INTEGRATED AGEING MANAGEMENT SOFTWARE TOOL.

In order to apply the methodology adopted in an easy and homogeneous way and to take advantage from previous developments performed within the sector, the Spanish utilities decides to develop jointly a computer system to facilitate condition evaluation as well as component life management of a NPP. This project was known as Integrated System for Life Management of NPPs (SIGEVI project) and took place from 1998 to 2001.

The project, in which all the Spanish NPP participated, was co-ordinated by UNESA and developed by different Spanish engineering companies. Vandellos II NPP was used as pilot plant for the development of the first prototype that included the following components: vessel, steam generator, pressurizer, vessel internals, diesel generator, turbine-generator set, transformer, relays and breakers, pumps, electrical machines, hydraulic valves, wiring, structures, and piping. Each of them constitutes a system module.

In each module several evaluation techniques are included. These have different complexity degrees, from just presenting parameters tendency to damage estimation or defect analysis, using fracture mechanics studies. In Figures. 4 and 5 examples of computer screens of the SIGEVI are presented.

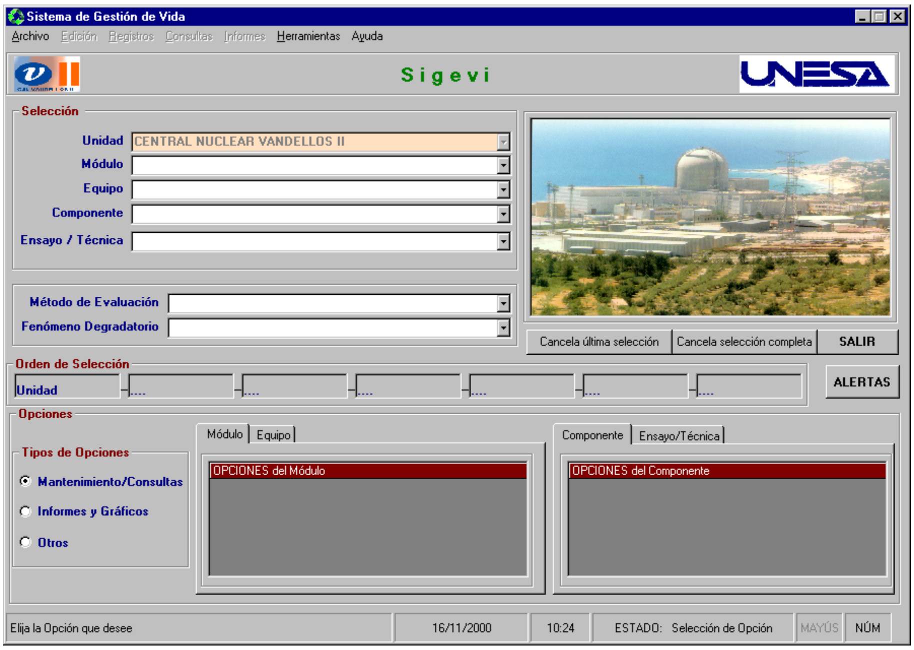

FIG.4. Integrated lifetime management application main screen

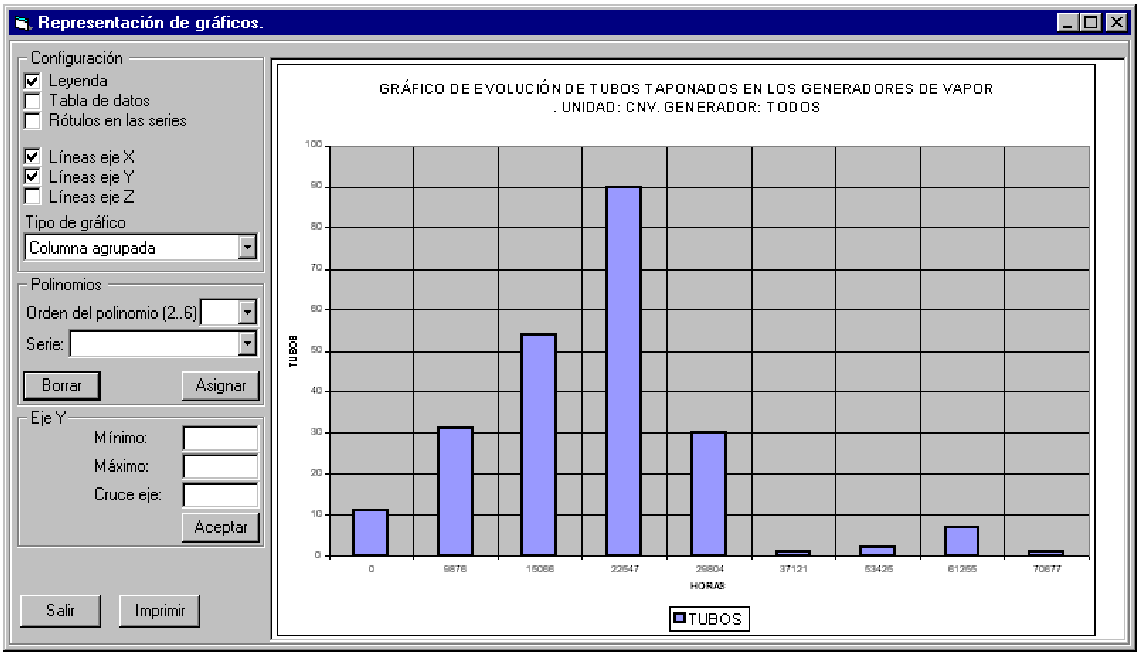

FIG. 5. Mitigation measure efficiency surveillance: Steam generator plugged tubes vs. time

5. CONCLUSIONS

1. Spanish utilities consider a strategic objective to operate their NPP on the long term, and so have set up a common Plant Life Management methodology that is developed through specific Life Management Programmes.
2. The current Spanish regulatory framework is based on periodical renewal of the NPP operating licence, without a specific time limit for operation. These renewals are done usually every ten years, although this period can be shorter if the Spanish Nuclear Regulatory Authority (CSN) considers the NPP does not meet all the requirements to operate safely through this period.
3. The CSN and the utilities are working to identify specific aspects of the licensing process that could be necessary for long term operation of NPP. Although currently under discussion, the Spanish utilities do not foresee that the long term operation of NPP will introduce significant changes in the regulatory process for licence renewal.
4. The actual common Plant Life Management methodology is divided into four main activities:
   - Definition of the priorities and scope of the Life Management Programme. Plant systems, structures and components important for lifetime management are selected and prioritised using a methodology based on screening criteria.
   - Evaluation of the initial condition as the basis for identifying ageing mechanisms and establishing the main preventive/corrective maintenance and monitoring actions. Periodic re-evaluation of condition confirm the corrective measures are the right ones or lead to adopt new measures, if necessary, as a result of the monitoring established.
   - In addition to the improvements in operation and service conditions, effects of ageing mechanisms have to be mitigated by maintenance work. The nature of these long-term ageing mechanisms is such that, in certain cases, current maintenance practices do not prevent them. This requires these practices to be evaluated and modified when necessary.
   - Based on results of previous activities, it is possible to propose measures to review the Operation and Maintenance programmes to optimise lifetime management.
5. It is foreseeable that the mentioned methodology be reviewed for an NPP long-term operation (beyond 40 years design life), in the framework of licensing process discussions, in order to adapt it to new regulatory requirements.
6. Finally, software tools to facilitate ageing assessment and condition evaluation for main important component for lifetime management have been developed and are available for NPPs.

REFERENCES

[1] IAEA, "Methodology for Ageing Control in Components Important for Nuclear Safety". Technical Report Series No. 338.

[2] CSN, "Revisiones Periodicas de la Seguridad de las centrales nucleares españolas". Guia de Seguridad No. 1.10, Diciembre 1995.

[3] IAEA, "Periodic Safety Review of Nuclear Power Plants: A Safety Guide". Safety Series No. 50-SG-012. Vienna, 1994.

[4] UNESA, "Guia para el desarrollo de Revisiones Periodicas de Seguridad". Enero 1997.

[5] UNESA, "Guia de seleccion de componentes". Documento GVR-EA-GU-00101.

[6] UNESA, "Guia de Evaluacion de Practicas de Mantenimiento". Documento GVR-EA-GU-04101.

[7] UNESA, "Estudio de Mecanismos de Degradacion de las Centrales Piloto". Documentos GVE-EA/IN-IT-03301 a 03317
[8] UNESA, "Evaluacion de Practicas de Mantenimiento de las Centrales Piloto". Documentos GVR-EA-IT-04401 and 04402
[9] CEC-ENV, "Safe management of NPP ageing in European Union". Final Report CONTRACT Nº 98/872/3040/DEB/C2.
[10] Nuclear Energy Institute, "Industry Guideline for Implementing the Requirements of 10CFR54". NEI 95-10 Rev. 3, April 2001.
[11] L. Francia, A. Cardoso, "Integrated tool for NPP Lifetime Management in Spain".

THE UNITED STATES NRC LICENSE RENEWAL PROCESS

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Abstract

The U.S. NRC license renewal process establishes the technical and administrative requirements for renewal of operating power plant licenses. Reactor operating licenses were originally issued for 40 years and are allowed to be renewed for an additional 20 years. The review process for renewal applications provides continued assurance that the level of safety provided by an applicant's current licensing basis is maintained for the period of extended operation. The license renewal review focuses on passive, long-lived structures and components of the plant that are subject to the effects of aging. The applicant must demonstrate that programs are in place to manage those aging effects. The review also verifies that analyses that are based on the current operating term have been evaluated and shown to be valid for the period of extended operation. The NRC has renewed the licenses for 10 reactors at 5 plant sites. Applications to renew the licenses of 16 additional reactors at 10 plant sites are under review. If the applications currently under review are approved, 25 percent of the licensed operating reactors will have extended their lifespan by 20 years. As license renewal is voluntary, the decision to seek license renewal and the timing of the application is made by the licensee. However, the NRC expects that, over time, essentially all U.S. operating reactors will request license renewal.

1. INTRODUCTION

Based on the Atomic Energy Act [1], the United States (U.S.) Nuclear Regulatory Commission (NRC) issues licenses for commercial power reactors to operate for up to 40 years and allows these licenses to be renewed. A 40-year license term was selected on the basis of economic and antitrust considerations -- not technical limitations. However, once the license term was selected, individual plant designs may have been engineered on the basis of an expected 40-year service life.

There are currently 103 licensed, operating commercial nuclear power plant reactors in the U.S. One additional licensed reactor, Browns Ferry Unit 1 is currently undergoing restart from an extended shutdown. The first 40-year operating licenses will expire for four reactors in the year 2009. More than 30 percent of the 99 remaining operating reactors will expire by the year 2015. License renewal is voluntary and the decision whether to seek license renewal rests entirely with nuclear power plant owners. This decision is typically based on the plant's economic situation and whether it can continue to meet NRC requirements.

The NRC has established a license renewal process that can be completed in a reasonable period of time with clear requirements to assure safe nuclear plant operation for up to an additional 20 years of plant life. Currently, nuclear power provides approximately 20 percent of the electricity generated in the U.S. and is the second largest source of electrical generation.

2. BACKGROUND

In the 1980s, the NRC established a comprehensive program for nuclear plant aging research. Based on the results of that research, a technical review group concluded that many aging phenomena are readily manageable and do not pose technical issues that would preclude life extension for nuclear power plants.

In 1991, the NRC first published safety requirements for license renewal as Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54 (the license renewal rule) [2]. The NRC then undertook a demonstration program to apply the rule to pilot plants and develop experience to establish implementation guidance. To establish a scope of review, the rule defined age-related degradation unique to license renewal. However, during the demonstration program, the NRC found that many aging effects are dealt with adequately during the initial license period. In addition, the NRC found that the review did not allow sufficient credit for existing programs, particularly the maintenance rule (10 CFR 50.65) [3], which also helps manage plant aging phenomena.

As a result, in 1995, the NRC amended the license renewal rule. The amended Part 54 established a regulatory process that is more efficient, more stable and more predictable than the previous license renewal rule. In particular, Part 54 was clarified to focus on managing the adverse effects of aging. The rule changes were intended to ensure that important systems, structures, and components will continue to perform their intended function during the 20-year period of extended operation.

NRC's responsibilities under the U.S. National Environmental Policy Act [4] call for a review of the impact of license renewal on the environment. In parallel with aging efforts, the NRC pursued a separate rulemaking to revise its environmental regulation, 10 CFR Part 51 [5], to focus the scope of review of environmental issues.

3. RENEWAL PROCESS

The license renewal process proceeds along two tracks -- one for review of safety issues (Part 54) and another for environmental issues (Part 51). An applicant must provide the NRC with an evaluation that addresses the technical aspects of plant aging and describes the ways those effects will be managed. It must also prepare an evaluation of the potential impact on the environment if the plant operates for another 20 years. The NRC staff reviews the application and verifies the safety evaluations through inspections.

Public participation is an important part of the license renewal process. There are several opportunities for members of the public to question how aging will be managed during the period of extended operation. Information provided by the licensee is made available to the public. A number of public meetings are held by the NRC, and NRC evaluations, findings and recommendations are published when completed. All public meetings are posted on NRC's web site and key meetings are announced in press releases and in the *Federal Register*. Concerns may be litigated in a formal adjudicatory hearing if any party that would be adversely affected requests a hearing. In addition, members of the public may petition the NRC for consideration of issues other than the management of the effects of aging during the period of extended operation of the plant.

A nuclear power plant licensee may apply to the NRC to renew its license as early as 20 years before expiration of its current license. License renewal is expected to take approximately 30 months, including the time to conduct an adjudicatory hearing, if necessary, or 22 months without a hearing. Upon receipt of a license renewal application, the review is conducted, in general, according to the following steps:

Licensing Milestones	Months Elapsed
Receive renewal application	0
Publish notice of opportunity for hearing	2.0
Conduct public meeting on scope of environmental impact statement	2.5
File hearing request	3.0
Issue environmental requests for additional information to applicant	5.0
Issue safety requests for additional information to applicant	8.0
Issue draft environmental impact statement for comment	11.0
Conduct public meeting on draft environmental impact statement	12.5
Issue safety evaluation report; identify open items	14.5
Receive responses to open items from applicant	17.0
Issue final environmental impact statement	18.0
Issue safety evaluation report supplement	19.0
Licensing Milestones (continued)	**Months Elapsed**
Complete Advisory Committee on Reactor Safety review	20.5
Decision on renewing license (without hearing)	22.0
Complete hearing process (if needed)	
Decision on renewing license (with hearing)	30.0

4. ENVIRONMENTAL REVIEWS

Environmental protection regulations were revised in December 1996 to facilitate the environmental review for license renewal. Certain issues are evaluated generically for all plants, rather than separately in each plant's renewal application. The NRC's evaluation, Generic Environmental Impact Statement for License Renewal of Nuclear Plants (GEIS) [6], assesses the scope and impact of environmental effects that would be associated with license renewal at any nuclear power plant site. A plant-specific supplement to the generic environmental statement is required for each licensee that applies for license renewal.

The NRC performs plant-specific reviews of the environmental impacts of license renewal in accordance with the National Environmental Policy Act [4] and the requirements of 10 CFR Part 51 [5]. A public meeting is held near the nuclear power plant seeking renewal to "scope out" or identify environmental issues specific to the plant for the license renewal action. The result is an NRC recommendation on whether or not the environmental impacts are so great that they should preclude license renewal. This recommendation is presented in a draft plant-specific supplement to the GEIS which is published for comment and discussed at a separate public meeting. After consideration of comments on the draft, NRC prepares and publishes a final plant-specific supplement to the GEIS.

The NRC issued a supplement to the environmental standard review plan [7], which provides guidance on how to review the environmental portions of renewal applications. The NRC also issued a supplement to the environmental regulatory guide [8], identifying the format and content of environmental reports which must accompany license renewal applications.

5. SAFETY REVIEWS

5.1. Principles and Process

The license renewal rule rests on the determination that current operating plants continue to maintain an adequate level of safety and over the lives of the plants, this level has been enhanced through maintenance of the current licensing basis, with appropriate adjustments to address new information from industry operating experience. Additionally, regulatory activities have provided ongoing assurance that the current licensing basis will continue to provide an acceptable level of safety. Based on this determination, the NRC established two fundamental principles for license renewal:

(1) The regulatory process is adequate to ensure that the licensing basis of all currently operating plants provides and maintains an acceptable level of safety so that operation will not be inimical to public health and safety or common defense and security. A possible exception may be the detrimental effects of aging on the functionality of certain systems, structures and components, and possibly a few other issues that arise only during the period of extended operation, and

(2) Each plant's licensing basis must be maintained during the renewal term in the same manner and to the same extent as during the original licensing term.

The NRC regulatory process ensures the safety of operating plants through a variety of methods. These methods include the issuance of NRC Bulletins, Generic Letters, Information Notices, and new or revised regulations which require action by licensees. These activities are supported by a number of special NRC inspections that are performed in addition to the continuous oversight and routine inspection activities performed by the NRC on-site inspectors. Because of this comprehensive regulatory process, compilation of the current licensing basis or re-verification of the current licensing basis is not considered necessary for license renewal.

The license renewal rule defines the technical and administrative process for evaluating the effects of aging on system, structure, and component performance, and for developing a license renewal application. There are two major safety assessments that an applicant must perform and submit in a license renewal application: 1) an integrated plant assessment and 2) an assessment of time-limited aging analyses. These assessments are in addition to the requirement to update the plant's Final Safety Analysis Report to include a summary description of the programs and activities for managing the effects of aging and the evaluation of time-limited aging analyses, and any revisions to the plant's operating technical specifications that may be required.

5.2. Integrated Plant Assessment

The applicant's integrated plant assessment must demonstrate that the structures and components within the nuclear power plant requiring aging management review have been identified and that the effects of aging on the functionality of such structures and components will be managed to maintain the current licensing basis to ensure that an acceptable level of safety will be maintained for the period of extended operation. The first step in the process is to identify all plant systems, structures, and components that are within the scope of the license renewal rule. The scope of the rule is defined as those systems, structures, and components that are:

(1) Safety-related and relied upon to ensure the following functions:
  (i) Maintain the integrity of the reactor coolant pressure boundary,
  (ii) Ensure the capability to shut down the reactor and maintain it in a safety shutdown condition, or
  (iii) Prevent or mitigate offsite radioactive exposures comparable to limits specified in the regulations.

(2) Non-safety related and whose failure could prevent satisfactory accomplishment of safety-related functions.

(3) Relied upon to perform a function that demonstrates compliance with the NRC's regulations for fire protection (10 CFR 50.48) [9], environmental qualification (10 CFR 50.49) [10], pressurized thermal shock (10 CFR 50.61) [11], anticipated transients without scram (10 CFR 50.62) [12], and station blackout (10 CFR 50.63) [13].

Once the systems, structures, and components that are within the scope of the license renewal rule are identified, the applicant must identify those structures and components that are subject to an aging management review. Structures and components subject to an aging management review are those that are passive and long-lived. Passive structures and components perform their intended function without moving parts or without a change in configuration or properties. Long-lived structures and components are those that are not subject to replacement based on a qualified life or a specified time period. Passive and long-lived structures and components include the reactor vessel, reactor coolant system piping, steam generators, pressurizer, pump casings, valve bodies, containment building, electrical cables, and electrical cabinets.

The NRC determined that structures and components with only active functions could be generically excluded from a license renewal review because functional degradation resulting from the detrimental effects of aging in active components is more readily detected and corrected by routine surveillance, performance indicators, and maintenance. Surveillance and maintenance programs for active components are required throughout the period of extended operation. Active components include equipment such as motors, diesel generators, control rod drives, cooling fans, switchgear, breakers, batteries, relays, and switches.

The renewal application must contain a list of all structures and components subject to an aging management review and a description of the methodology used to identify them. The application must also identify the materials of construction, the environment in which the structure or component must function, and all potential aging effects. The applicant is then required to demonstrate that aging management programs exist such that the effects of aging will be managed so that the intended functions of those structures and components will be maintained for the period of extended operation.

For some passive structures and components within the scope of the renewal evaluation, no additional action may be required where an applicant can demonstrate that the existing programs provide adequate aging management throughout the period of extended operation. However, if additional aging management activities are warranted for a structure or component within the scope of the rule, applicants will have the flexibility to determine appropriate actions. These activities could include, for example, adding new monitoring programs, increasing inspections, or revising design criteria to manage the effects of aging.

5.3. Time-Limited Aging Analyses

Another requirement for license renewal is the identification and update of time-limited aging analyses. During the design phase for a plant, certain assumptions about the length of time the plant would be operated are made and incorporated into design calculations of certain plant systems, structures, and components. Under a renewed license, these calculations must be shown to be valid for the period of extended operation.

Time-limited aging analyses are defined as those licensee calculations and analyses that meet all six of the following criteria:

(1) Involve systems, structures, and components within the scope of the license renewal rule;
(2) Consider the effects of aging;
(3) Involve time-limited assumptions defined by the current operating term (40 years);
(4) Were determined to be relevant by the licensee in making a safety determination;
(5) Involve conclusions or provide the basis for conclusions related to the capability of system, structure, or component to perform an intended function; and
(6) Are contained or incorporated by reference in the plant's current licensing basis.

The license renewal application must contain a list of all time-limited aging analyses that were identified. The application must also contain a demonstration that for each analysis:

(1) The analysis remains valid for the period of extended operation;
(2) The analysis has been projected to the end of the period of extended operation; or
(3) The effects of aging on the intended functions will be adequately managed for the period of extended operation.

Typical time-limited aging analyses that must be evaluated include reactor vessel neutron embrittlement, metal fatigue, environmental qualification of electrical equipment, concrete containment tendon pre-stress, and containment liner plate and penetration sleeve fatigue.

6. SAFETY REVIEW GUIDANCE DOCUMENTS

To facilitate the review of license renewal applications, the NRC established a streamlined process for reviewing applications consistently and expeditiously. The license renewal process was developed using input from interested parties within the nuclear industry and the public and was documented in three improved NRC guidance documents, namely, the standard review plan for license renewal [14], Generic Aging Lessons Learned (GALL) Report [15], and regulatory guide for the standard format and content of renewal applications [16]. These guidance documents are based on NRC research activities and the experience gained from the applicant's preparation of renewal applications and the NRC's review of the applications. These documents provide guidance for future renewal applicants in preparing applications, for the NRC staff in performing reviews, and for the public by informing them of the license renewal process.

The GALL Report documents the basis for determining when existing programs are adequate and when existing programs should be augmented for license renewal. During the review of the first renewal applications, the NRC and industry recognized that many of the existing programs at the plants could adequately manage aging effects for license renewal without change. Therefore, the NRC staff undertook a generic review and a technical evaluation of existing plant programs to determine which programs would adequately manage aging effects without change and which programs would need to be augmented. The technical evaluation is documented in the GALL Report.

The GALL report is referenced in the standard review plan for license renewal as the technical basis for identifying those programs that warrant particular attention during NRC's review of a license renewal application. The standard review plan provides guidance to NRC reviewers performing the safety review of license renewal applications. The principal purposes of the standard review plan are to ensure the quality and uniformity of reviews and to present a well-defined base from which to evaluate renewal applicant programs and activities for the period of extended operation. The standard review plan is also intended to make information about regulatory matters widely available in order to enhance communication with interested members of the public and the nuclear power industry.

The NRC also issued the license renewal regulatory guide which provides applicants with the format and content for a license renewal application. The regulatory guide endorses a guideline for implementing the requirements of the license renewal rule [17], prepared by an industry organization, the Nuclear Energy Institute (NEI).

The NRC has established a process to document lessons learned from ongoing license renewal reviews and to use these lessons to continue to improve the efficiency and effectiveness of the license renewal process. Experience gained is disseminated as interim staff guidance for use by licensees and other interested parties until the guidance can be incorporated into the guidance documents. The guidance documents and the interim staff guidance process have helped improve the efficiency of the license renewal process by (1) clearly defining the scope of information required from applicants, (2) clearly defining the scope of the NRC's review, and (3) focusing attention on issues related to aging.

The guidance documents were developed to be used collectively and to provide consistent guidance to NRC reviewers and license renewal applicants. While the standard review plan, GALL Report, regulatory guide, and NEI guideline are not mandatory NRC requirements, they represent an acceptable method for meeting the requirements of the license renewal rule. The NRC will continue to update these documents as lessons are learned and process improvements are identified from the review of renewal applications and from operating experience. Based on process efficiencies identified to date, the NRC expects to significantly reduce the resources required to review license renewal applications and still maintain established schedules.

7. INSPECTIONS

The NRC has established an inspection program for license renewal that verifies, at the plant and in the applicant's engineering offices, that aging management programs have been implemented consistent with the renewal application and the NRC's safety evaluation and that the programs are effective at managing the effects of aging. The inspections sample the results of the process used by the licensee to identify those structures and components within the scope of license renewal, aging management programs, and design analysis changes. At a minimum, the NRC performs a □cooping inspection and an aging management inspection. An optional third inspection is performed, if needed, to complete any items remaining from previous inspections or from the review.

8. HEARINGS

The NRC's regulations permit any member of the public whose interest may be affected by the renewal of the plant's operating license to petition the NRC to conduct an adjudicatory hearing. The petition must set forth the interest of the petitioner in the proceeding and must identify the issue that the petitioner wishes to have litigated in a hearing. The petitioner needs to provide sufficient information with the request to demonstrate that a genuine dispute exists with the applicant on a specific issue of law or fact. Hearings are expected to be conducted on an efficient and reliable schedule while still ensuring fair resolution of contested issues.

9. INDUSTRY ACTIVITIES

The industry has submitted technical reports on particular license renewal topics for NRC approval. These reports identify a specific component or structure within the scope of license renewal, the materials of construction, applicable aging effects, and the programs credited for managing the identified aging effects. Activities that are plant-specific are identified as action items for each applicant referencing the report. This approach, along with compilations of past aging research programs, established a foundation of technical information that licensees used to evaluate the feasibility of license renewal, and later as references in license renewal applications.

With regard to pressurized water reactors, the Babcock & Wilcox Owners Group established a generic license renewal program which submitted generic license renewal reports on the reactor coolant system piping, the pressurizer, the reactor pressure vessel, and reactor vessel internals. The Westinghouse Owners Group also established a program for license renewal and submitted technical reports on the aging management activities for the reactor coolant system supports, the pressurizer, certain piping, the containment structure, and the reactor vessel internals. The Boiling Water Reactor Owners Group has concentrated its efforts on reports related to the reactor vessel internals program.

Industry representatives participated in working groups and technical committees, coordinated by the Nuclear Energy Institute, to address generic technical and process issues. The resolution of the generic renewal issues and lessons learned during the review of the initial renewal applications was documented and included in revisions to license renewal implementation guidance documents.

10. STATUS OF LICENSE RENEWAL APPLICATIONS

The NRC has renewed the licenses for 10 reactors at 5 plant sites (Calvert Cliffs Units 1 and 2; Oconee, Units 1, 2, and 3; Arkansas, Unit 1; Hatch, Units 1 and 2; and Turkey Point, Units 3 and 4). Currently, applications to renew the licenses of 16 additional reactors at 10 plant sites are under review (North Anna, Units 1 and 2; Surry, Units 1 and 2; Catawba, Units 1 and 2; McGuire, Units 1 and 2; Peach Bottom, Units 2 and 3; St. Lucie, Units 1 and 2; Fort Calhoun; Robinson, Unit 2; Ginna; and Summer). If the applications currently under review are approved, 25 percent of the licensed operating reactors will have extended their lifespan by 20 years.

The NRC expects that over time, essentially all U.S. operating reactors will request license renewal. As license renewal is voluntary, the decision to seek license renewal and the timing of the application is made by the licensee. However, experience to date is that the licensees applying for renewal have chosen to apply earlier in the 20 year eligibility period, rather than wait until later in the current license period.

The status of pending and planned applications as well as additional information on license renewal can be found on the NRC web site at: http://www.nrc.gov.

REFERENCES

[1] UNITED STATES, Atomic Energy Act of 1954

[2] UNITED STATES, *Code of Federal Regulations*, Title 10, Part 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants, 2002

[3] UNITED STATES, *Code of Federal Regulations*, Title 10, Part 50, Section 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, 2002

[4] UNITED STATES, National Environmental Policy Act of 1969

[5] UNITED STATES, *Code of Federal Regulations*, Title 10, Part 51, Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions, 2002

[6] U.S. NUCLEAR REGULATORY COMMISSION, Generic Environmental Impact Statement for License Renewal of Nuclear Plants, NUREG-1437, May 1996

[7] U.S. NUCLEAR REGULATORY COMMISSION, Standard Review Plans for Environmental Reviews for Nuclear Power Plants, NUREG-1555, Supplement 1, October 1999

[8] U.S. NUCLEAR REGULATORY COMMISSION, Regulatory Guide 4.2, Supplement 1, Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses, September 2000

[9] UNITED STATES, *Code of Federal Regulations*, Title 10, Part 50, Section 50.48, Fire Protection, 2002

[10] UNITED STATES, *Code of Federal Regulations*, Title 10, Part 50, Section 50.49, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants, 2002

[11] UNITED STATES, *Code of Federal Regulations*, Title 10, Part 50, Section 50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, 2002

[12] UNITED STATES, *Code of Federal Regulations,* Title 10, Part 50, Section 50.62, Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants, 2002

[13] UNITED STATES, *Code of Federal Regulations*, Title 10, Part 50, Section 50.63, Loss of All Alternating Current Power, 2002

[14] U.S. NUCLEAR REGULATORY COMMISSION, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, NUREG-1800, July 2001

[15] U.S. NUCLEAR REGULATORY COMMISSION, Generic Aging Lessons Learned (GALL) Report, NUREG-1801, July 2001

[16] U.S. NUCLEAR REGULATORY COMMISSION, Regulatory Guide 1.188, Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses, July 2001

[17] NUCLEAR ENERGY INSTITUTE, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - the License Renewal Rule (NEI 95-10), Revision 3, March 2001

PLANT LIFE MANAGEMENT (PLIM) IN SWISS NUCLEAR POWER PLANTS

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Abstract

In December 1991 the Swiss Federal Nuclear Safety Inspectorate (HSK) which was concerned with aging and its consequent safety effects in the nuclear plants, requested the Swiss utilities to start a systematic review of all safety relevant components and building structures in their plants. The review was required to evaluate individual effects of aging and was to be documented in an aging management programme (AMP). This programme was required to address the structures and mechanical and electrical equipment in the plants. It has to demonstrate, that at any time of plant operation the safety margins will not be violated by aging degradation of components and structures.

To fulfil this requirement a Swiss Utility Working Group (GSKL) for Aging Management was set up. The scope of a Swiss AMP was defined and key guideline documents have since been issued. So-called "Steckbrief"-files for several Safety Class 1 (SC1) have also been issued in the meantime. Those documents already issued are *Programme for Reviewing and Optimising Aging Management*, *Catalogue of Aging mechanism (KATAM), Guidelines for Preparing Component "Steckbrief"-files* and *Specimen „Steckbrief"-Files for Selected Components*. A "Steckbrief"-file is a summary report containing the component history, including the results of the reviews performed and measures taken or planned to counteract aging mechanisms for each safety relevant system.

The scope of these activities not only serves a reliable plant service but also facilitates component and plant life extension feasibility determination. The older plants have been in operation now for up to 30 years so that plant life extension (PLEX) is becoming an important topic of plant life management (PLIM) of the Swiss NPPs.

The operating life-time for nuclear plants in Switzerland is principally not restricted by time-limited operating licenses. The licensing authority requires that the utilities report periodically the plant's condition in order to prove that safety requirements are still fulfilled (periodic safety review, 10 years interval). The AMP is part of this assessment and with our understanding of the AMP as a plant life management tool it is planned to extend the operational life beyond the design life of 40 years to at least 50 (older plants) or even to 60 years (newer plants).

We see the Swiss AMP not only for its own purpose but also as a link to other programmes such as the probabilistic safety assessment (PSA). In setting priorities for SC2 and SC3 components, an important input is that information obtained from the PSA. For the possible implementation of risk informed applications such as RI-ISI, information such as degradation mechanisms resulting from the AMP can be applied in assessing failure potential in piping systems.

1. GENERAL

The operating life-time for nuclear plants in Switzerland is principally not restricted by time-limited operating licenses. The licensing authority requires that the utilities report periodically the plants condition in order to prove that the safety requirements are still fulfilled. All Swiss NPP’s have to perform Periodic Safety Review (PRS) every 10 years. The PSR in Switzerland must content (Regulatory rules, Guide HSK-R-48):

- Actually safety concept
- Judgement of operating management and operating experiences
- Deterministic safety analysis
- Probabilistic safety analysis

AMP = PLIM is a very important basis to fulfil this requirements

In December 1991 the Swiss Federal Nuclear Safety Inspectorate (HSK), concerned with aging and consequent safety aspects in the nuclear plants, requested the utilities to commence a systematic review of all safety relevant components and building structures regarding effects of aging and to set up an aging management programme. This programme should address the structures as well as the mechanical and electrical equipment of the plants. It has to be shown, that at any time operating the plant, the safety margins will be untouched.

2. SWISS UTILITY WORKING GROUP FOR AGING MANAGEMENT

To fulfil the requirement of the authority and to perform the first steps to reach this goal, a utilities working group was formed to set up a programme for a joint approach on aging matters. This group defined the basics of the Swiss Aging Management Programme (AMP) and described them in several documents:

- **Programme for Reviewing and Optimising Aging Management Measures**

This programme describes aging phenomena encountered in nuclear plants and provides an overview of aging management measures. In addition the programme defines the objectives of the working group and the applicable technical fundamentals.

- **Catalogue of Ageing Mechanism (KATAM)**

This catalogue categorises the different aging mechanisms. It defines all types of aging encountered in light-water reactor plants together with interactions between material, environment, medium and mechanical loading. Component reviews are performed on the basis of these criteria and categories listed in order to determine the endangered regions. The aging mechanisms for mechanical components are defined in a special catalogue. For electrical components the structure of the document is a little different because of the wide range of components involved. For this reason the catalogue constitutes part 1 of the electrical component „Steckbrief“.

For building structures the catalogue is issued as an Appendix to the guideline for preparing „Steckbrief“-file documents.

– **Guidelines for preparing component „Steckbrief“-files**

These guidelines are the working groups basic working document. They describe the methodical and technical approach for all three fields and aim at standardising reviews in the different plants. For the fields electrical components and building structures they are supplemented with special technical appendices, for example: aging modelling, thermal aging of polymers, useful life determination, detection methods, building structure anchors and interfaces in buildings, so that they can almost be considered textbooks.

– **Specimen „Steckbrief“-files for selected components**

These documents have been prepared as examples to assist and standardise the preparation of „Steckbrief“-files in the different plants. Safety class 1 components and buildings and electrical 1E qualified components, have been given the first priority. Specimen documents have been prepared for:

- Beznau NPP Reactor Recirculation System (mechanical component)
- Beznau NPP Reactor Building (building structure)
- Listing of all aging mechanisms which can be encountered in electrical components with details of detection methods. The various components are divided into groups such as cables, motors, transmitters, valve drives etc.

The group is divided into three subgroups, concerning with the aging of electrical, mechanical components and civil structures. So it was necessary to define the interfaces and the responsibilities of the groups and therefor an additional document was issued, the so called „interface document“. The basis structure and the responsibilities are shown in figure 1.

After the four documents, listed on the page before, had been issued, the plants started to review their systems, based on these documents. It was the main target of the working group to fulfil the authorities requirements which are quoted as follows in the programme document:

Aging management in accordance with the working group programme shall provide satisfactory proof that for all safety relevant components all known aging mechanisms will be taken into account in maintenance and quality assurance, and that measures will be taken for any omissions revealed.

In addition, aging management shall also provide a technical basis for optimising maintenance programmes, improving reliability of components and keeping maintenance costs down. One of the most important elements of aging management is condition monitoring to provide information on the present state of the component or building structure.

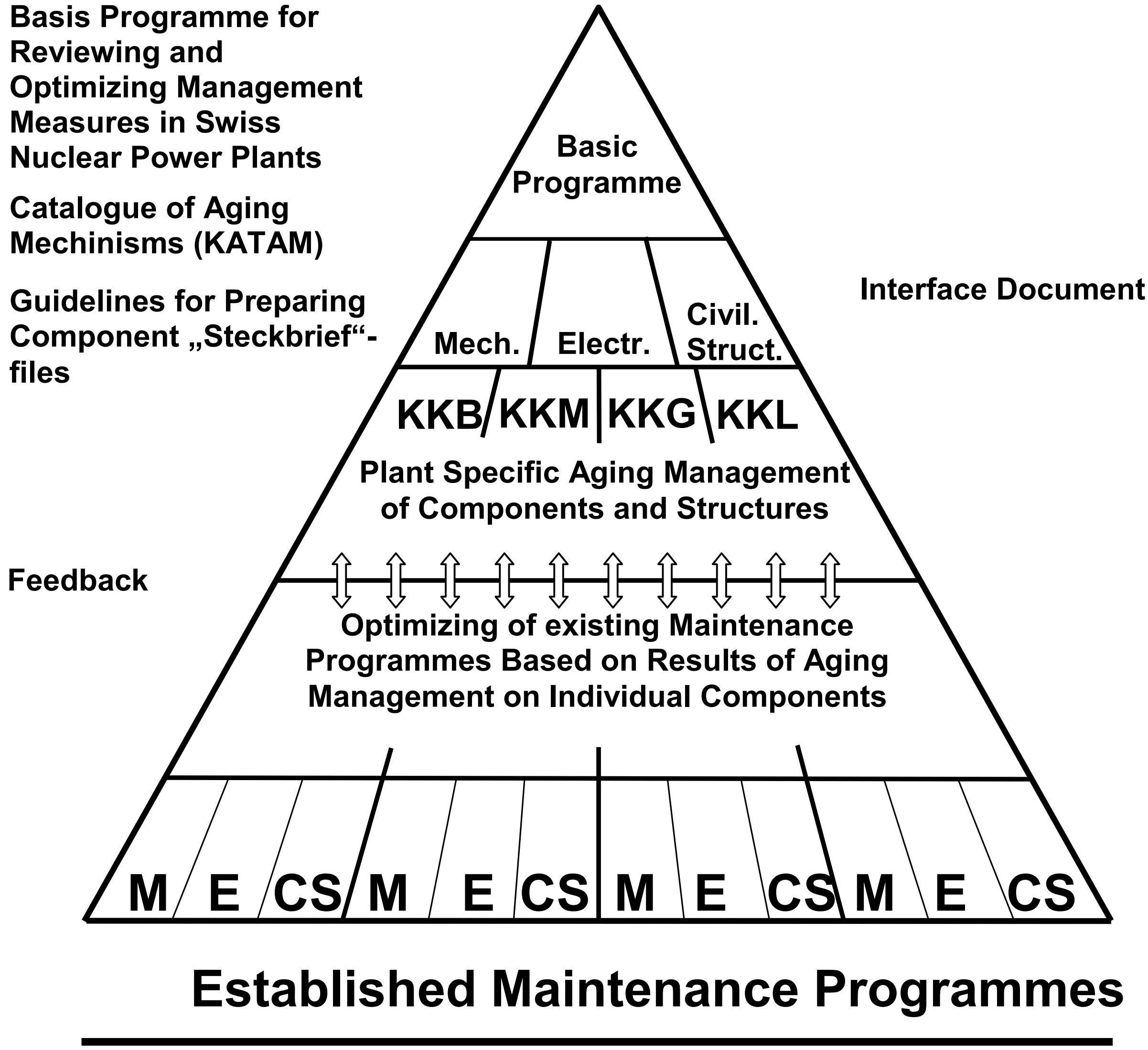


FIG. 1 Basic Structure of Aging Management in Swiss NPP

During the course of this work, utilities concerns on securing future electricity supplies for the Swiss market encouraged closer attention to aspects such as component life which has a direct effect on the life of the whole plant.

2.1. Status of the Work in the Group for Electrical Components

Due to the wide range of electrical components encountered, electrical components have been broken down into component groups such as cables, transmitters, motors, switches, contractors etc. all together 31 groups. This fact determined the structure of the „Steckbrief“-files.

Part 1: Aging mechanisms applicable to the component group

Part 2: Potential diagnostic methods

Part 3: Plant specific review of the components used in a component group taking into account the experience from the maintenance programme

The component reviews performed till now had shown that a greater part of aging management performed is actually an integral part of preventive maintenance measures. The current status of the „Steckbrief“-files is shown in Attachment 1 (Component Groups List Electrical Components).

The group is currently focussing on three items:

- **Maintenance rules:**
  Updating of the current rules as a result of the aging assessments performed so far.
- **Evaluation of the restlife of cables:**
  Surveillance samples of various cables installed in older plants (Beznau and Mühleberg NPP) placed in special exposed areas in the Containment for PLEX purposes.
- **Aging of polymers:**
  Evaluation of a computer code for the determination of the aging of polymers.

2.2. Status of the Work in the Group for Civil Structures

The inspection programme defines the procedure for systematic monitoring of building structures and takes into account the special features of such structures. It provides for:

- **Baseline or principle inspections on the outside and inside**
  These are performed at the beginning of the programme and are repeated after 10 years at the latest.
- **Intermediate inspections**
  Documented findings from routine control rounds are summarised in a list after a period of 6 years after each principle inspection.
- **Special inspections**
  These are inspections for which special measures are necessary such as use of special instruments. These inspections become necessary:
  - when recommendations from findings during principal inspections require additional measures,
  - when relevant changes of condition or behaviour of the structure, or of its usage factor have occurred or
  - when special inspections are deemed necessary after a certain period of time to check the effectiveness of implemented maintenance measures.

Table 1 shows the four utilities with five plants in Switzerland and some typical experience related to the maintenance of their concrete structures. The completed repair and mitigation measures were taken mostly for preventive reasons.

Table 1. Maintenance and repair experience in NPP concrete structures

Plant / Reactor Type	Startup / Age / Major Upgrades	Maintenance Experience		
		Building / Structure	**Degradation**	**Repair / Mitigation Measures**
Beznau Units I and II 365 & 357 Mwe PWR	1969/71 32/30 y NANO 1991/92	Reactor building - outer concrete shel - fuel pool - decontamination coatings Auxiliary buildings - roof insulation - penetrations fo pipings	- minor cracks due to shrinkage - borated water leakage - minor cracks due to shrinkage - humidity / cracks - splitting off	- no action required - helium leakage test - recoated - repaired - repaired
Mühleberg 355 MWe BWR	1972 29 years SUSAN 1989	Reactor building - roof (dome) - outer concrete wall - drywell structure Ventilation stack	- minor cracks due to shrinkage - sporadic corrosion of reinforcing bars due to insufficient concrete coverage - minor cracks due to shrinkage - cracks, bad compacted concrete	- new insulation - protective coating - crack monitoring - repaired; protective coating
Gösgen 970 Mwe PWR	1979 22 years	Reactor building - dome of outer concrete shell	- minor cracks due to shrinkage	- preventive maintenance: reprofilation and water repellent (hydrophobic) coating
Leibstadt 1145 Mwe BWR	1984 17 years	Reactor building - outer concrete shel - fuel pool - decontamination coatings Auxiliary buildings - roof insulation - penetrations fo pipings	- no degradation dark discolourings due to lichen were examined and have no influence on the quality of the concrete structure	- no action required

The utilities group for the structural systems addresses the maintenance, surveillance, rehabilitation and documentation measures. The technical basis is given by the utilities experience, the aging management technology for non-nuclear structures and the international state-of-the-art, such as summarised by the International Atomic Energy Agency in IAEA-TECDOC-1025.

The GSKL-group worked out a Guide Manual which is controlling the procedures for concrete structures, steel structures, anchorage elements, fire proof closings and all types of interface elements. It serves as the common technical basis for all plants and for their structure specific aging management.

The Guide Manual provides the general procedure for condition assessment and documentation. It supplies as technical attachments a catalogue of degradation mechanisms, the available inspection methods and the criteria for classifying the condition of the investigated structure. Figure 2 shows the structure of the Guide Manual and some representative features.

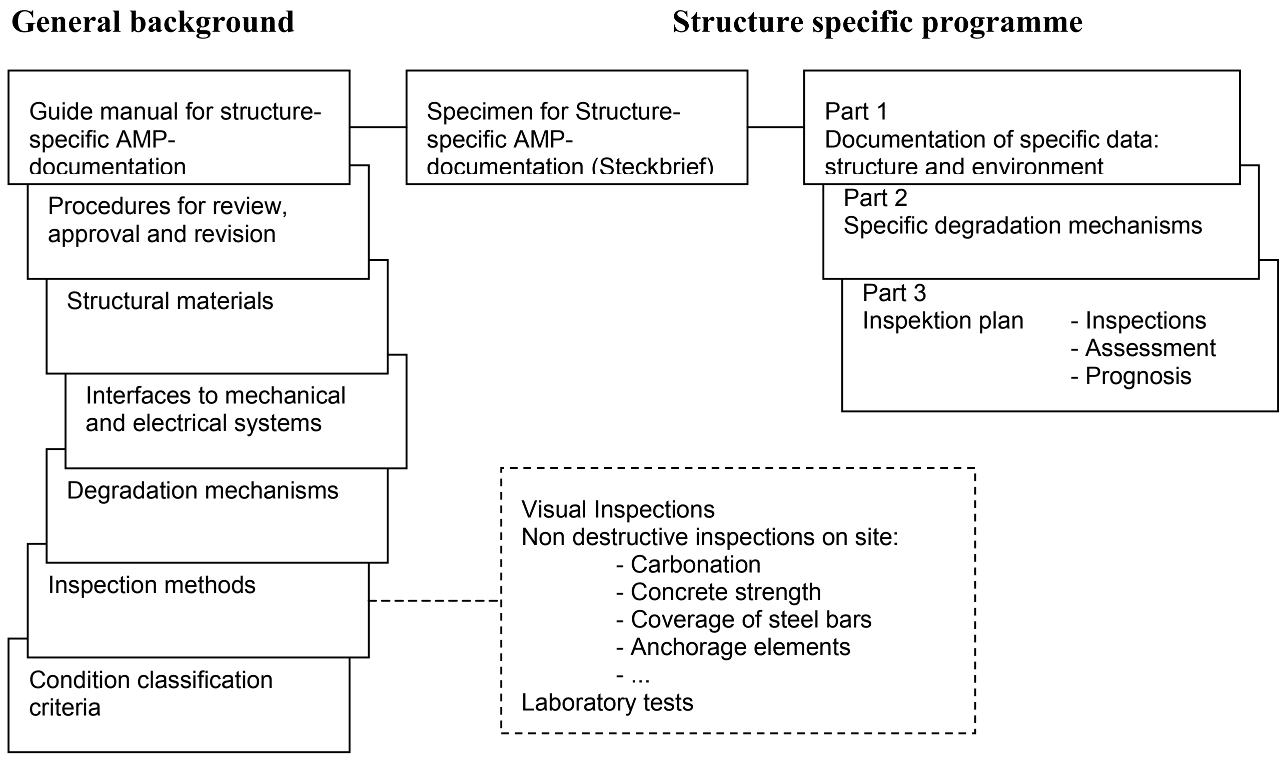


FIG. 2 Guide Manual (Leitfaden) for the aging management of safety related structures

In general the complete visual inspection is the first step to the condition assessment. Further more detailed investigations are required where failures are detected or where the visual assessment is not conclusive.

Special investigations or repair actions are required for cases where the acceptance criteria are not fulfilled. The acceptance criteria for condition assessment was subject of controversial discussions with authority representatives. They are specified case-by-case, considering the safety and aging relevance of the structures. In most cases the acceptable condition of the investigated structure is or will be specified to be either class 2 or 3, within the overall range of 1 to 5, defined by typical damage patterns (type and amount of cracking, corrosion, etc). Figure 3 gives the logic background for this decision. The acceptable condition has to be specified considering the expected degradation curve, the inspection interval and the functional limit of the specific structure.

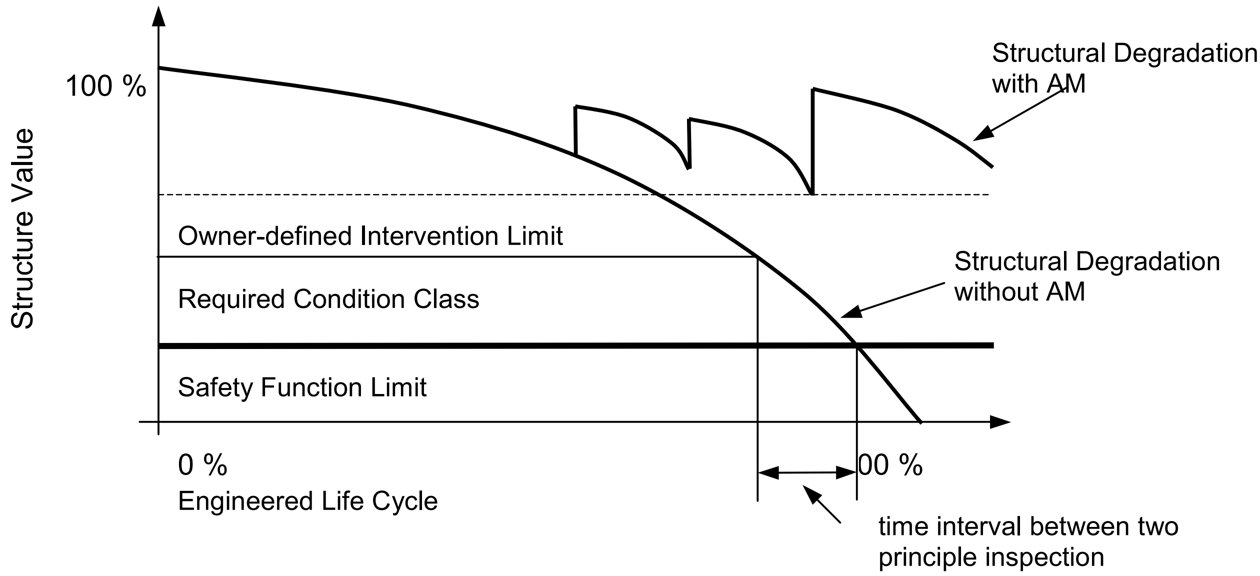


FIG. 3. Specification of acceptable condition class, based on degradation curve

State of Practice

As an immediate measure the utilities are collecting the material fragments becoming available from construction works. The fragments are stored as a source of information on aging related properties.

Basic inspections have been completed and documented for the reactor buildings in all of the five plants. In some cases restoration measures were taken for concrete structures, as indicated in table 1. Table 2 highlights the state of current practice.

Table 2. Overview of the state of practice for structural Condition assessment

Plant	**AMP-documents (Steckbriefe)**	**Inspections for condition assessment**
Beznau Units I and II	9 of 26 planned documents finished	The inspection programme has been started, it is planned to finish the basic inspections by July 2002
Mühleberg	All 22 planned documents finished	The inspection programme has been started, it is planned to finish the basic inspections within 4 years
Gösgen	23 of 24 planned documents finished	The inspection programme has been started
Leibstadt	26 of 46 planned documents finished	The inspection programme will be started next year

The completed inspections and the condition assessments come up with results in the range of expected degradation. So far all detected degradations have been mitigated with commonly applied repair technology.

The current understanding is that the service life of the plants will not be limited by degradation of the concrete structures. The concrete aging is considered to be manageable with a rigorous and systematic inspection and rehabilitation strategy.

2.3. Status of the Work in the Group for Mechanical Components

In the mechanical group the work in the first phase was focussed on the assessments of the systems and components related safety class 1. In the meantime more or less all these systems are finished and sent to the HSK. The status is listed in attachment 2 (Status Report GSKL working group aging management mechanical components).

2.3.1. Example for an assessment, Reactor Pressure Vessel Mühleberg NPP(KKM; figure 4 & 5):

From the point of view of aging management the reactor vessel is quite a complex component. This is because of the large number of parts and materials involved and the varying mechanical and thermal loadings and radiological exposures encountered in service. In addition local effects of service and other transients have to be taken also into consideration. For this reason each part was identified and evaluated separately for possible occurrence of anyone of the aging mechanisms listed and defined in the KATAM.

Furthermore the component service history and the evaluation of occurrences in this and in other similar plants was documented and analysed. All these data accumulated were documented systematically in a special component aging report. Of all the aging mechanisms considered only five of these were determined as relevant for the vessel and for subsequent evaluation in greater detail, these were:

- intercrystalline corrosion cracking
- stress corrosion cracking
- fatigue corrosion
- fatigue caused by service transients
- neutron embrittlement

One of the important conclusions drawn from the study was that the reactor vessel is in a condition to perform its safety function over a greater period of time than that of the 40 years determined by analytical design methods. This fact means that there is nothing against formally extending the life of the vessel.

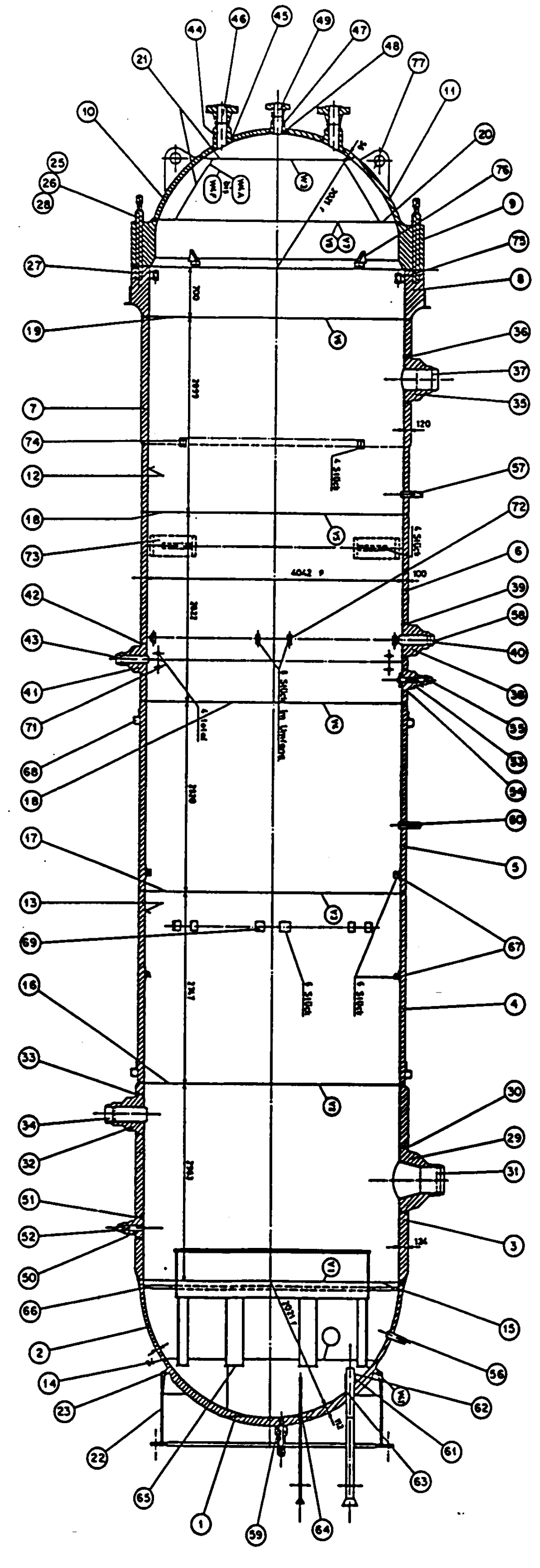
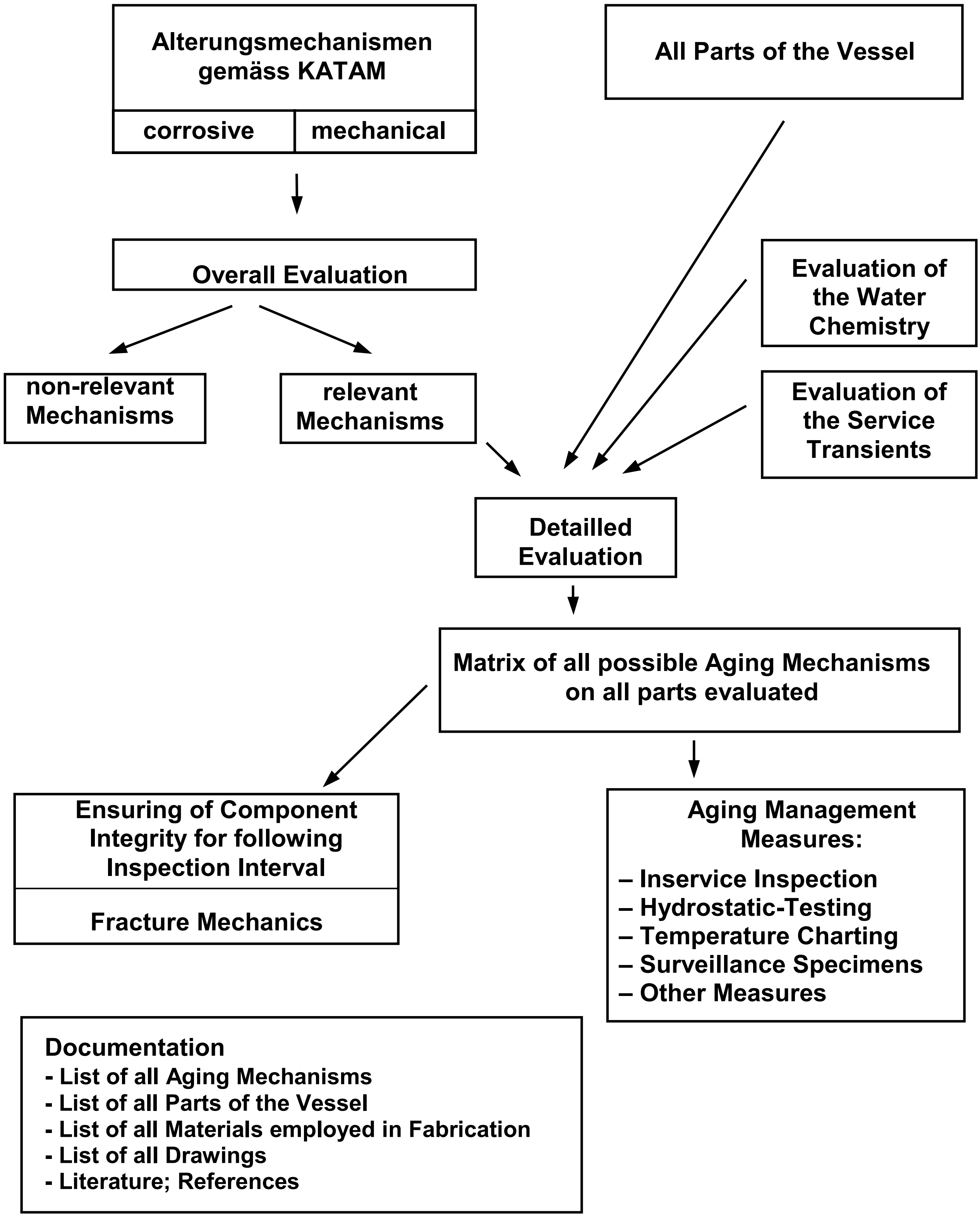


FIG. 4 Rector Pressure Vessel Mühleberg NPP

FIG. 5 Flowchart for the review of the KKM RPV



After finishing the works on components of SC1 systems we started to discuss, how we could manage the assessments of SC2 & 3 components. It was a reasonable amount of work we spent so far for SC1 and it is our goal to reduce this work. The problem was that we have much more components related SC2 & 3 than SC1. So we need to find some reductions and we want to do this in two ways:

(1) Prioritisation of the Components

(2) Generic „Steckbrief"-file on the basis of materials used under our plant-specific operational conditions

2.3.2 Prioritisation:

For a reduction we have to find a valid selection method, which will be approved by our authority, this means, we must:

- identify all safety significant components
- get approval by the authority
- get considerable reduction of manpower
- use existing data's
- easy to use

The way we want to make it is as follows and we will:

- use PSA data's to identify the most safety significant components
- get input from system engineers
- get input from radiation specialist
- get input from operation engineers
- look at the main function of the system

1 Step:

We have to make a ranking with the method of PSA and we defined the following criteria:

FV > 1 E-3 &
FV < 1 E-3 & RAW > 2 component included in the AMP

FV < 1 E-3 & RAW < 2 component not included in the AMP

(FV = „Fussel Vesely importance" and stands for the contribution of a component for the core damage; RAW = „Risk Achievement Worth" and stands for the factor the risk of core damage would increase under the assumption, this component would definitely fail compared to the normal failure rate)

2 Step:

We get the input from the several engineers.

3 Step:

After we have got the ranking and the input from the other disciplines (system engineer identifies his most important components from his point of view,...) we mark the identified components (this are all “active” components) in the P & ID. Now we need the „passive“ components, the pipings and we look at the main function of the system. This means, an emergency system has the function to bring water from a tank to the reactor pressure vessel and we take all the pipings into the aging management programme, that are necessary for identified „active“ components to fulfil their function. This pipings are the mainlines. Additionally all components in this mainlines will also be put into the AMP. Offgoing lines from the mainlines less 1/10 of the mainlines-diameter are not included, bigger lines are included to and with the first isolation valve. Finally all components identified by PSA, system engineer, radiation specialist, operation engineer plus the ones necessary to fulfil the function of the system are included in the AMP.

With our pilot-study of the Leibstadt HPCS we got a reduction of about 80 percent from totally 252 components we came down to 43 after the assessment.

2.3.3 Generic Documents

- We identify all materials used for components (Pumps, valves, pipes, heat-exchangers,..) in our plants.
- We identify all operational conditions (water chemistry, temperature, pressure,..)

This will be limited numbers for the materials and then operational conditions as well. Then we make assessments for all these materials under all operational conditions and report the results in the generic „Steckbrief“-file for materials. For all these identified aging mechanism we will make recommendations for the necessary maintenance and inspection works to prevent failures.

With this document as basis we make the specific review of the identified components and we look for components, that use materials with identified aging mechanisms under our conditions. Then we check our specific maintenance programme for this component and compare it to the recommendations. Depending on the findings we have to set corrective actions on our maintenance programme or not.

The “prioritisation method” as well as the “generic documents” are quite simple ideas, we are currently in the phase of finishing the pilot-study and beginning with the generic document. We cannot say, this is the 100 percent final versions, we know there is space for improvements, which we plan to make when we go on with the practical work of the assessments. Both methods can only work, because we can rely on a very good basis. This means excellent maintenance and inspection programmes, excellent performance, consequent efforts to keep the plant in good technical shape (e.g. backfitting of additional emergency systems in the older plants)

As an additional information all the plants work on procedures and regulations how to implement the aging management in the plant organisation. This is necessary, because till now we spent the most effort on the assessments of the components. But it is also important to implement the aging management programme as a living programme, this means we have to install automatism’s for the interdependence of the AMP and the maintenance.

3. PARTICIPATION IN IAEA-PROGRAMME SAFETY SERIES „ASSESSMENT AND MANAGEMENT OF AGING OF MAJOR NUCLEAR POWER PLANT COMPONENTS LMPORTANT TO SAFETY“ WORK

The Swiss utilities working group also participates in the above mentioned programme. This provides us with the opportunity for a wide, high level international exchange of technical information. Emphasis has been put on the established working groups for the following components:

- PWR / BWR Reactor Pressure Vessel
- BWR Reactor lnternals
- BWR Metal Containments
- PWR Primary Circuit
- Steam Generators

Parallel are two specialist groups involved in Aging Management for Concrete Containments and for Review, Upgrading and Maintaining Electrical Equipment Qualification.

4. CONCLUSIONS

- The Swiss AMP has to be seen as a PLIM
- It is intended to operate the plants beyond their design life of 40 years
- A successful PSR is the basis for that
- Actually results show the good conditions of the Swiss NPP’s
- Economic considerations support our PLIM programme

PLANT LIFE MANAGEMENT IN RUSSIAN NPPS

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Abstract

Taking into account the fact that in Russia there are significant number of NPP units beyond half of design service life, a necessity in an integrated approach which could provide flexible solutions both operational life extension problems and the decommissioning based on unified engineering and informational technology. Such an integrated approach to solution of these problems is referred in worldwide practice as “Lifetime Management”. “Strategy of nuclear industry development in the first half of the 21st century” approved by the Russian Government on February 25, 2000 foresees ensuring of the lifetime extension of the operating power units upon expire of their designed lifetime. The lifetime extension of the first generation NPP units is of general industry character. It is necessary to perform a set of activities on modernization and lifetime extension for 10 NPP units. The goal of the paper is to present on the concept of PliM and PLEX Program applicable to Russian NPP, also to present the information on realization of PliM and PLEX Program on Russian NPP.

1. **In accordance with the «General Directions on Power Engineering Policy of the Russian Federation up to the year 2010»** the total electricity production should include a share being generated, in appropriate economic scale, by nuclear power plants. One of the main goals in this view is to ensure continuation of reliable and safe operation of existing nuclear power plants (NPPs), and actuality of this goal becomes more important with regard to growing crisis of investments into construction of new power units as well as into completion of the power units being under construction.

International experience of nuclear power engineering development shows that for any nuclear facility the tasks of optimization of performance indicators and of ensuring its acceptable safety level could be solved in the most effective manner on the basis of not only review of the facility’s current status but also taking into account influence of previous and upcoming phases of the facility’s life cycle on this status.

Insufficient financing does not allow to perform completion of construction and commissioning new NPP units timely. Taking into account the fact that in Russia there are significant number of NPP units beyond half of design service life, a necessity in an integrated approach which could provide flexible solutions both operational life extension problems and the decommissioning based on unified engineering and informational technology.

Such an integrated approach to solution of these problems is referred in worldwide practice as “Lifetime Management”.

In accordance with the plant life management (PLIM) concept, possible options of the problem solution include the following two: the decommissioning, and the lifetime extension with due regard to ensuring necessary level of safety for population, personnel and the environment.

NPP life management is a complex of measures aimed to promote realization of the operation goal while acceptable safety level is ensured.

The goal function of PLIM is to optimize the "cost-safety" ratio. The way of using the means and resources oriented to achieving the PLIM goal function up to the decommissioning phase determines the strategy, whereas the description of the strategy implementation in chronological manner gives us the working programme.

The main principles of PLIM activities are the following:

- The principle of ensuring safety on the basis of safety-in-depth and safety culture;
- Socioeconomic principle;

The principle of social acceptance of PLIM for an operated NPP power unit includes two alternatives: the continuation of operation and the decommissioning. The last alternative should be followed by replacing power units being put into operation.

2. **"Strategy of nuclear industry development in the first half of the 21-st century"** approved by the Russian Government on February 25, 2000 foresees ensuring of the lifetime extension of the operating power units upon expire of their designed lifetime. The lifetime extension of the first generation NPP units is of general industry character. It is necessary to perform a set of activities on modernization and lifetime extension for 10 NPP units of 4345 MW total rated power.

3. **The NPP lifetime extension activities** are performed in accordance with the requirements of Russian valid legislation and federal norms and rules in the area of the atomic energy use first of all including:

3.1. Federal law "About use of atomic energy";

Article 9 authorizes the Government to make the decisions in the area of the atomic energy use regarding design, construction, operation and decommissioning of the nuclear installation owned by federal state.

3.2. Federal norms in the area of the atomic energy use:

- "General provisions of ensuring NPP safety" (OPB-88/97; NP-001-97; PNAE G-1-011-97);
- "Rules for configuration and safe operation of NPP equipment and pipelines" (PNAE-G-7-008-89)

allow principle possibility of lifetime extension of the NPP equipment and NPP in the whole.

3.3. State standard "NPP and NPP equipment reliability" determines the NPP established lifetime as a calendar period of the NPP operation pointed out in the design upon expire of which the further NPP operation could be extended with a decision made on the basis of its safety and economical efficiency investigation.

To develop the existing normative and methodological basis concerning the issues of the NPP lifetime extension in 1999-2001 there were developed and taken into force the federal documents containing technical requirements in the area of the works performance and preparation of the NPP units for lifetime extension and criteria to estimate its successfulness, namely:

- Federal norms “Main requirements for lifetime extension of the NPP unit”, NP-017-2000;
- Regulation “Requirements to the content and composition of the documents justifying safety for the period of the NPP extended life”, RD-04-31-2001.

In addition the utility (“Rosenergoatom”) to develop the federal norms and rules developed and took into force in the established order a set of guidelines and methodological documents including:

- “Guidelines for organization and performance of the NPP equipment modernization”;
- “General program for comprehensive NPP unit investigation to extend the lifetime”;
- “Guidelines for management of the life features of the NPP unit elements”, RD EO 0281-01;
- “Quality assurance program for design, construction and operation activities at the first generation NPP units”, RD EO 0283-01;
- “General guidelines for the first generation units life extension”, RD EO 0291-01;
- “General guidelines for the second generation units life extension”, RD EO 0327-01;
- “Methodology of the equipment residual lifetime justification”.

4. According to the Russian valid normative documents the activities on lifetime extension of the NPP unit shall include:

- Comprehensive NPP unit investigation;
- Evaluation of the technical possibilities for the lifetime extension of the NPP unit elements;
- NPP unit safety assessment;
- Assessment of the economical reasonability for the lifetime extension of the NPP unit under consideration.

The result of the completed activities is the decision to be made regarding reasonability for the lifetime extension of the NPP unit considered.

The further activities are devoted to the development of the program for NPP unit preparation to the additional operation period.

It includes:

- Justification for the lifetime extension of the irreplaceable elements;
- Implementation of the comprehensive program for the NPP unit modernization;
- Testing of the NPP systems and elements;
- Justification of the NPP unit safety;
- Independent expertise of performed activities by Gosatomnadzor of Russia.

The results of these activities and submitted by the utility to the regulator for independent expertise and licensing for the NPP operation during the additional period.

When performing the NPP unit lifetime extension activities it is necessary to take into account the following:

- Determination of the necessary scope of the modernization activities for the unit to be performed for the lifetime extension is executed on the following results:
  - Deterministic analysis of the NPP unit design compliance with the valid normative documents in the area of safety including identification of the discrepancies and their impact to ensuring of the in-depth protection and development of the corrective measures on elimination of the main safety issues;
  - Probabilistic safety assessment including identification of the total core damage frequency and its value upon completion of the modernization activities.
- The main purposes of the unit comprehensive investigation are as follows:
  - Obtaining and analysis of the information regarding real status of the unit elements;
  - Preliminary residual lifetime assessment of the unit elements;
  - Identification of the unit elements including "critical" with residual life remained and planned for further operation;
  - Identification of the unit elements to be replaced due to the lifetime expired.
- The following is planned on the basis of the unit comprehensive investigation results:
  - Replacement of the expired elements;
  - Maintenance and repair of the elements, which lifetime restoration is necessary to meet the normative requirements;
  - Justification of the residual lifetime of the "critical" (irreplaceable) elements, i.e. operating in the most severe conditions and endured to the maximum impact of the operating factors.

5. **As an example of implementation of the life extension programme one could consider activities performed at the unit 3 of Novovoronezh NPP.** These life extension activities have been performed on the basis of valid federal norms and rules in the field of use of atomic energy as well as number of guidelines and methodological documents developed by the utility, mainly during the period of implementation of these activities. Such a development of the regulation basis has been carried out taking into account international experience and the IAEA recommendations given for NPPs built under earlier standards.

In 1999-2001, the first time in history of national nuclear power engineering, a comprehensive work programme of ensuring extension of Novovoronezh NPP Unit 3 operation has been accomplished:

- Regulation basis for operational life extension has been established;
- Cost-benefit analysis of the unit operational life extension has been fulfilled;
- Modernization of the unit to improve its safety has been accomplished;
- Comprehensive investigation of the unit has been carried out, and residual lifetime of equipment has been justified;
- The in-depth safety assessment report (IDSR) has been elaborated taking into account all the modifications implemented at the unit;
- Start-up test jobs have been performed, and necessary testing the modernized systems has been carried out;
- Re-training the NVNPP unit 3 personnel has been conducted.

During 09.04.2001-12.11.2001 all the planned activities on implementation of the technical measures at unit 3 were completed including upgrading of the accident mitigation system on the basis of implementation of the vortex-stream condenser.

In addition the reactor control, monitoring and protection system, reliable power supply system, process systems and safety systems were upgraded.

The start-up activities and necessary testing of the upgraded systems and equipment were completed, which confirms their operability within the scope of the design and construction requirements.

Novovoronezh NPP unit 3 personnel were correspondingly trained. The operating documents were revised in accordance with the established order.

In the result of the performed modernization the safety of Novovoronezh NPP unit 3 was significantly enhanced. The modernization results are reflected in the final report "Lifetime extension and safety enhancement as a result of the first generation VVER-440 units modernization (Novovoronezh NPP unit 3).

The core damage frequency upon completion of the modernization did not exceed the value of $3{,}44 \times 10^{-5}$ per reactor per year. Prior to the modernization that value was $1{,}8 \times 10^{-3}$ per reactor per year.

The utility developed and approved the in-depth assessment report in the scope necessary for obtaining Novovoronezh NPP unit 3 long-term operation license in accordance with Recommendations of Russian Regulatory Body (RB G-12-42-97). IDSR reflects the state of Novovoronezh NPP unit 3 equipment and systems state for the time of its preparation to the operation within the additional life period.

IDSR materials were endured to the independent expertise in Russian Regulatory Body.

A number of leading experts in the nuclear industry area were involved to collaborate with Scientific and technical council of Russian Regulatory Body for revision Novovoronezh NPP unit 3 safety upon completion of the modernization with concern to its possible lifetime extension.

December 28, 2001 the utility obtained a long-term license of Russian Regulatory Body # GP-03-101-0734 for operation of Novovoronezh NPP unit 3 beyond the limits of the design lifetime.

6. The implementation of the lifetime extension program is also performed at the first generation of RBMK-1000 reactors.

Based on the results of the completed safety analyses and operational experience the following main activities are implemented to enhance these units safety:

- Upgrading of the control rods, implementation of the fast-acting protection system;
- Upgrading of control and protection system;
- Improvement of the reactor physical features;
- Restoration of the gap between fuel channel and graphite lay-out;
- Modernization of the accident mitigation system;
- Modernization of the primary circuit.

The results of the units upgrading allow compensating the discrepancies from the normative requirements and enlarge the list of the design basis accidents.

The in-depth safety assessment reports are elaborated at all the RBMK power units.

Thus, according to “Agreement on the projects of nuclear safety account in Russia” signed between European Bank for reconstruction and development and “Rosenergoatom” in 2000, the elaboration of IDSR was completed for Kursk NPP unit 1.

Now this report is studied by the international experts.

The intermediate report for the first stage of the international expertise was reviewed by EBRD in SRG group and received a positive assessment.

The elaboration of IDS reports for Kursk NPP unit 2 and Leningrad NPP units 1-2 should be completed in 2002 – first quarter of 2003.

The results of the performed modernization at the RBMK units shall significantly improve their safety.

The expected core damage frequency upon the upgrading completion shall not exceed the value of 2,0-6,2 x 10^{-5} per reactor annually.

7. **Decommissioning of a NPP power unit is the final phase of unit life cycle.** This phase, in accordance with the General provisions of ensuring NPP safety OPB-88/97 is a process of implementation of a set of activities following the fuel removal that is aimed to eliminate further using the unit as a source of power and to ensure protection of personnel and the environment.

The following fundamental options have been accepted:

- Prompt dismantling & demolition (the basic option for Russian NPPs);
- «safe – store» with consequent dismantling & demolition.
- As provided by the NPP decommissioning technology, upon ultimate shutdown of the unit the preparation stage starts. This stage lasts 3-5 years and includes the following activities:
- transfer of the unit to the nuclear safe state (removal of fuel from reactor core and from the site);
- removal of radioactive working environment and operational radwaste from the site;
- standard decontamination of equipment, systems and civil structures of the unit;
- development of the entire document package necessary to obtain in regulatory organization (Gosatomnadzor of Russia) the license on decommissioning activities.

- Then the following three stages are provided to be implemented during the decommissioning phase:
  a) The stage of unit preparation to monitored conservation (5-6 years)
  b) The stage of «safe – store» of the unit (30 – 70 years);
  c) The stage of unit demolition as a radiation object (5-6 years)

Costs of the activities on preparation and decommissioning are covered from a reserve fund specially assigned for financing NPP decommissioning activities and performing R&D studies on justification and improving safety of the Rosenergoatom NPP power units being decommissioned. The charge basis of the fund is actual income (after VAT) from energy sales in a current year.

8. Conclusions

The activities performed in the area of the PLIM and PLEX resulted in the following:

- Justification of the NPP lifetime extension technical and economical reasonability;
- Development and implementation at Novovoronezh NPP unit 3 of the activities aimed at the lifetime extension justification. Similar activities are close to the completion at 10 more NPP units.
- Continuous work on further improvement of the legal, normative, methodological and guidelines documentation in the area PLIM and PLEX of the NPP units.
- All the necessary prerequisites and preconditions required for successful implementation of the PLIM and PLEX programs for the first and second-generation NPP units exist in Russia.

POSTER PRESENTATIONS

TREATMENT OF FUEL CHANNELS MADE OF ZIRCALOY-4 FROM PHWVR POWER PLANT

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Abstract

Atucha I Nuclear Power Plant is a PHWVR (pressurized heavy water vessel type reactor), with heavy water as moderator. It was designed and built by KWU - Siemens and has been operating during 29 years. The fuel channels inside the reactor are under a neutron flux and this process produces the activation and deformation of their structures. In Atucha I channels are periodically replaced and the radioactive ones are stored in water pools until a proper decay of activity. As a result, the inventory of radioactive material has increased, but this situation has not been taken into account in the original design of the power station. Consequently, channels must be transferred to other location, requiring a previous treatment for volume reduction. Each channel consist of three different sections: upper and bottom zones made of stainless steel, and a intermediate section where the fuel elements are located, made of Zircaloy-4 (Zry-4). Each section has a different level of activation. Treatment of the radioactive intermediate section requires direct cutting, compacting and burying or alternatively to be chemically processed before final disposal. In this work two different chemical processes, both currently under development, are described. In the first one the Zry-4 is electrochemically dissolved. The resulting solution, containing cobalt and iron as impurities, is extracted with chloroform after formation of the corresponding metal complexes. In the other process Zircaloy-4 is electrochemically dissolved and the cobalt present in the solution is retained on a mercury cathode. In both procedures the resulting zirconium containing aqueous solution is treated with ammonium hydroxide and the metal recovered as zirconium oxo-hydroxide.

1. INTRODUCTION

Atucha I Nuclear Power Plant is a PHWVR (Pressurized Heavy Water Vessel type reactor), that employs heavy water as moderator. It was designed and built by KWU - Siemens (Germany) and has been under operation during 29 years

The coolant channels, located inside the pressure vessel, support the fuel elements, so the fuel channels inside the reactor are under neutron flux. This results in the activation and deformation of their structures. Due to hydrothermal conditions (high temperature and pressure) present in the operation of the reactor it is covered with an estimated oxide thickness of 80 μm.

Periodic replacement of the channels and their storage in water pools until appropriate decaying has determined in a significant increase in the inventory of radioactive material originally not considered in the design of the power station, and resulting in a shortage of storage space for future burnt fuel elements.

Each channel (Figure 1) is of 10,26 m length. It consists of three different sections: upper and bottom sections made of stainless steel, including a 10 cm stripe of stellite (an alloy containing 52% cobalt) and the intermediate section made of zircaloy-4 (Zry-4) where the fuel elements are located. Table 1 shows the composition of the Zry-4 alloy. Each section of the channels reaches a different activation level.

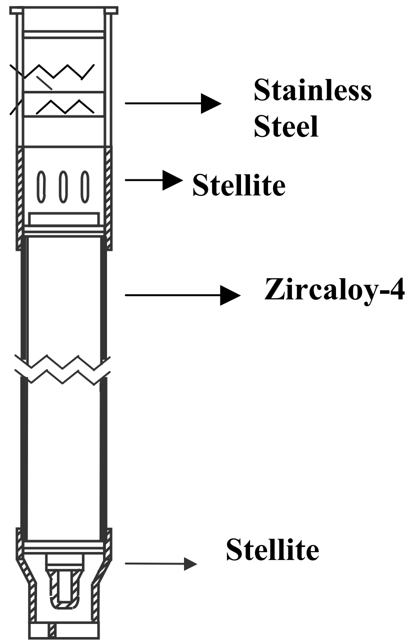


FIG. 1: Schematics of a coolant channel

Table 1: Composition of zircaloy-4

Element	Nominal Composition (g/100g)	Composition of the treated material (g/100g)
Zr	bal.	bal.
Sn	1.2-1.7	1.56
Fe	0.18-0.24	0.21
Cr	0.07-0.13	0.11
Fe+Cr	0.28-0.37	0.32
O	0.05-0.125	0.12
Co	<20 ppm	< 10 ppm
Other comp.	<1446 ppm	< 567 ppm

The activation of the stellite alloy is responsible of the activity of the upper and bottom section. The minor components, as the cobalt present in the Zry-4 alloy, produce most of the activity of the intermediate section.

One of the existing techniques proposed for treatment of the channels stored in the pools consists on cutting the channels in three separate sections: stainless steel, stainless steel plus stellite and zircaloy-4, and processing each one separately using different methods.

After separation of the upper and bottom sections of the channel, the Zircaloy –4 section (contaminated with Co60) must be appropriate disposed.

It can be cut, compacted and buried or chemically processed. Table 2 shows the activities for iron, cobalt, tin and zirconium after 20 years of irradiation and 5 years of decay. Because of the low radiation energy the iron 56 isotope doesn't generate significant doses. Cobalt 60 is the only isotope contributing with significant doses after five years of storage.

We are presently developing two processes of chemical treatment. In the first one the Zry-4 alloy is electrochemically dissolved. The resulting solution, containing radioactive cobalt and iron as minor components is treated with a complexing agent and subsequently extracted with an organic solvent (chloroform). Separation of the zirconium as oxo-hydroxide is achieved by precipitation from the remaining aqueous solution with ammonium hydroxide.

The second process involves the electrochemical dissolution of Zircaloy-4, but the cobalt present in the resulting solution is deposited on a mercury cathode. Separation of the zirconium oxo-hydroxide is also achieved as described above.

Table 2. Activity of zircaloy-4 after irradiation

Element	**g / 100 g**	**Activity of a channel after 20 years of irradiation in the reactor vessel (Ci)**	**Activity of an irradiated channel after 5 years decay (Ci)**
Zr	98	1288	$5.37\ 10^{-2}$
Sn	1.7	120	$10.1\ 10^{-4}$
Fe	0.24	72.1	19.5
Co	0.002	76.8	39.8

2. EXPERIMENTAL

Schematic diagrams of both processes are presented in Figure 2. Electrochemical dissolution of the alloy and recovery of zirconium oxo-hydroxide have the same operation characteristics for both procedures. Separate descriptions of both methods are included in the following sections.

2.1 Dissolution by electrolysis

Since the constituent material of the fuel element clads is the same one as the one used in the coolant channels (Zircaloy-4), samples of these clads were used as a starting point in our experiences.

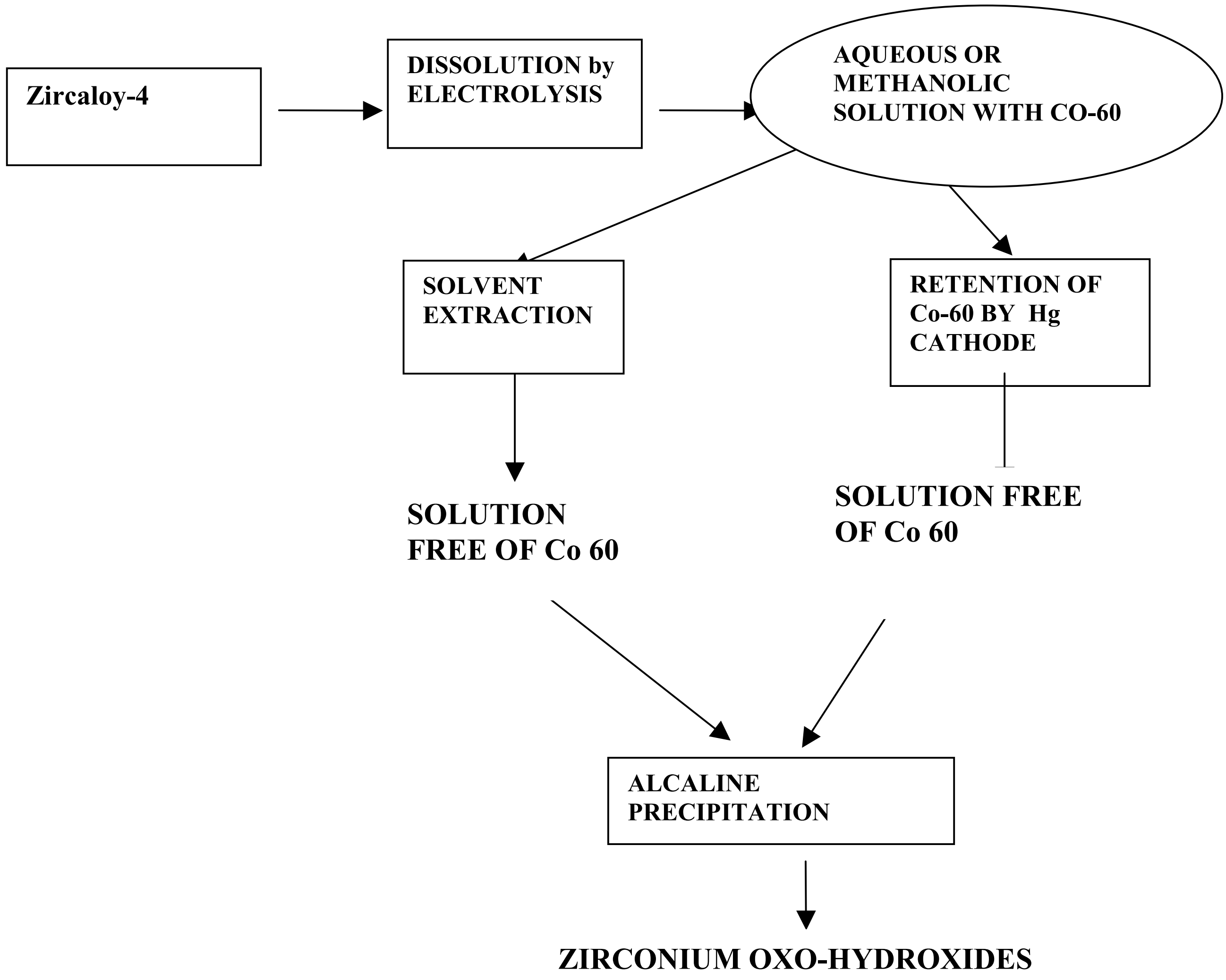


FIG. 2: Chemical treatments

2.1.1 Aqueous electrolytic dissolution

The dissolution of Zircaloy-4 sample (length tube: 5 cm and diameter: 1.3 cm) acting as anode was carried out in 1N hydrochloric acid (HCl). A nickel foil was used as cathode.

The applied voltage was varied on the 5 - 25 V range with the objective of maintaining a constant current of 0.500 Amp during all the experience, which allowed to calculate the process efficiency (80%). The solution acidity must be controlled carefully because zirconium hydroxide precipitates at pH over 1.3. Major components of the resulting solution are zirconyl chloride and iron, tin and cobalt in chloride form. These solutions were used for solvent extraction experiences.

2.1.2 Non aqueous electrolytic dissolution

An interesting alternative studied is based on the possibility of an electrochemical oxidation in non aqueous medium.

According to the literature the zirconium, hafnium and other transition metals can be dissolved anodically in methanolic ammonium chloride solutions [1,2]. In precious or highly passivated metals the discharge reaction of oxygen normally appears in the anode. In this case the literature reports a diversification of the reactions that take place at the anode, being also present the dissolution of Zr with an efficiency of 50% [1]. The reactions and species that are liberated at the cathode were not identified since this topic was not under study.

The dissolution of Zircaloy-4 sample (length tube: 5 cm and diameter: 1.3 cm) acting as anode was carried out in methanolic solution of NH_4Cl. A stainless steel mesh was used as cathode. The applied voltage was varied between 3.5 - 8 V with the objective of maintaining a constant current of 0.250 A during all the experience, which allowed to calculate the process efficiency. It was observed that the sample dissolution was complete during the electrolysis process.

It was found from our experiences that operating at a current density of 20 mA/cm^2 the reaction takes place with efficiency higher than to 90%, accepting that the reaction involves four electrons per mol of Zr.

When working with Zry-4 oxidized in autoclave (90 days, 350 ºC, length tube: 5cm, diameter: 1.3 cm,) the dissolution is accompanied by a detachment of black material, corresponding to 5 % of the sample total mass. Taking into account that the samples from autoclave have an estimated oxide thickness of 1 μm, this value represents a 0.2 % by weight of the sample. It is probable that the detached mass belongs to the insoluble oxide film in this medium and to a part of the alloy material that detaches during the electrolysis.

By attempting to minimize this detached mass, in further experiences the sample was covered with filter paper and the counter electrode (stainless steel mesh) was located closer to the filter paper and surrounding it. Under these conditions that minimize the IR drop and prevents some detachment, the differences between the calculated mass and the measured mass were considerably lower.

A small active sample was taken from a K-17 coolant channel of Atucha I (that had been removed from the reactor 10 years ago due to mechanical failures). This material presented characteristics of long oxidation period under high neutron flux (the channel was in operation during 15 effective full power years). During the sample dissolution this material presented a structure of scales, with a great tendency to detach them from the original material when compared with the oxidized material test in autoclave.

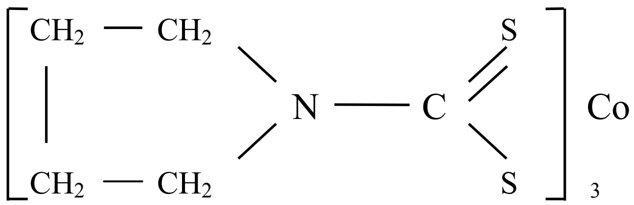
The resulting solution, which was active, was used in the experiences of Co retention process that will be described in the following sections, since Co 60 in such low concentrations is an effective tracer to follow the process. The estimated oxide thickness of the active sample was around 80 μm. If it is compared with the channel thickness the oxide percentage is about 1.3 % by weight.

2.2. Separation by solvent extraction

Solvent extraction is a method based on the transfer of a solute from one liquid phase into another, usually an organic solvent. The method is generally performed in few minutes with quite simple apparatus. Transfer of inorganic components from aqueous solution into the essentially non-miscible organic solvent involves in general the chemical transformation of dissociated or highly polar species into uncharged compounds soluble in the organic phase. In the case of metallic ions, this can be accomplished by chelate formation. The term chelate is referred to complexes formed with ligands having two or more donor atoms which simultaneously complex a single metal atom to form a stable ring geometry. The four more commonly used chelating agents for trace metal extractions are 8-hidroxyquinoline, cupferron, dithizone and dithiocarbamate. The dithiocarbamate ligans have two donor sulphur atoms able to react with metals that form insoluble sulphides in water but are not soluble in organic solvents.

The solvent extraction using dithiocarbamates was applied by Tseng et al. [3], who described a method for the isolation of radioactive cobalt from a large volume of seawater by concentration in to small volume of chloroform. Chai Mei Ju et al. [4] described solvent extraction of dithiocarbamate complexes prior to the determination of trace elements in environmental samples.

The result of the electrochemical dissolution of zircaloy is a solution containing zirconyl chloride and traces of components such as cobalt (0.35 ppm). Zirconium oxo-hydroxide precipitate from this solution at pH over to 1.5. Ammonium pyrrolidindithiocarbamate (APDC) was chosen as complexant due to its ability to form complexes with cobalt at pH about 1.0. The structure of the complex may be represented as follows:



At the working pH the formation of the complex is accompanied by a redox reaction [5], in which the cobalt is oxidized to a trivalent oxidation state

Chloroform was selected as the organic extractant because its solubility in water is low. In addition, being a solvent heavier than water, vaporization into environment is reduced. Also, solubility of the APDC-Co complex in chloroform is high.

Laboratory scale tests were carried out using separation funnels [6]. Synthetic solutions of composition similar to that of the real solutions, and liquid phases arising from the electrochemical dissolution step of non-irradiated Zircaloy were tested. A flow sheet of the method is included in Figure 3. About forty experiments were run under different conditions by varying the operation pH, the organic to aqueous phase volume ratios, the concentration of APDC and the stirring time.

Co-60 radioactive tracer and a multichannel gamma spectrometer provided with a NaI (Cs) detector was used for measurement of the cobalt distribution between the phases. The following conditions were found acceptable for the process:

a) Organic/Aqueous phase ratio: 1: 2
b) pH : 0.9 - 1.2
c) APDC concentration in aqueous phase: 0,5% (m/V)
d) Number of chloroform extraction steps: 2

The recovery of cobalt achieved under these experimental conditions was 99 %.

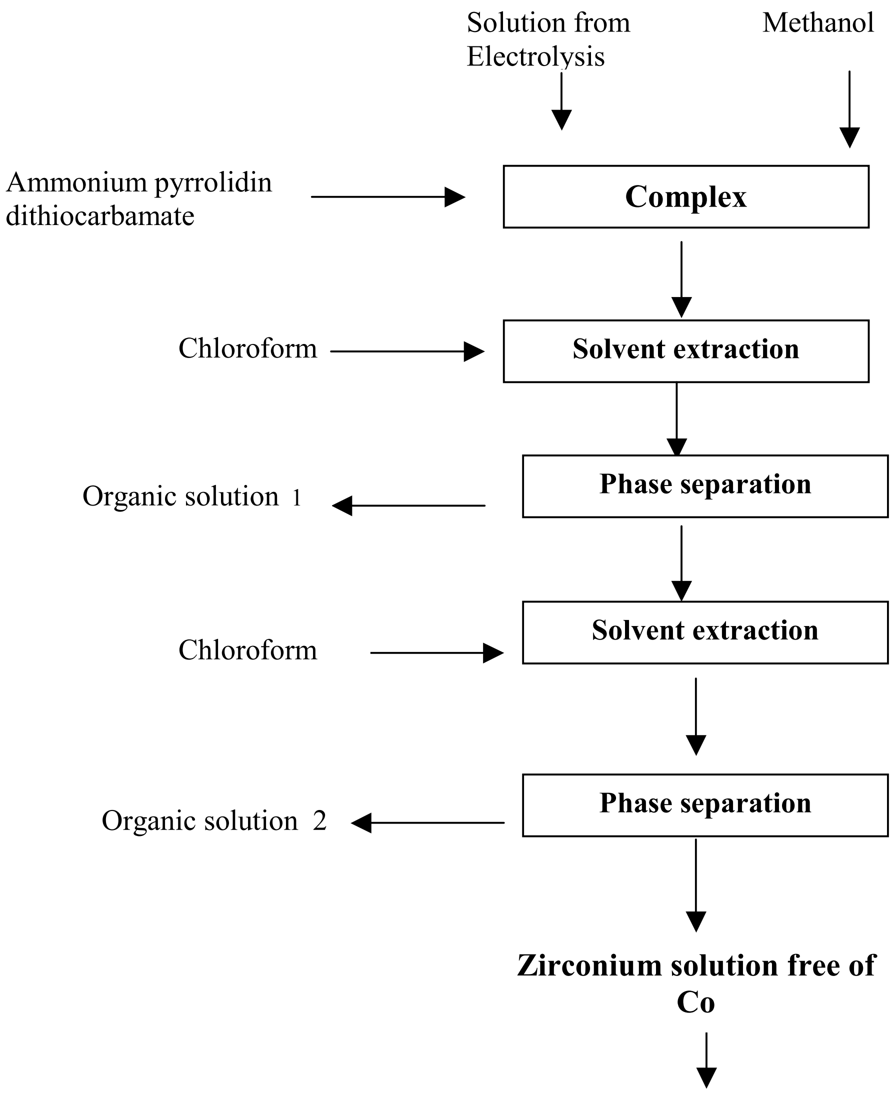


FIG. 3: Solvent extraction procedure

2.3 Electrochemical Stripping of Cobalt

With the objective of retaining the Co present in the Zircaloy-4 dissolution, a mercury film was studied. This film could act as cathode in alcoholic and in aqueous media [7].

Different experiences were carried out. These had in common the preparation of the cathode and anode. A copper sheet that was prior submerged in hot saturated aqueous solution of Hg_2Cl_2 to provide the first superficial mercury layer, was put in contact with metallic mercury several hours. A thin brilliant layer was obtained and it was used as cathode.

In a first experience a bar of Pb was used as anode without any superficial treatment as it is usual in the electroplating techniques (Pb had a concentration of Sn lower than 10%). The copper sheet covered by mercury film with an apparent area of 25 cm^2 was placed in a small glass cell surrounding the Pb bar (diameter: 0.5 cm).

2.3.1 *Analyzed solutions*

Different solutions were studied with the above mentioned cell.

a) An aqueous solution of 0.01M $CoCl_2$ using Na_2SO_4 as supporting electrolyte.
b) Idem a) but adding active Co in such a way that a dose rate of 10 µSv/h was obtained.
c) A saturated solution of NH_4Cl in methanolic solution with the addition of $CoCl_2$ to reach a 0.01 M Co concentration.
d) An aliquot of the resulting solution from the Zry-4 electrochemical dissolution, to which $CoCl_2$ was added until a Co concentration of 0.01 M was reached.
e) An aliquot of the resulting active solution from the Zry-4 electrochemical dissolution.

2.3.2. *Results*

The solution a) showed a dependence on the Co concentration in solution [8] that follows the equation:

$$\ln(\text{Absorbance}) = -0.544 \times T - 0.632 \qquad (1)$$

where:

T is the time (hours),

For solution b) the concentration decrease of the radioactive tracer [8] was measured during the experience. The following equation was obtained:

$$\ln(\text{Co60 A}) = -0.172 \times T + 5.1992 \qquad (2)$$

where:

Co60 A is the Co60Activity (counts / s),
T is the time (hours),

The potential of the mercury cathode vs. SCE (Saturated Calomel Electrode) was -1.35 V.

For the cases c) and d) the electrolytic solution coming from a reservoir was impelled by a peristaltic pump with a flow of 100 ml/min during approximately three hours. The current was kept constant at 100 mA and the applied potential was varied on the 4 - 10 V range. The potential of the mercury cathode vs a saturated calomel electrode was -1.5 V.

The decrease in the concentration of Co became evident from the color change of the solution. In addition, by employing the EDAX technique along with scanning electron microscope it was possible to detect the presence of the Co retained on the cathode surface.

The structure of the Co deposit can be observed in the micrographs of Figures 4 and 5. Figure 4 corresponds to the solution c), which shows Co clusters that have an approximate diameter of 100 microns.

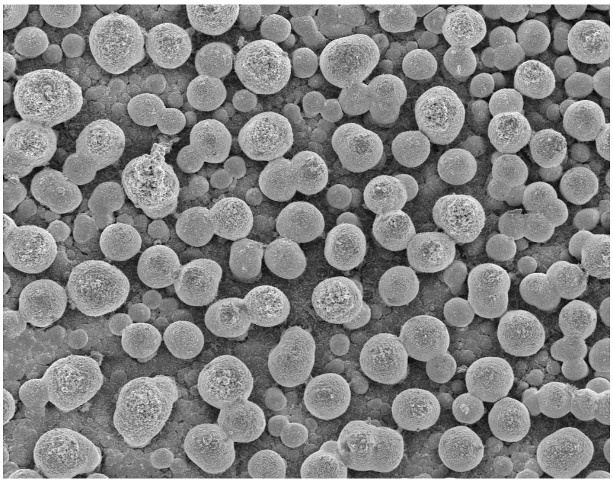


FIG. 4

Figure 5 shows the morphology change obtained on the mercury film when the treated alcoholic solution contained the products of the Zry-4 dissolution.

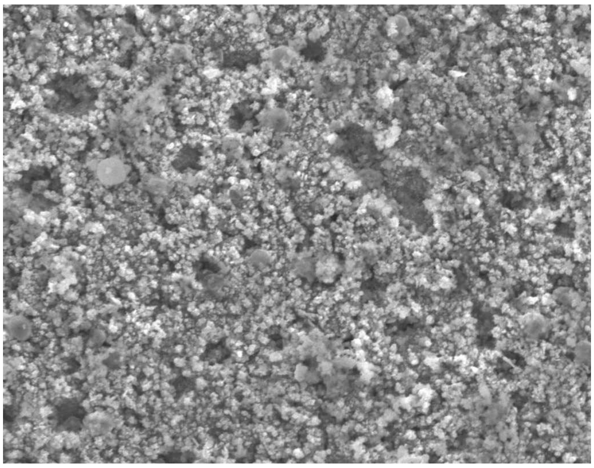


FIG. 5

In both cases the signal for Cl indicated the retention of this element in the mercury film.

The anode of Pb was slightly attacked during the process. This was shown by the Pb signal detection when the mercury cathode was analyzed after the experiments.

A qualitative study was performed with the active solution e). After several hours of recirculating it, it was possible the detection of activity on the cathode surface.

2.4. Zirconium oxides precipitation

Separation of the zirconium oxo-hydroxide is achieved by precipitation with ammonium hydroxide from the remaining aqueous solution. Recovery of zirconium was 95 %.

3. CONCLUSIONS

The possibilities of an electrochemical treatment combined with an extraction process to separate the Co60 was shown to be feasible (Co60 is the main factor of the inventory of Zry-4 activity). The separation stage presents two alternative possibilities:

A) Solvent extraction with high selectivity.

B) Electrochemical retention on a mercury cathode making the process simpler.

Both possibilities are linked to a final zirconium precipitation as an oxo-hydroxide.

The operations involved in these processes performed in plant scale may be of a complexity comparable to the cutting and crashing techniques. There is no doubt that the separation of the 10 ppm of Co representing an order of 250 mg of Co per channel, allows the disposal of the Zr as a low activity metal or directly as a non active metal, simplifying its final disposal treatment.

REFERENCES

[1] ALAJDEM A., Anal. Chem.,41,989 (1969).
[2] ALLEN J. B. Encyclopedia of Electrochemistry of the elements.. Volume IX. Part B, Marcel Dekker,Inc. New York and Basel.(1986)226-248
[3] TSENG C.L., HSIEK Y.S., YANG M.H. "Determination of Radiocobalt in Seawater by Successive Extractions with Reagents of Tris / Pyrrolidine Dithiocarbamate / Bismuth / III and Ammonium Pyrrolidin Dithiocarbamate and Back Extraction with Bismuth/ III", J.Radioanal. Nucl. Chem. Letters 95/6(1985)359-366
[4] CHAI - MEI JU, J. "Solvent Extraction of Dithiocarbamate Complexes for Determination of Trace Elements in Environmental Samples", Thesis, Graduate School, University of Idaho (1985)
[5] HULANICKI A. "Complexation Reactions of Dithicarbamates", Talanta, Vol. 14, (1967) p1371-1392
[6] IGLESIAS, A., LAGAMMA, A., ARDUINO, M. Zircaloy-4 treatment to obtain Zirconium Cobalt free. XXIV Annual Meeting of AATN. Buenos Aires, Argentina (1997).
[7] BARENDRECHT, E. "Stripping Voltammetry". Electroanalytical Chemistry. A Series of Advances.Volume 2, Allen J. Bard.. Marcel Dekker,Inc, New York (1967) 53 –110.
[8] IGLESIAS, A.,JIMENEZ REBAGLIATI, R. Stripping of Co from aqueous solution.
[9] XXV Annual Meeting of AATN.Buenos Aires, Argentina (1998)

IMPROVING THE PEAK POWER DENSITY ESTIMATION FOR THE DNBR TRIP SIGNAL UTILIZING OUT-OF-CORE DETECTOR SIGNALS

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Abstract

This paper aims to show that in small power reactors the power density peak factor can be estimated during reactor operation from the axial power differences measured from out-of-core detector signals, and from signals indicating the control rod positions in the core. The response of out-of-core detector signals were measured in experiments performed in the IPEN/MB-01 zero-power reactor. Several reactor states with different power density distribution were obtained by positioning the control rods in different configurations. The power distribution and its peak factor were calculated with the CITATION code for each of these reactor states. The obtained results show that for the small reactors it is possible to correlate without difficulty the power peak factor and the control rods position. The results also show that correlating the peak factor with the axial power differences obtained from out-of-core detector signals is not a straightforward task. There is a complex pattern between the peak factor and the axial power difference. In order to improve the accuracy, the results indicate that the peak factor should be determined through a correlation, which takes into account the axial power difference, and the quadrant power difference or quadrant tilt.

1. INTRODUCTION

In power reactors the design limits on peak power density require control of the power density distribution to ensure that local conditions in the fuel rods and coolant channels do not challenge the cladding integrity [1]. For instance, a DNB occurrence in the reactor core may cause overheating of the fuel that results in possible cladding perforation with possible release of fission products to the reactor coolant. In establishing the DNB safety limit, one assumes that all transients begin with power density distributions with strong peak factors, which cannot be violated during normal operation, otherwise would produce unacceptable consequences to the reactor fuel.

In PWRs the power peak factor is not directly measurable. To circumvent this problem, what is done is to perform a power distribution map with the movable incore detectors at some time during the core cycle. A computer program processes this power density map to furnish the power peak factor which is considered valid for some period of time. The changes in the core during this period must be taken into consideration through some measurable variables. This is done by multiplying the power peak factor obtained previously by a correction factor based on the axial power difference, quadrant power difference, and indication of control rod insertion. These variables are directly and continuously measured during the reactor operation [1].

The correction factor utilized to improve the power peak factor information during operation is estimated through different correlations according to its objective. For the automatic protection system, the correlation is based on the axial power difference inferred from the signals of out-of-core detectors located in the top and the bottom parts of the neutron flux instrumentation channels. In the other hand, for observing the reactor limiting conditions of operation, which require several actions from the operator following operation procedures, the correlation can be based also on information from the control rods position. In any case, it is necessary to develop a correlation which produces the power peak factor information out of control rod position and axial power difference data which are directly measurable in power reactors.

In small power reactors, the smaller size of the core makes the correlation between the power peak factor and the axial power differences from out-of-core detector signals less accurate [1]. The bottom out-of-core detector can monitor the top of the core, and top out-of-core detector can monitor the bottom part of the core. Their relative difference may not be as close to the actual axial power offset as they are in large power reactors.

In this work an investigation is carried out in order to determine correlations for obtaining the power peak factor out of axial power difference and control rods positions for small power reactors. The axial power difference of out-of-core detector signals are measured in the IPEN/MB-01 reactor, a zero-power nuclear reactor for different power density distributions. Axial power differences are measured for the north and west sides of the core. In section 2 is presented the IPEN/MB-01 reactor and in section 3, the definitions of axial power difference and quadrant power differences from out-of-core detector signals. Section 4 presents how the axial power difference and power peak factor results were obtained in the experiments, and in section 5 are presented the results for the determination of peak factors from control rod position data, and for the determination of the peak factor from the axial power differences data. Finally, in section 6, are presented the conclusions.

2. THE IPEN/MB-01 REACTOR

The experiments were performed in the IPEN/MB-01 reactor, a zero-power light water reactor with fuel rods made of slightly enriched UO_2 pellets cladded with stainless steel[2]. The fuel rods are individually placed in the core according to the desired geometry configuration. The assembly is located inside a stainless steel open tank full of light water which acts as the reactor moderator. The maximum core power level is 100 W corresponding to a thermal neutron flux level around 10^9 n/cm^{2}s.

Figure 1 presents a top view of the core with fuel rods assembled in a way to simulate four PWR fuel elements. The small squares in the figure represent one fuel rod in the assembly. Each one of the four control elements are made of 12 absorber rods connected together similarly to PWRs. Each control element can move independently. The letters “A” and “B” represent the control elements BC1 and BC2 and the letter “S” represents the control elements BS1 and BS2, respectively. The control elements, in spite of being formed by a set of 12 absorber rods are also designated simply by control rods. The shaded rectangle on the core west side represents stainless steel plates which can be placed inside the stainless steel tank in order to simulate the internal structure of a PWR pressure vessel.

The large circle in Figure 1 represents the stainless steel tank, the small circles near the core represent the several neutron detectors available for the reactor control instrumentation and for experiments.

The shaded two circles outside the stainless steel tank represent the experimental out-of-core detector channels located in the north and west sides of the reactor. Each channel is composed of two detectors viewing the top and bottom parts of the core.

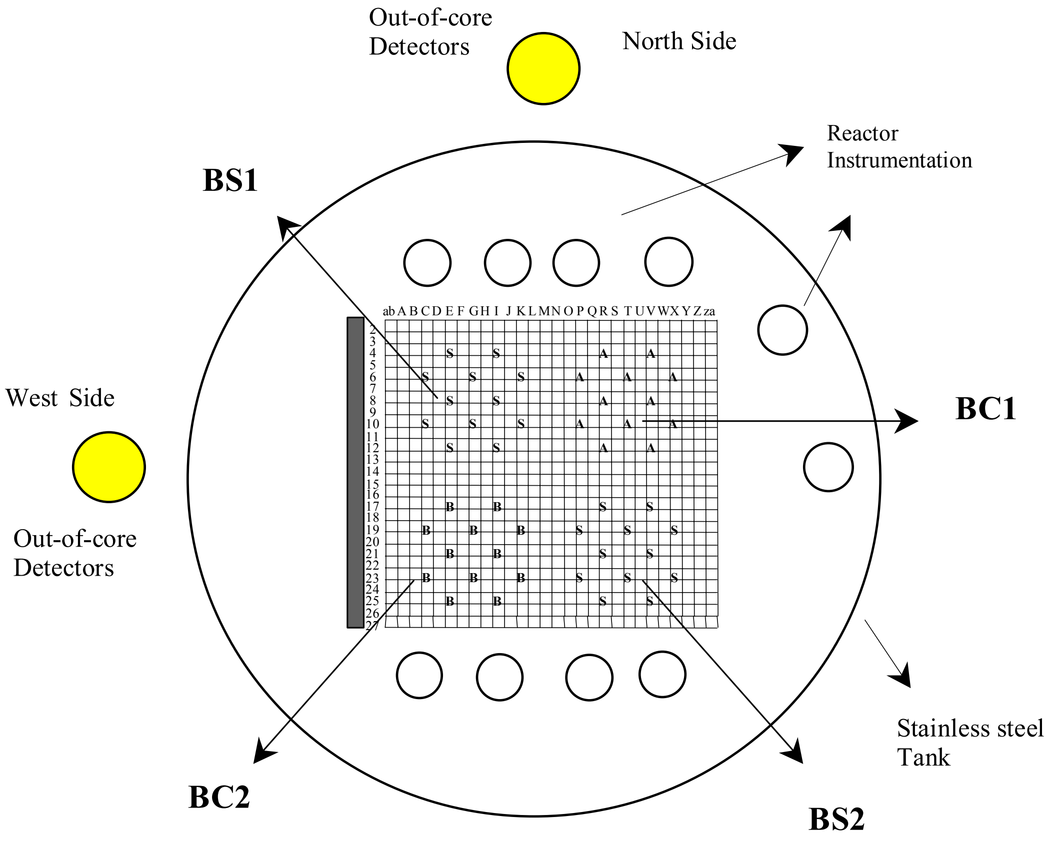

FIG 1. View from the IPEN/MB-01 zero-power reactor

3. AXIAL POWER AND QUADRANT POWER DIFFERENCES MEASURED FROM OUT-OF-CORE DETECTORS

The out-of-core detectors from power reactors are responsible for monitoring the core power level for the control system, and also to furnish the core axial power difference, APD, and the core quadrant power difference, QPD. The APD is obtained for the same channel, north or west, as the difference between the signals from the detectors viewing the bottom and the top of the core, divided by the sum of both signals, that is,

$$APD^{N} = \frac{S_b^N - S_t^N}{S_b^N + S_t^N}. \tag{1}$$

where APD^N is axial power difference from the north side, S_b^N and S_t^N denote the detector signals viewing the bottom and the top of the core from the north side. A similar definition can be made for the APD^W, the core west side APD. APD^N and APD^W are used to infer how skewed in the axial direction the power density distribution is.

In this work the QPD is not utilized but its definition is presented for completeness. The QPD is obtained for the bottom and top parts of the core as the difference between the detector signals from the north and west sides divided by the sum of them,

$$QPD_b = \frac{S_b^N - S_b^W}{S_b^N + S_b^W}. \tag{2}$$

This definition would be more properly called power octant difference since in principle those out-of-core detector signals view the bottom quarter of the core or its bottom octant. The quadrant term is used here because it follows more closely the quadrant tilt terminology used for power reactors. The definition for the QPD for the top half of the core is the same of Eq. 2 changing the subscripts from “b” to “t”.

4. THE ESTIMATION OF THE APD AND POWER PEAK FACTOR RESULTS IN THE EXPERIMENTS

The experiments were devised to determine a way for inferring the power density peak factor out of APD^W, APD^N, QPD_t and QPD_b results obtained from a power reactor out-of-core instrumentation. This work presents only the APD results obtained from the signals of the detectors positioned outside the stainless steel tank, resembling out-of-core power channels. ^{10}B neutron detectors were utilized in the out-of-core channels because the neutron flux level outside of the tank was about 10^3 n/cm^{2}s.

The experiments took into consideration the fact that the control rods usually determine the shape of the power distribution in nuclear reactors [3,4]. The different power density distributions in the IPEN/MB-01 reactor core were obtained by different control rods positions which defined a core state. For each core state the out-of-core detector signals were measured, and the APDs were obtained through Eq. 1. The power peak factors were obtained from three-dimensional core calculations which produced the k_{eff} and power density distribution [5]. At the end, for each core state there were the four measured out-of-core detector signals, the APDs, the calculated power peak factor and its location.

The power density distribution inside of the core could also be measured by foil activation, wire activation or through a miniature fission chambers but it would be almost impossible to determine with good accuracy the power peak factor. Therefore, it was decided that the information about power peak factors would be calculated, which could be easily done for the IPEN/MB-01 reactor with an accuracy better than 4 %. In sections 4.1 and 4.2 are presented the details regarding the measurements and calculations.

4.1. Measurement of the APD in the IPEN/MB-01

The core states with different power distributions were obtained by establishing critical states with different control rods positions. In order to increase the number of different states a maximum excess reactivity was introduced in the core [3,4]. The sensibility of the four detectors was obtained by locating them outside the stainless steel tank at the north side bottom position, and taking their counts at power levels ranging from 1 to 100 W. After that, the four detectors were located in the top and bottom positions at the north and west sides for simulating a typical out-of-core instrumentation of power reactors.

The first state was the one with the four control rods placed at the height which was the 67,1 % withdrawal position at 10 W power level. Then it was taken the counts from the four detectors positioned outside the stainless steel tank which produced the APDs data. After that, similar measurements were repeated with the control rods in different configurations in order to establish different power density distributions in the core.

The control rods movements were performed in a way that the reactivity introduced in the core by the withdrawal of one or more control rods would be compensated by the insertion of other control rods in order to maintain criticality. Starting from the first state with all control rods positioned at 67,1 %, one or two control rods were withdrawn to the 77,1 %, 87,1 % and 100 % positions. In each case, the following procedure was taken to maintain criticality:

- Insertion of a diagonal control rod;
- Insertion of a parallel control rod;
- Insertion of two control rods together.

Altogether, 56 different critical states were obtained furnishing for each one the following data: the position for the control rods, and the signals from the four out-of-core detectors. From these detector signals the APD^N and APD^W were obtained for each one of the 56 states.

4.2. Calculation of the power peak factors

The power density distribution for each of the 56 states was obtained through a three-dimensional, four energy groups, pin-by-pin core calculation using the CITATION code [6], which solves the three-dimensional multigroup neutron diffusion equation. The cross-sections were generated with a modified version of the unit cell HAMMER/TECHNION code [7]. The three-dimensional core model could represent the control rods positions with detail, and, therefore, yielded very accurate power density distributions with uncertainties below than 4 %. The power peak factor and its location for each of the 56 states were obtained from the power density distribution results produced by the CITATION code.

5. RESULTS OF POWER PEAK FACTORS AND AXIAL POWER DIFFERENCES

With the out-of-core signals, control rod positions, and power density distributions a data set of 56 control rods configurations correlated to the APD, and power peak factors could be constructed. From this data set, a correlation between the APD with the peak factors could also be obtained. Figures 2 and 3 show the APD^N and APD^W, and the corresponding peak factors, as a function of the BS1 control rod position; Figures 4 and 5 show the behaviour of the same parameters as a function of the BS2 control rod position. The several states, with different control rods positions, produced power peak factors ranging between 2.08 and 2.27.

It is seen that there is a pattern in all figures that can be described as follows: when the BS1 control rod is positioned around 60 % one finds the greatest values for the axial power difference, about 18 %. As the control rod is moved inside the reactor up to the 50 % position the APD^N and APD^W decrease rapidly. As the control rod is moved out of the core, both APD^N and APD^W decrease but more slowly. The dispersion of data points observed in the figures is because one can obtain the same APD for different control rods configurations.

The power peak factor shows a similar behaviour but with a flatter maximum from the 60 % to the 90 % positions. This means that the peak factor is the same for many different control rods configurations, changing its location but not its magnitude.

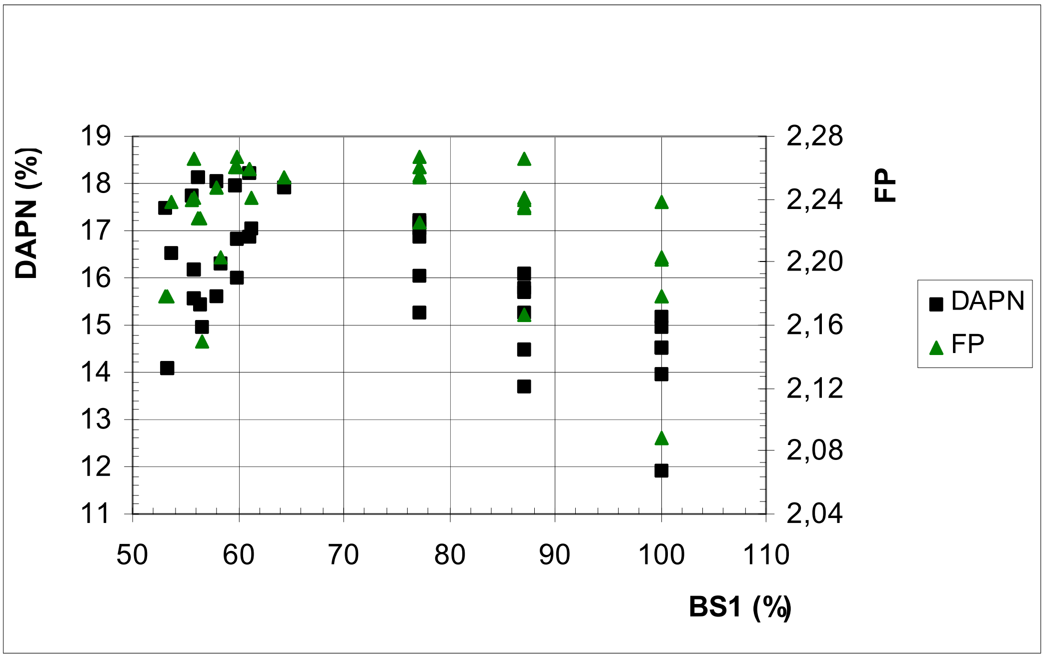

FIG. 2 The axial power difference from the north side (DAPN) and the peak factor (FP) as a function of the BS1 control rod position

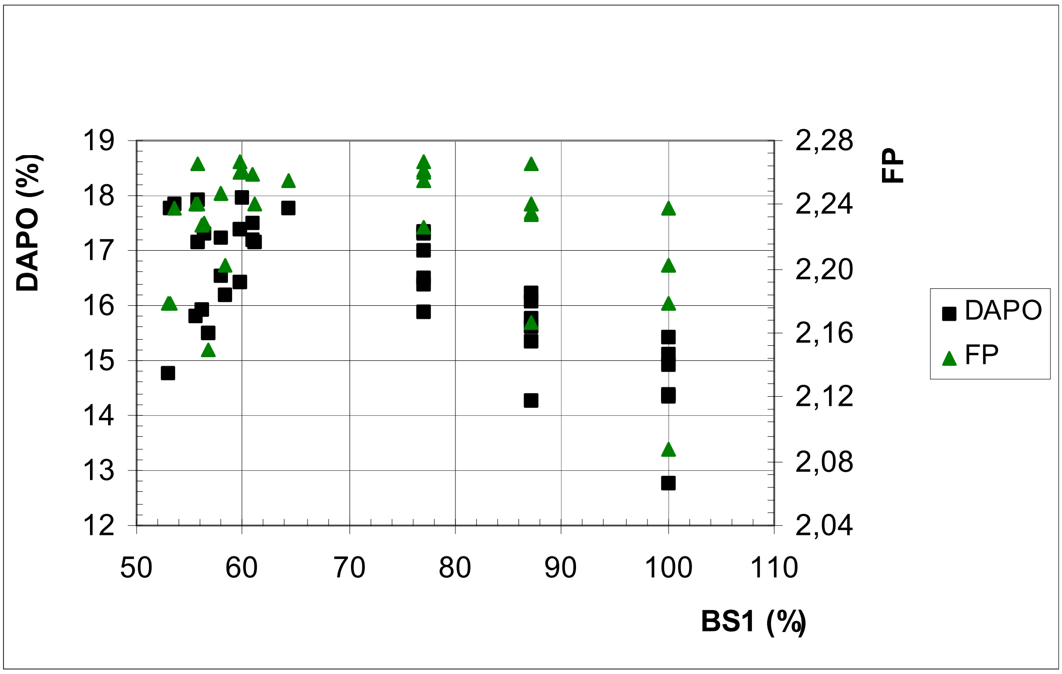

FIG. 3 The axial power difference from the west side (DAPO) and the peak factor (FP) as a function of the BS1 control rod position

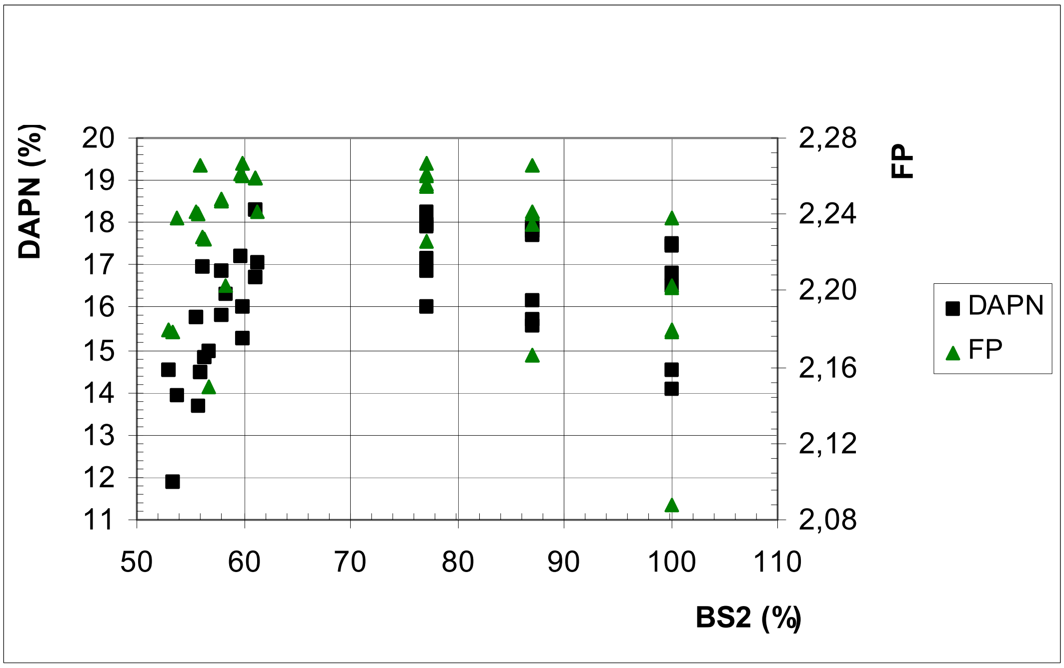

FIG. 4 The axial power difference from the north side (DAPN) and the peak factor (FP) as a function of the BS2 control rod position

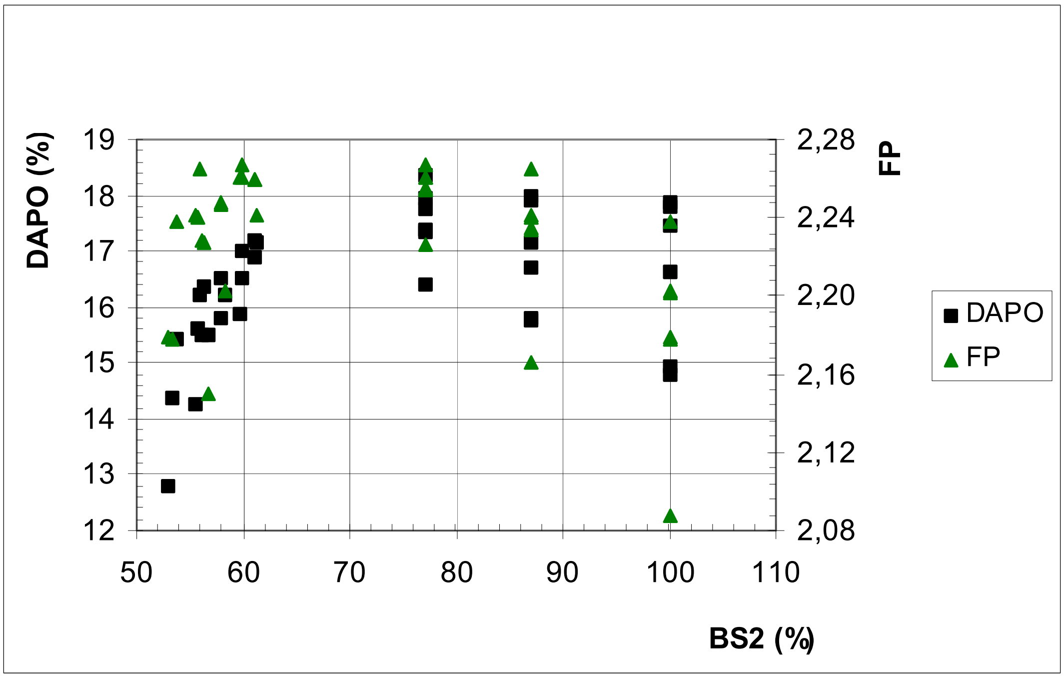

FIG. 5 The axial power difference from the west side (DAPO) and the peak factor (FP) as a function of the BS2 control rod position

The behaviour shown in Figures 2 to 5 is due to the fact that as a control rod, which is initially completely withdrawn, moves inside the core it pushes the neutron flux or the power density to the bottom of the core, causing a non-symmetric distribution with high APD and peak factors. After some point, the neutron flux moves to other quadrants causing a decrease in the peak factor and in the APD results. As the rod is inserted even more, the neutron flux tends to return to the top of the core to attain a more symmetric distribution. For the IPEN/MB-01 reactor core configuration, one verifies that as the BS1 control rod passes about the 60 % position there is the transition from pushing the neutron flux to the bottom part of the core to pushing it to the other quadrants. Similar results were obtained for the behaviour of APD and peak factors regarding the BC1, BC2 and BS2 control rods.

5.1. Determination of peak factors from control rod position data

Figures 2 to 5 shows that for a given control rod position there are several different values of APDs and peak factors. This occurs because it was obtained in the experiments several states with the BS1 control rod fixed and the others in different positions. The same happened to the BC1, BC2 and BS2 control rods. For example, each point presented in Figures 2 and 3 for the same BS1 control rod position corresponds to one of these reactor critical states.

Therefore, correlating the peak factor to the position of only one control rod is not sufficient to obtain a good estimate. Figures 2 to 5 indicate that it is necessary to consider the position of the four control rods since all of them affect the power distribution in the reactor, and show also that the correlation between peak factor and control rods positions are strong. The reactor protection system can utilize the control rods to infer the peak factor during the reactor operation.

5.2. Determination of the peak factor from the axial power differences data

The axial power difference obtained from the out-of-core detectors can be utilized to infer the power peak factor. Figures 6 and 7 present the peak factor as a function of the APD^N and APD^W obtained from the out-of-core detectors located at the north and west sides of the core. The points presented in these figures are the same of those from Figures 2 and 3 produced by the movement of the BS1 control rod. Figures 6 and 7 show that the correlation between the peak factor and the APDs are not as clear as the ones involving the control rods positions. There are several reactor states with different peak factors for the same axial power difference, and one does not find a clear pattern relating the APDs and the peak factor.

There are two main reasons for this behaviour. First, the out-of-detectors, located far from the core, are not able to detect small differences in the power density distribution inside the core. Small changes in the peak factor are not detected by APD^N and APD^W. The second reason is related to the fact that the reactor has states with different power density distributions which produce similar APD^N and APD^W.

Observing Figure 1, one sees that inserting the BC1 control rod (A) close to the north core side, and compensating that negative reactivity by withdrawing the BC2 control rod (B) close to the core west side (with the BS1 and BS2 fixed) produces the same APD^N as if one inserted the BS1 control rod (S) and withdrew the BS2 control rod (S) (with the BC1 and BC2 fixed). The APD^N are similar but the peak factors are located in different core quadrants.

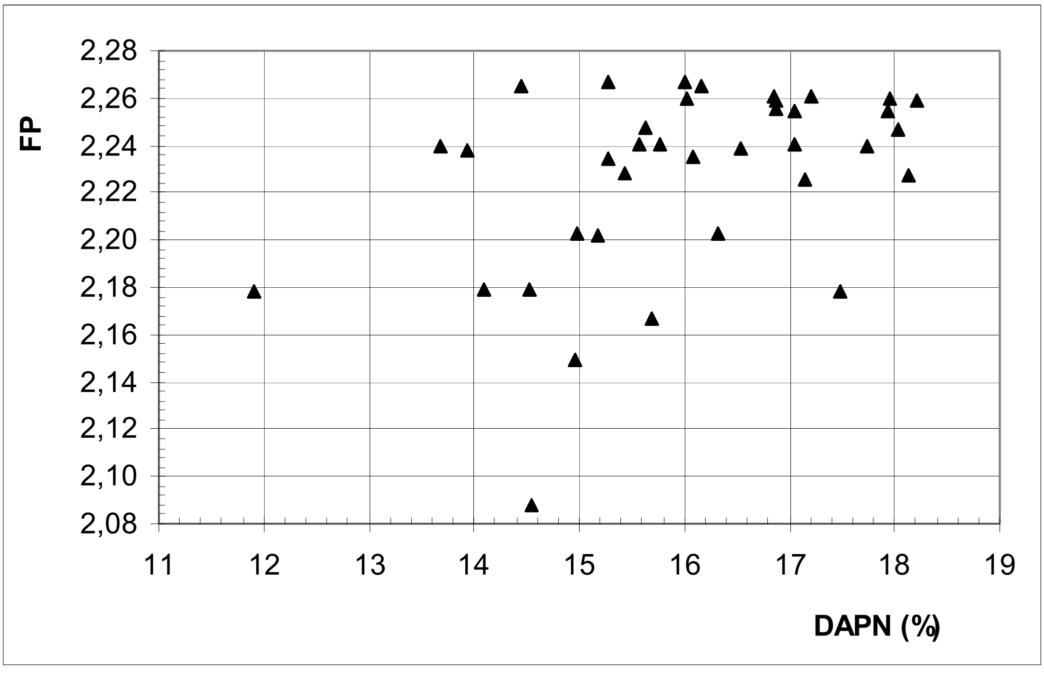

FIG. 6 Peak factor (FP) as a function of the axial power difference from the north side (DAPN)

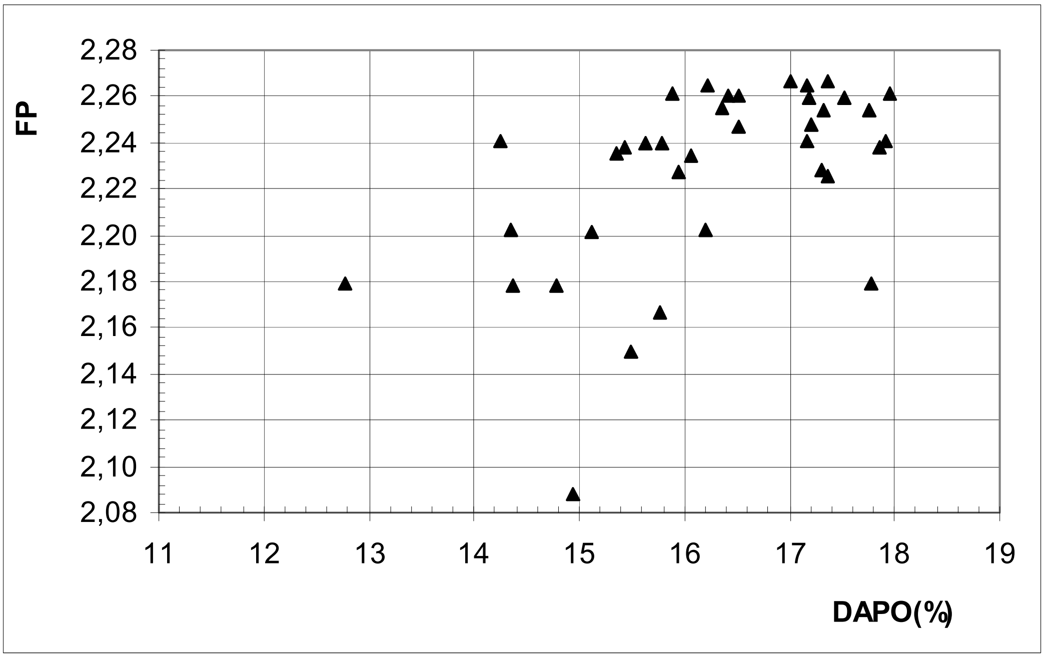

FIG. 7 Peak factor (FP) as a function of the axial power difference from the west side (DAPO)

In this case, the symmetry which exists in the core since the four control rods have similar worths shows that the APD^N itself is not able to identify the peak factors. The two reasons mentioned before do not facilitate the work for the APDs.

Figures 8 and 9 show two subsets from the data presented in Figure 6. Figure 8 shows the behaviour of the peak factor as a function of the APD^N considering only those states in which the BS1 control rod was inserted and the BS2 control rod withdrawn, while the two others did not move. The peak factor changes from 2.09 to 2.24 while the APD^N changed from 14,5 % to 17,2 %. Figure 9 shows the behaviour of the power peak factor as a function of the APD^N for other core states with different control rods configurations. In this case, while the BC1 and BC2 control rods are inserted into the core, the BS1 and BS2 control rods are withdrawn to maintain criticality. The power peak factor does not change much, varying from 2.24 to 2.27 in the APD^N interval between 14 % and 16,5 %.

Figures 8 and 9 show that exists a correlation between the power peak factor and the axial power difference from out-of-core detectors but it is necessary a detailed analysis of the data to identify it. This analysis has to be made in order to decrease the uncertainties caused by the large scattering of the data as can be seen in Figures 6 and 7. The approach which appears to be the best to solve the problem is to consider in the correlation to obtain the power peak factor information from both the APDs and the QPDs, quadrant power differences.

In this way, the power peak factor would be a function of the axial power difference from detectors from each core side and also from the quadrant power difference from the top and the bottom of the core. With such inputs it seems possible to differentiate those peak factor data points with the same axial power difference value.

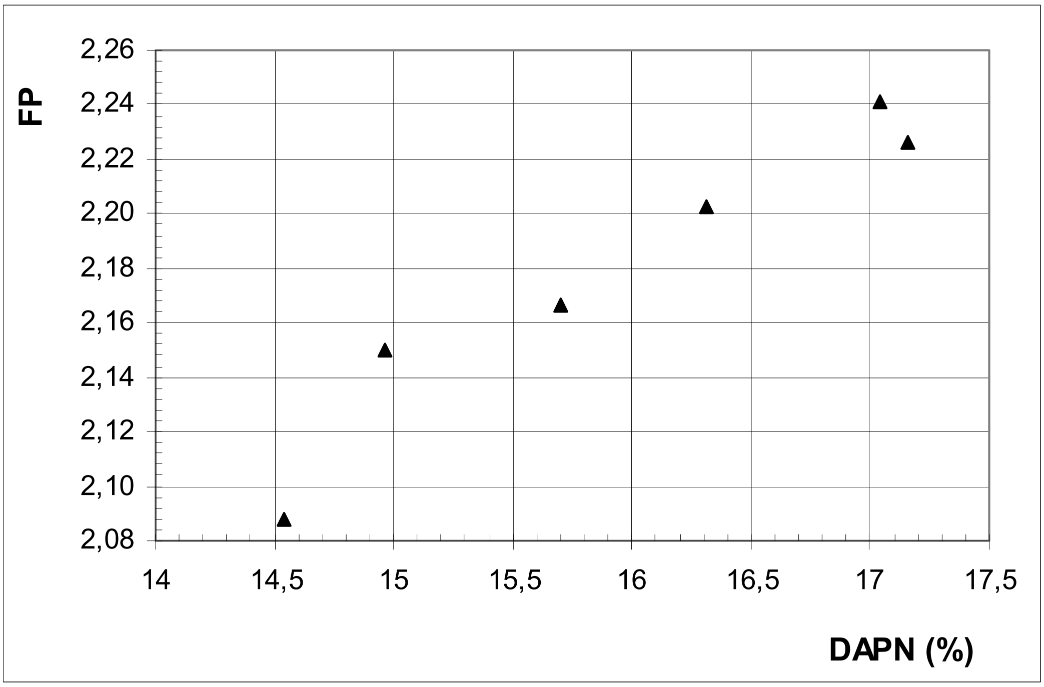

FIG. 8 The power peak factor (FP) as a function of the APD^N (DAPN) for the case of BS1 and BS2 control rods moving and the other two fixed

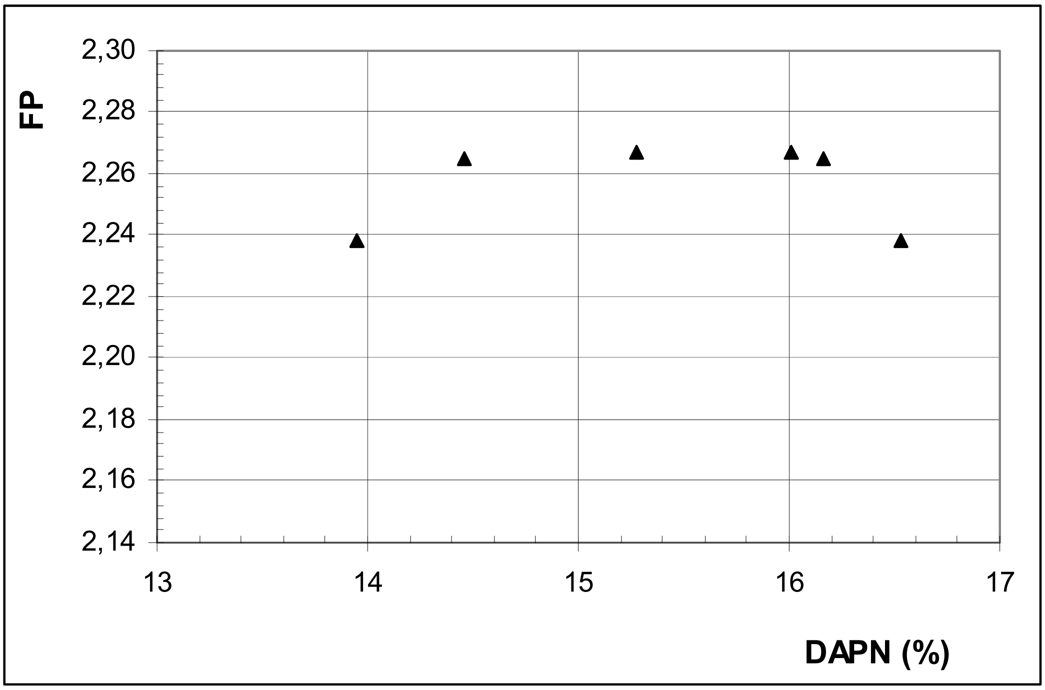

FIG. 9 The power peak factor (FP) as a function of the APD^N (DAPN) for the case of BS1 and BS2 control rods moving together and compensation by the BC1 and BC2 control rods also moving together

6. PRELIMINARY CONCLUSIONS

The experimental results obtained in the IPEN/MB-01 reactor have shown that there is a correlation between the power peak factor and the control rods positions. There is a clear pattern in their relationship, which allows one to say that the control rods positions can be used as a means to determine the power peak factor during a reactor operation and be utilized in a small reactor protection system.

The same experimental data have shown that the relationship between the power peak factor and the axial power differences from out-of-core detector signals is much more complex. There is a large scattering of the data points and very different behaviours for different core configurations. The axial power difference cannot describe accurately what goes on inside the reactor core. In this case it appears necessary to obtain the power peak factor taking into consideration the axial power differences and the quadrant power differences. A correlation considering these two parameters will be studied as a continuation of this work.

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REFERENCES

[1] SOUZA, R.M.G P., MOREIRA, J.M.L., "Determinação do fator de pico utilizando sinais de detectores out-of-core em reatores de pequeno porte", International Nuclear Atlantic Conference, (Proc. Conf., Rio de Janeiro, 2002) (CD-ROM).

[2] MAIORINO, J.R., et al., "Projeto Nuclear da Unidade Crítica IPEN/MB-01", VII Encontro de Física de Reatores e Termo-hidráulica, ENFIR, (Proc. Conf. Recife, PE, 1989), p. 311.

[3] MOREIRA, J.M.L. e SOUZA, R.M.G.P., "Rotina Experimental - Determinação de Fator de Pico Utilizando Detectores Externos ao Tanque de Moderador", Internal Report, CTMSP (2001).

[4] MOREIRA, J.M.L. e SOUZA, R.M.G.P., "Determinação de Fator de Pico Utilizando Detectores Externos ao Tanque de Moderador - Plano de Trabalho", Internal Report, CTMSP (2000).

[5] MOREIRA, J.M.L. e SOUZA, R.M.G.P., "Determinação da Densidade de Potência Local Utilizando Sinais da Instrumentação Nuclear", Internal Report, CTMSP, rev. 01 (2001).

[6] FOWLER, T. B., VONDY, D. R., CUNNINGHAM, G. W., "Nuclear Reactor Core Analysis Code: CITATION", ORNL-TM-2496 (1971).

[7] BARHEN, J., et al., "The HAMMER Code System", Technion, EPRI-NP-565 (1978).

REACTOR DOSIMETRY IN REACTOR PRESSURE VESSELS LIFETIME MANAGEMENT

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Abstract

Assessment of the current state and residual lifetime of the reactor pressure vessels (RPV) of Kozloduy NPP Units 1 to 6 has been carried out by monitoring the irradiation conditions of RPV and metal surveillance specimens irradiated during the reactor operation. Calculational methodology has been developed for the evaluation of the RPV neutron fluence of VVER-1000 and VVER-440. Verification of calculated neutron fluence on the vessel has been performed by the use of the experimental data of induced activity of ex-vessel detectors that have been installed on all units of Kozloduy NPP. Various loading patterns of the reactor core have been investigated in order to achieve neutron fluence reduction, which is an important requirement for managing and extension of the RPV lifetime.

1. INTRODUCTION

The reactor pressure vessel (RPV) integrity of VVER-type reactors is a problem of present interest, in connection with the substantiation of their safety and service lifetime. In Kozloduy NPP, four reactors of type VVER-440/230 (Units 1 to 4) and two of later generation of type VVER-1000/320 (Units 5 and 6) are in operation.

The steel embrittlement of a reactor pressure vessel caused by the neutron irradiation during the operation of NPP could be the reason for a break of the vessel integrity when accident shutdown of the unit is performed. The assessment of the fast neutron fluence onto the RPV is therefore required for evaluation of vessel steel degradation and for revision of the safety margins limits. These limits determine the lifetime of the reactor relevant to the irradiation embrittlement.

Following the requirements for safety operation a Reactor Pressure Vessels Surveillance Program is being carried out on Kozloduy NPP. According to this program assessment of the current state and residual lifetime of RPV has been carried out by monitoring the irradiation conditions of RPV and metal surveillance specimens (SS) from the same vessel metal irradiated during the reactor operation. The RPV neutron fluence evaluation methodology has been developed and applied on Units 1 to 6 of Kozloduy NPP since 1989. The evaluation of neutron fluence is used for determining the current state of RPV metal embrittlement and the RPV residual lifetime, and for selecting the fuel loading scheme, which warrants reducing the RPV neutron fluence rate.

2. EVALUATION OF THE METAL EMBRITTLEMENT UNDER NEUTRON IRRADIATION

The Russian standard [1] for VVER reactors uses the following relation for determination of the temperature of neutron embrittlement:

$$T_k = T_{k0}^B + \Delta T_F, \quad (1)$$

where T_{k0}^B, ^0C, is the initial temperature of neutron embrittlement after annealing, and ΔT_F, ^0C, is the shift of the temperature of neutron embrittlement. The nondestructive method for evaluation of the metal embrittlement under neutron irradiation is based on the relation between ΔT_F and the fluence F, cm^{-2}, of neutrons with energy above 0.5 MeV:

$$\Delta T_F = A_F (F/F_0)^{1/3}, \quad (2)$$

where A_F, ^0C, is a chemical coefficient of neutron embrittlement, and $F_0=10^{18}$, cm^{-2}.

3. NEUTRON FLUENCE CALCULATIONAL METHODOLOGY

Calculational methodology has been developed for the evaluation of the RPV neutron fluence of VVER-1000 and VVER-440 [2-11]. The software package used is presented by the flowchart in Figure 1.

The distribution of the neutron flux and its responses on the RPV and adjacent zones of VVER are obtained as a solution of the kinetic equation of neutron transport. Because of the complexity of neutron production and transport from the core to the RPV the task is divided into two parts. The first part describes the generation of fission neutrons and their distribution in the reactor core, and the second part describes the neutron transport with fixed sources. These sources are determined by the fission neutrons generated in the core, and are appropriate for neutron fluence and induced activity calculation [4].

The determination of the power distribution and the fuel burnup distribution in the core taking into account the operational history has been carried out using the three-dimensional codes PYTHIA [12] for VVER-440, and TRAPEZ [13] for VVER-1000. These codes use libraries of effective diffusion cross sections generated by the spectral code NESSEL [14].

The ASYNT method [5] has been developed for calculations of neutron fluence on the vessel. The evaluation of flux/responses is reduced to the space and energy integration of the product of three-dimensional adjoint solution and the appropriate source, determined by the loading patterns and operational regime realized. The three-dimensional adjoint solution is being synthesized using the two- and one-dimensional solutions of adjoint neutron transport equation obtained by the code DORT [15]. The three-dimensional code TORT [16] is applied for calculations of neutron fluence on SS.

The DOSRC software package [17] has been used to transform the assembly-wise and pin-wise output data of PYTHIA and TRAPEZ codes for power and burnup distribution in hexagonal three-dimensional geometry to neutron source input data in (r,θ,z)-geometry format appropriate for neutron transport calculation by the discrete-ordinates codes DORT and TORT. The neutron fixed source is generated in multigroup presentation accounting for the burnup-dependent neutron production and energy distribution. There are two branches of the package for VVER-440 and VVER-1000.

For both types of reactors, VVER-440 and VVER-1000, two problem-oriented libraries have been created [7]: BGL440 and BGL1000 respectively, by collapsing the fine-group library VITAMIN-B6 (199 neutron and 42 gamma groups) to 67-group structure (47 neutron and 20 gamma groups). The libraries consider the features (detailed one-dimensional geometry and material compositions) of the respective reactor and contain upscattering data for the five thermal-energy groups. The order of scattering of the Legendre expansion is P5.

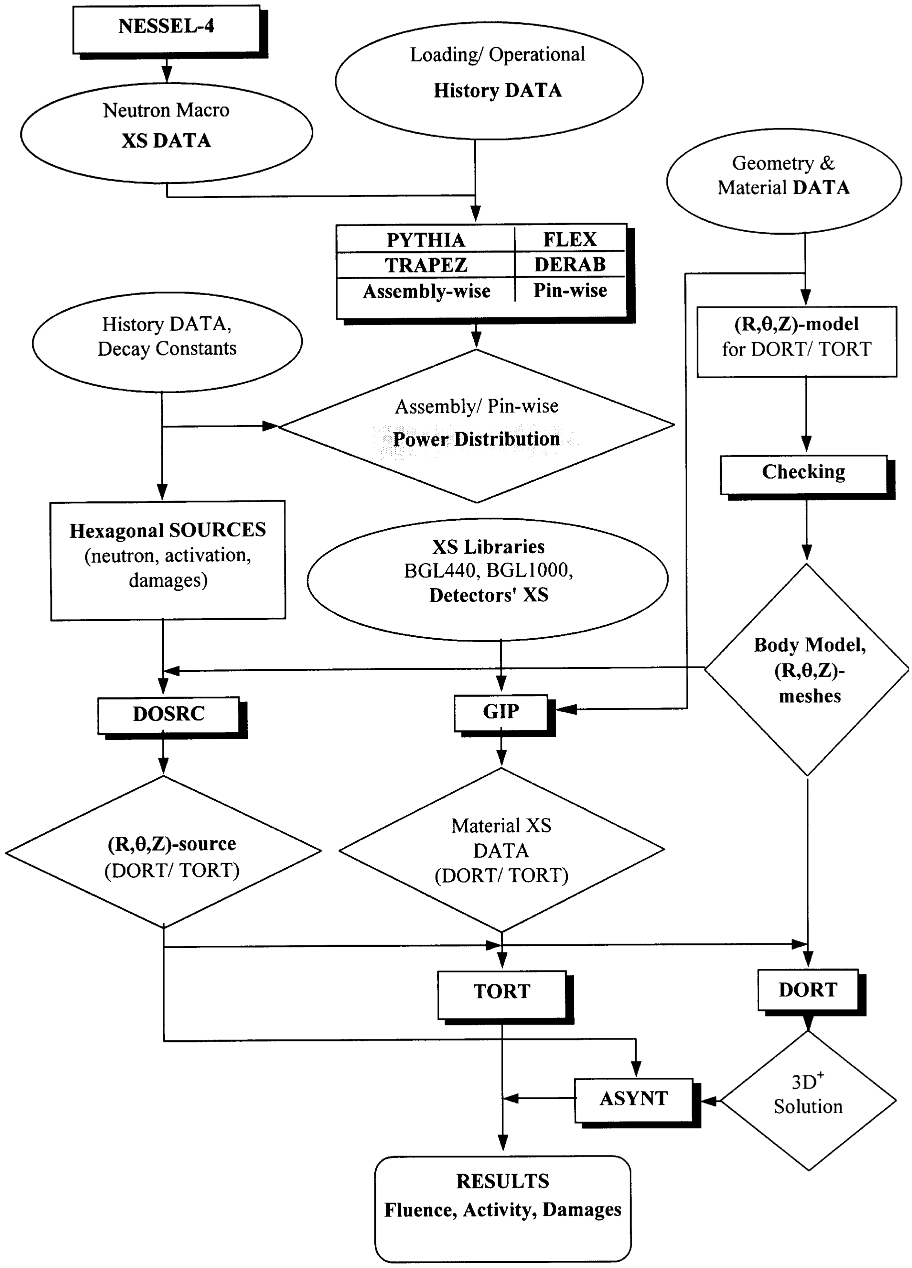

FIG. 1. Flowchart of the Software Package for Neutron Fluence Responses Calculation

Input data sensitivity analyses for choice of spatial coordinates' meshes, order of velocity angle quadratures S_n, polynomial neutron scattering anisotropy approximation P_n, convergence constant value, and neutron constants libraries have been carried out to reduce the calculational method errors [2, 3, 5, 6, 8].

4. VERIFICATION OF THE NEUTRON FLUENCE

The reliable calculation of the neutron fluence is limited by the data (nuclear and structural) used in the neutron transport calculations. That is why it is always advisable to compare calculational with measured values. Since the neutron flux is not directly measurable, flux-related values, such as radioactivity of an isotope generated by the neutron interaction, are measured and compared.

4.1. Neutron fluence on RPV

The neutron fluence is evaluated by calculations at the surveyed places of VVER-1000 RPV (the seam welds 3 and 4, and base metal, on 8 deg, which is the direction of maximal neutron exposure) and VVER-440 RPV (the seam weld 4 and base metal, on 30 deg for standard core loading, and on 13 deg for loading with dummy cassettes in the periphery). The developed RPV neutron fluence evaluation methodology has been applied since 1989 on Units 1 to 6 of Kozloduy NPP.

Following the RPV Surveillance Program verification/validation of the calculated neutron fluence on the vessel for each operated cycle is being carried out by the use of the experimental data of induced activity of ex-vessel detectors. For this purpose, racks' device with iron, niobium, and copper wire detectors is being disposed in the air cavity behind the vessel on each unit of Kozloduy NPP. The horizontal wire detectors comprise a 60-deg sector, and the vertical ones are situated along the core height [6]. The calculated ex-vessel detector activities of ^{54}Mn are in good consistency with the measured ones. The calculated ^{60}Co and ^{93m}Nb activity data overestimate the measured ones. For example, the mean overestimation for Unit 5 is about 10% and 26%, respectively.

The verification of calculated neutron fluence onto the VVER-440/230 pressure vessels is a very topical task in particular referring that some of the reactors of this type have been operated for the major part of their design lifetime. There are no surveillance assemblies in VVER-440/230, but other activities have been carried out in order to assess experimentally the RPV metal embrittlement and the RPV residual lifetime. Templates have been taken out from the inner wall of the Unit 2 vessel (1992), and scraps (1995) and templates (1996) of the Unit 1 vessel for embrittlement study. The verification of neutron fluence calculation based on measured ^{54}Mn activity of scraps demonstrated a good consistency between calculated and measured results (within 9%) [9].

4.2. Neutron fluence on surveillance specimens

There are six sets of surveillance assemblies in the VVER-1000/320 type of reactors. Each set is placed within 60-deg sector of symmetry of the reactor core, and contains five cylindrical assemblies located at different azimuth positions on the baffle, about 30 cm over the reactor core upper edge. Each assembly contains six cylindrical containers with SS and neutron detectors. The containers are arranged at two axial levels, symmetrically around the axial axis of the assembly [10].

Detailed calculational modeling of the neutron fluence onto the surveillance specimens of VVER-1000/320 was performed, taking into account the spatial and time power distribution in the reactor core. The three-dimensional code TORT, the neutron cross sections library BGL1000, and activation data from the IRDF-90 [18] file were used for these calculations.

The strong neutron flux radial and axial gradients through each surveillance assembly do not allow obtaining a set of specimens with close neutron irradiation needed for metallurgical analysis. The neutron fluence at the middle of the specimens varies around the axis of each assembly within 15% for the upper level and 35% for the lower level.

The measured ^{54}Mn activity of some specimens was applied to determine the experimental neutron fluence values according to the fluence/ activity relation [4].

The results obtained [10] for the neutron fluence of metal specimens from surveillance assemblies' sets of Kozloduy NPP Units 5 and 6 have been used for assessment of the current embrittlement and prognosis for the years to come in accordance with the Surveillance Program.

5. RPV LIFETIME PROGNOSIS

The RPV residual lifetime relevant to the neutron irradiation is limited by the maximal neutron fluence F_{max} and depends on the fluence of the operation time. The residual lifetime (RLT) in cycles is determined as

$$RLT=(F_{max}-F)/F_{avr}, \tag{3}$$

where F_{avr} is the prognostic fluence of standard cycle of 290 FPD. The maximal neutron fluence is determined by the maximal critical temperature T_{ka} which is a characteristic of the vessel:

$$F_{max}=(T_{ka}/A_F)^3. \tag{4}$$

The results obtained for the residual lifetime relevant to the RPV neutron embrittlement of Kozloduy NPP Units 1 to 6 are presented in Table I.

Table 1 Residual Lifetime Relevant to the RPV Neutron Embrittlement

Unit / Reactor Type	Start Year	Operated Cycles to 2002	Residual Lifetime	
			Cycles	Until Year
1 / VVER-440	1974	21	7	2008
2 / VVER-440	1975	21	11	2012
3 / VVER-440	1980	16	26	2027
4 / VVER-440	1982	16	> 26	> 2027
5 / VVER-1000	1987	7	> 40	> 2041
6 / VVER-1000	1991	6	> 40	> 2041

6. CONTROL APPLICATION AND MANAGEMENT DECISIONS

On the basis of the applied methodology more realistic evaluation for the neutron fluence of the RPV of Units 1 and 2 has been obtained. Thus the operation was substantiated for the 16th cycle of Unit 2 reactor, the 17th and the 18th cycle of Unit 1 reactor.

The calculational and experimental data obtained for metal specimens and neutron detectors of the Units 5 and 6 surveillance sets have been used for modification and improvement of the Surveillance Program of Kozloduy NPP Units 5 and 6.

Various loading patterns of the reactor core have been investigated in order to achieve neutron fluence reduction, which is an important requirement for managing and planning RPV lifetime.

Diverse low-leakage loading schemes with burned-up assemblies in the periphery of the reactor core are applicable. Low-fluence loading (LFL) is the scheme when burned-up assemblies are put only in direction of maximal exposure. In this case the reduction of the neutron fluence is about 20 to 30% toward the standard core loading (Table II). Low-leakage-and-fluence loading (LLFL) is the scheme with 36 burned-up assemblies. It reduces the neutron fluence with about 30 to 40% toward the standard loading.

Another way to reduce the neutron fluence is to put 36 dummy cassettes in the reactor core periphery instead of fuel assemblies. In this case the direction of maximal azimuth neutron exposure is shifted from 30 deg to 13 deg and 47 deg (for 60-deg sector of symmetry), and the neutron fluence decreases with about 70% toward the standard loading. The combination of dummy and low-fluence loading additionally diminishes the neutron fluence. This scheme is being applied on Units 1 to 3 of Kozloduy NPP.

Table 2 Neutron Fluence Reduction

Loading Scheme	Description	Reduction Effect
LLL (low leakage)	IN-IN-OUT[a] & max OUT-IN-IN	Insignificant
LFL (low fluence)	OUT-IN-IN[b] & max IN-IN-OUT	~ 20-30%
LLFL (LL & LF)	IN-IN-OUT	~ 30-40%
Dummy	Dummy cassettes in the periphery of the reactor core	Shift max ~ 70%
Dummy & LFL	Dummy & LFL	≥ 80%

[a] IN-IN-OUT - the assembly is in the periphery during its third or fourth cycle in the core.
[b] OUT-IN-IN - the assembly is in the periphery during its first cycle in the core.

7. CONCLUSION

Calculational methodology for the evaluation of the RPV neutron fluence of VVER-1000 and VVER-440, and based on advanced methods, codes, neutron constants libraries and experimental verification by induced activity measurements, has been developed.

The neutron fluence validated data have been used for assessment of the vessel metal degradation and the residual lifetime relevant to the RPV neutron embrittlement of Kozloduy NPP Units 1 to 6.

The applying of loading patterns with burned-up assemblies in the periphery of the reactor core leads to maximal reduction of the neutron fluence, which is an important requirement for managing and extension of the RPV lifetime.

REFERENCES

[1] GOSATOMENERGONADZOR, Regulations and Calculation Norms for Equipment and Tubing Strength of the Nuclear Energy Facilities (PNAE G-7-002-86, PNAE G-7-008-89), Energoatomizdat (1989).

[2] ILIEVA, K., APOSTOLOV, T., BELOUSOV, S., ANTONOV, S., "Neutron Fluence Determination for VVER-440 Reactor Pressure Vessel Ageing Surveillance", Reactor Dosimetry, ASTM STP 1228, Harry Farrar IV, E. Parvin Lippincott, John G. Williams, and David W. Vehar, Eds., American Society for Testing and Materials, Philadelphia (1994) 38-44.

[3] ILIEVA, K., BELOUSOV, S., ANTONOV, S., ZARITSKY, S., BRODKIN, E., "Test Calculation of the Neutron Flux on VVER-1000 Reactor Pressure Vessel", Reactor Dosimetry, ASTM STP 1228, Harry Farrar IV, E. Parvin Lippincott, John G. Williams, and David W. Vehar, Eds., American Society for Testing and Materials, Philadelphia (1994) 343-347.

[4] ILIEVA, K., ANTONOV, S., BELOUSOV, S., "Calculation Modeling of Detector Activity in the VVER/ PWR Reactor Pressure Vessel Surveillance", Nuclear Science and Engineering **122** 1 (1996) 131-135.

[5] BELOUSOV, S., ILIEVA, K., "ASYNT - Adjoint Synthesis Method for Neutron Irradiation Assessment of VVER/ Pressurized Water Reactor Pressure Vessel", Nuclear Science and Engineering **126** 2 (1997) 239-244.

[6] ILIEVA, K., APOSTOLOV, T., BELOUSOV, S., PENEV, I., POPOVA, I., "Validation of Neutron Fluence Calculation Methodology for VVER-440 Vessel", Nuclear Science and Engineering **125** (1996) 243-247.

[7] BUCHOLZ, J., ANTONOV, S., BELOUSOV, S., "BGL440 and BGL1000 Broad Group Neutron/ Photon Cross Section Libraries Derived from ENDF/B-VI Nuclear Data", IAEA, INDC(BUL)-15 (1996).

[8] BELOUSOV, S., ILIEVA, K., ANTONOV, S., "Influence of Iron Data from Different Neutron Cross Section Libraries on VVER PV Fluence Calculation", Reactor Dosimetry, Hamid A. Abderrahim, Pierre D'hondt, Bohumil Osmera, Eds., World Scientific Publishing Co. Pte. Ltd., Singapore (1998) 473-479.

[9] APOSTOLOV, T., ILIEVA, K., BELOUSOV, S., PETROVA, T., PENEV, I., POPOVA, I., ANTONOV, S., "Experimental Assessment by Scraps Activity of Neutron Fluence onto the Reactor Vessel of Unit 1 at Kozloduy NPP", Nuclear Engineering and Design **186** (1998) 361-368.

[10] BELOUSOV, S., ILIEVA, K., POPOVA, I., "Calculation of Neutron Flux/ Responses at the Surveillances and Pressure Vessel of VVER-1000/320 Type of Reactor", Proc. of RP&S Topical Meeting, Nashville, USA, ANS (1998) **I** 440-447.

[11] ILIEVA, K., BELOUSOV, S., APOSTOLOV, T., "Testing of Multigroup Neutron Cross Section Libraries on Neutron Transmission through VVER-440 Vessel", Nuclear Science and Engineering **131** (1999) 282-285.

[12] THOMAS, S., "Programm PYTHIA zur Berechnung des Makroabbrands in Druckwasserreaktoren vom Typ WWER", Berlin, K.A.B. (1996).

[13] SCHULZ, G., “Code Manual TRAPEZ Version 6.07a”, Berlin, K.A.B. (1997).
[14] SCHULZ, G., “NESSEL Version 6.0”, Berlin, K.A.B. (1994).
[15] RHOADES, W. A., CHILDS, R. L., “DORT”, Nuclear Science and Engineering **99** 1 (1988) 88-89.
[16] RHOADES, W. A., CHILDS, R. L., “TORT Three-Dimensional Discrete Ordinate Neutron/ Photon Transport Code with Space-Dependent Mesh and Quadrature”, ORNL-6268 (1987).
[17] ANTONOV, S., POPOVA, I., BELOUSOV, S., ILIEVA, K., “Software for Neutron Source Input for Discrete Ordinates Neutron Transport Codes”, Report of IAEA RC 8621/1996; BgNS Transactions **3** 1 (1998) 100-103.
[18] INTERNATIONAL ATOMIC ENERGY AGENCY, “IRDF90 v.2, the International Reactor Dosimetry File”, IAEA-NDS-141.

R&D BACKGROUND FOR LIFE MANAGEMENT IN HUNGARY

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Abstract

Plant Life Management is the most important task for the TSO organizations in Hungary for the following years. NPP Paks has decided to extend the lifetime to 50 years operation. Four VVER-440/V213 units are in operation a Paks, and they produce 40% of nation electricity. The nuclear electricity is not only contributing the country to keep the release of the polluting gases low, but one of the cheapest too. Several institutes and universities are working on the life extension program. The life barriers of the Paks units are: the embrittlement of the RPV-s and the corrosion of the steam generator pipes. Both barriers allow the planned 50 years operation without the reduction of safety in case of using correct mitigation actions. This paper introduces the R&D efforts for Life Management of the RPV-s.

1. INTRODUCTION

The lifetime of structural materials influences the safe service time of equipment to a great extent. Longer lifetime is generally more advantageous both from the economic and environmental aspect, as waste volume decreases if decommissioning occurs less frequently. Enhancement of lifetime calculation is an important economic aspect in case of valuable equipment. WWER-440 V213 units provide about 40% of the electricity of Hungary, and play a similarly important role in several countries in the area.

The units of NPP Paks are in good shape. In 2000 three units of NPP Paks have been on the IAEA list TOP 25. This list shows the highest efficiency operated rectors worldwide. Compare these results with the fact, that only 2 reactors from US and 1 from Japan is on the list this is an excellent result showing the high reliability of the plant. The utility has life management plan to extend the lifetime until 50 years of operation and increase the power of the units up to 500 MW output.

To support the life management plan of NPP Paks Atomic Energy Research Institute (AEKI) made a research proposal for the last National Research Foundation Call, which is part of the Hungarian Economic Development Program named: "Széchenyi program". The proposal got support, and the project started. AEKI also participates in several cost shared programme in the 5. Framework program of the EU, and in Coordinated Research Programmes of the IAEA. All of these activities import the necessary knowledge into the country and ensure a firm basis for the Life Management of the RPV-s.

2. MITIGATION ACTIONS FOR WWER-440/V213 TYPE RPV-S

In lifetime management of nuclear power plants the reactor pressure vessels play a crucial role in safety. Their safe lifetime can be prolonged by mitigation actions, thus plant lifetime is not limited technically by the RPV. Generally the life limiting consideration of the WWER-440 vessels are the PTS calculations results.

To extend the safe operational lifetime of the WWER-440 V213 vessels –among other options using hardware solutions- the following actions -or at least one of them -are suggested:

1. Consideration of the clad in elastic-plastic PTS analysis.
2. Application of Master Curve
3. Increasing the understanding of vessel annealing

These actions require further study of the material ageing mechanism. The paper introduces the ongoing national project to enhance the cooperation with the partners having similar problems, and provides the list of the international coordinated projects that also will contribute the RPV Life Management in the country.

2.1. The clad properties

Modern RPV-s is cladded with stainless steel on the inside to reduce the radioactive corrosion products, and to protect the base material of the vessel from corrosion degradation. The clad also facilitates the cleaning of the vessel.

However, the mechanical and physical properties of the clad are different from those of the base steel of the vessels. The clad is usually produced by UP (under powder) strip welding resulting in a special and non-homogeneous welded structure. The WWER-440 V213 clad consists of three layers.

The first layer is welded by over alloyed electrode and it is mixed with the melted base material. This layer is quite inhomogeneous, and the mechanical properties differ from site to site. Under this layer there is a narrow heat affected zone of the base material, the properties of which are unknown. Underclad microcracks (size under NDT- non destructive testing- sensitivity) may exist in this zone due to the high thermal stresses caused by the different linear heat expansion coefficients of the two materials.

The second and third layers are made of stabilized austenitic steel, and the structure is weldment. It is well known that the weldment of austenitic steels contains a relatively high amount of delta ferrite, mainly on the grain boundaries. Thus the grain boundary is sensitive to thermal and radiation embrittlement, and behaves differently from the austenitic stainless steel When transient temperature changes (like heat-up, shut down, etc.) occur during service high thermal stresses develop at the clad - base metal boundary due to the different thermal expansion coefficient of the two materials and this stress can initiate a propagating crack.

Stress distributions have been calculated at LBLOCA (large break loss of coolant accident) supposing different elastic and elastic-plastic behaviour of non-aged cladding. The results clearly show that linear elastic stress analyses of the clad area are quite over conservative, but elastic-plastic analyses can be performed only if the tensile flow curves and fracture toughness of the aged clad are available [FIG.1] .The IAEA guidelines [1] prescribes the analysis of surface cracks where mechanical properties of the clad are not known. In the WWER-440 reactors the size of the postulated defects is up to ¼ wall thickness (35 mm). Without knowing the properties of the 8 mm thick cladding no correct analysis can be performed.

The OECD organized the so called "ICAS" project and the IAEA co-coordinated research project on WWER 440 weld resulted in about 40% scatter in lifetime assessment of RPV-s when PTS (pressurized thermal shock) transients were calculated. The scatter is mainly caused by the use of different and uncertain clad properties in the calculations, or in some cases the mechanical properties of the clad are considered to be the same as those of the base material. Similar assessments were performed at AEKI [2,3].

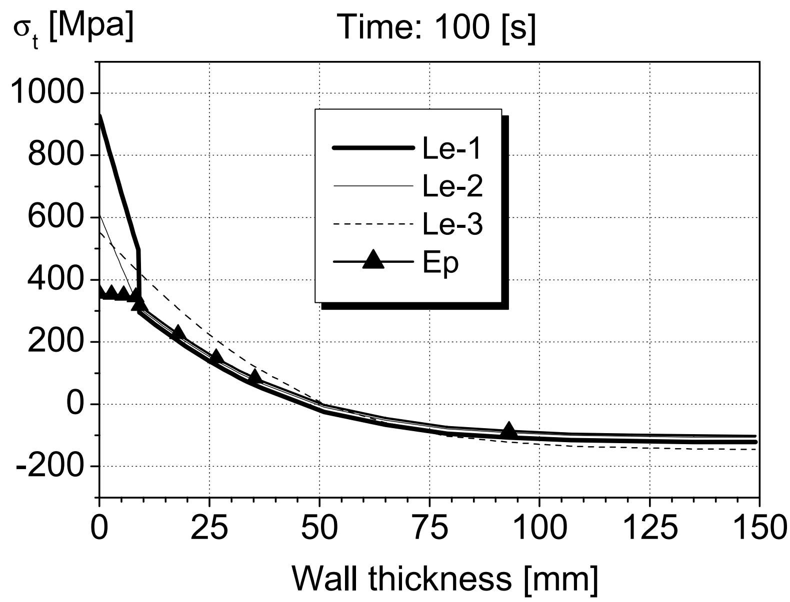

FIG. 1 Stress distribution in WWER 440 RPV wall during LBLOCA transient calculated with different linear elastic approaches and with elastic plastic analyses.

These facts prove the necessity of knowing the aged clad properties. Presently very few data is available. A limited number of clad specimens were irradiated in a research reactor in Hungary [4]. The results are summarized in FIG. 2.

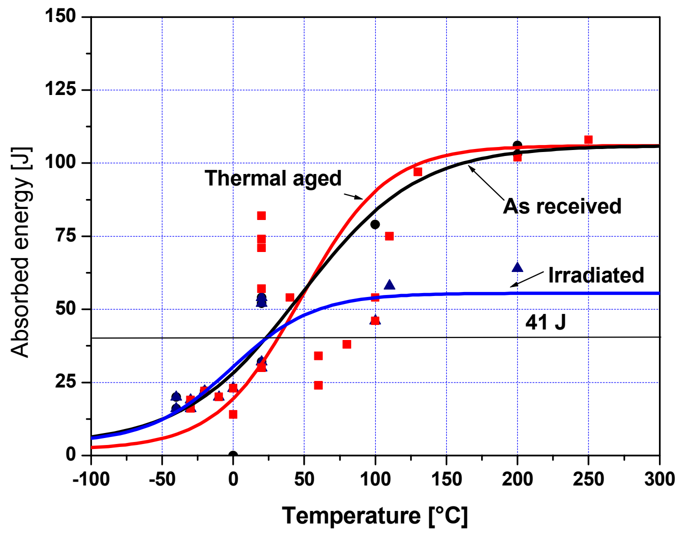

FIG. 2 Effect of accelerated thermal ageing (2000 hours at 350°C), and irradiation ($2*10^{23}$ n/cm^2 E>1MeV) on WWER-440 clad.

The results clearly show that the irradiation and thermal ageing affect the clad properties to some extent, but not sufficient clad ageing information exists. Within the frame of the EU FP5 ATHENA network several existing clad data in Russia, Czech Republic, Germany, and France and in US have been gathered. Based on them further study of the clad behaviour and ageing mechanisms started. Clad material cut from unit 8 at Greifswald is provided for the project by FZR institute (Germany). Several Hungarian institutions and two foreign partners (FZR and JRC Institute for Energy) are participating in the consortium to study the clad ageing.

2.2. Use of Master Curve

The future trend is to use the so-called "master curve" instead of the presently applied trend curves. The ASTM 1921 standard describes the method [5]. The temperature dependence of all pearlitic steels is similar. Using this similarity a uniform curve was elaborated for all pearlitic steels (steels of RPV-s) at VTT, which is called "Master Curve". The evaluation of the transition temperature is based on fracture mechanics measurements. The master curve generally gives a 10 % higher transition temperature shift at high fluence, when compared with the transition temperature measured on Charpy specimens, but the initial master curve transition temperature is generally below the value measured by Charpy specimens.

Validation of the master curve is under process now. In 1996-99 IAEA organized a co-coordinated research program to inspect the measuring technology. About 20 laboratories participated, including AEKI. The scatter of the results does not exceed the scatter of the Charpy transition temperature measurements. The EU sponsors the shared costs program FRAME to contribute the necessary further validation before the master curve is widely used in reactor safety calculations. Using the so-called NAS call of the EU FP5 AEKI also participating in this project.

The multitemperature master curve determination is expected to be useful for WWER-440 type reactors, where fracture mechanics specimens are used in the surveillance program. As an example the results obtained at NPP Paks unit 1 surveillance after four years of irradiation (equal to about 70 years of operation due to the high lead factor) fit well to the specific Russian trend curve for 15H2MFA welds shifted on the basis of Charpy measurements (see FIG.3.) [6].
At lower fluence the use of master curve provides more benefits. Even at high fluence the use of the master curve is beneficial, as it is a direct fracture toughness measurement it is not necessary to apply an extra safety margin.

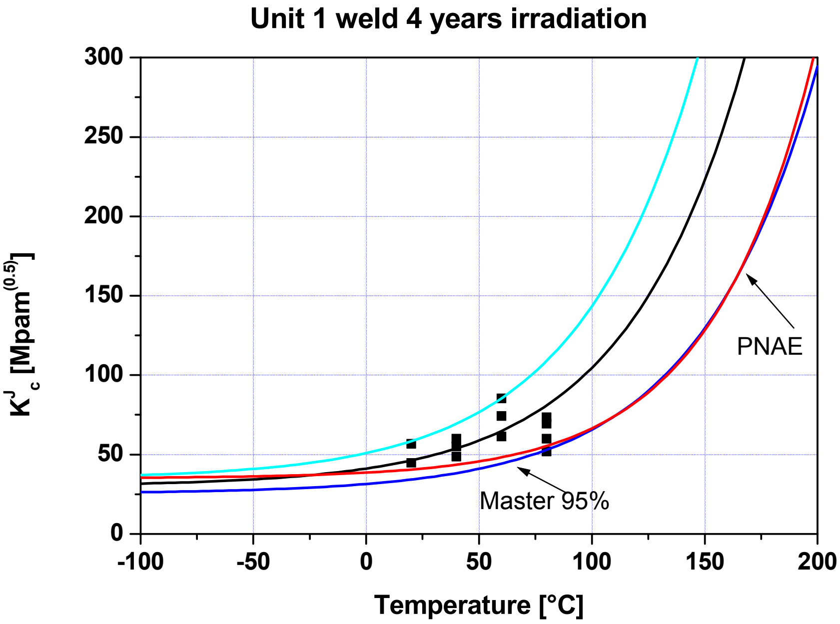

FIG. 3 Comparison of the Master Curve and trend curve using the surveillance data of Paks unit 1 weld

2.3. Annealing of Reactor Pressure Vessels

Irradiation damage of the material of reactor walls influences the lifetime of nuclear power plants to a great extent. Neutron fluence at the RPV wall is the highest in WWER reactors as compared to all other types of pressurized water reactors. One of the possible alternatives in the lifetime extension of WWER-440/V213 units is the annealing of the RPV. During neutron irradiation and annealing several diffusion mechanisms occur. Irradiation increases the number of dislocations, causes segregations and irradiation also increases diffusion rate. Diffusion partly restores the dislocation density thus reducing irradiation damage. Annealing at 450-500 ^{o}C may cause complete regeneration of the dislocation structure; segregations re-dissolve into the base matrix while impurities, mainly phosphorus, segregate at the grain boundary causing further embrittlement. Depending on the dominance of one of the two opposite processes annealing can lead either to regeneration of the original material properties or eventually cause further damage. This damage is mainly caused by the segregation of phosphorus at the grain boundary. The structure of the irradiated materials differs after annealing from the original; consequently the rate of re-embrittlement can also be different. Most of the annealing and reirradiation tests were performed in research reactors with a fluence of 2-5*10^{23} n/cm^2 (e.g. the IAEA Round Robin on WWER-440 weldment. AEKI participates in this project and the results obtained after irradiation, annealing and reirradiation in the Budapest Research reactor are shown on FIG.4. Several unclad WWER-440/V230 vessels have been annealed, but few experiences exist with the cladded ones. The WWER-440 end of life fluence is one order of magnitude higher than most of the irradiation fluences applied in research. Further studies are needed to obtain more information on the synergism of the different ageing and annealing mechanisms, and to elaborate the utility strategy to use annealing (to find the optimum age for annealing providing the maximum results with low costs.)

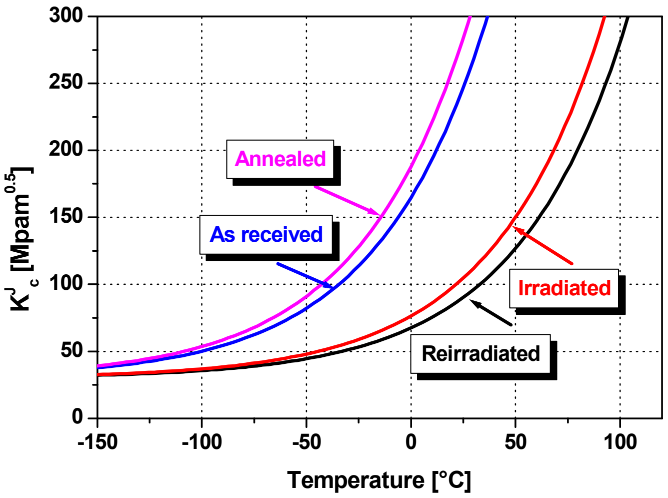

FIG. 4 Effect of irradiation, annealing and reirradiation on the WWER-440 weld. The irradiation and reirradiation fluence are $3*10^{23}$ n/cm^2 E>1MeV.

3. SUMMARY

Several technical solutions are available for lifetime extension of the WWER/V213 RPV-s. Two real life barriers exist at WWER-440 units: the lifetime of the 6 steam generator/units, and the RPV lifetime limited by the PTS scenario. The PTS results are dependent on the material properties. In most cases the material properties (or at least some of them) are not known precisely, so over conservative estimations are used for the calculations. To extend the PTS lifetime elastic-plastic evaluation of the clad behaviour, use of Master Curve, and in many cases RPV annealing are the realistic and economic solutions. All of these developments require further material research. The projected benefit of the started studies is such that the RPV-s of the WWER440/V213 units will not be the life-limiting components of the plants.

REFERENCES

[1] IAEA-EBP-WWER-08. Guidelines on Pressurized Thermal Shock Analysis for WWER Nuclear Power Plants. 1997

[2] Fekete T.; Gillemot, F.; Tatár L.: "Investigation of the Cladding Effect Using Elastic-plastic FE simulation of PTS Phenomena" IAEA Specialist Meeting "Methology for Pressurized Thermal Shock Evaluation" IWG-LMNPP-00/1 pp151-161, Rockville, Maryland USA 18-19 July 2000

[3] F. Gillemot, G. Uri, T. Fekete: "Special Features of VVER-440 Reactors at PTS Analysis". IAEA Specialist Meeting "Methodology for Pressurized Thermal Shock Evaluation" IWG-LMNPP-00/1 pp.151-161. Rockville, Maryland USA 18-20 July. 2000

[4] F. Gillemot: "Life time management of the nuclear power plant equipment" 5th Nuclear Technology Symposium. Paks, October 4-6. 2000

[5] ASTM Standard E-1921-2002. „*Standard Test Method for Determination of Reference Temperature, T_o, for Ferritic Steels in the Transition Range".*

[6] PNAE-G-002-86. Russian code.

METHODOLOGIES AND TECHNOLOGIES FOR LIFE ASSESSMENT AND MANAGEMENT OF COOLANT CHANNELS OF INDIAN PRESSURISED HEAVY WATER REACTORS

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Abstract

Zirconium alloy pressure tubes of Indian Pressurised Heavy Water Reactors (PHWRs) undergo irradiation enhanced creep, growth and hydriding. The calandria tubes also undergo axial creep and growth under the influence of neutron irradiation. The irradiation induced creep-growth manifests. as axial elongation, diametrical expansion and sagging of the pressure tube. Another degradation mechanism, hydriding, can lead to failure of pressure tubes through hydride blister formation at a cold spot, hydrogen embrittlement and delayed hydride cracking (DHC). Considering these degradation mechanisms a set of analytical methodologies and associated computer codes along with various inspection and diagnostic technologies have been developed to facilitate safe management of ageing of coolant channels of Indian PHWRs. This paper dwells briefly upon the aspect of development and application of the above mentioned degradation models and, tools and technologies for diagnosis, inspection and life extension of pertaining to ageing management of coolant channels of Indian PHWRs.

1. INTRODUCTION

Zirconium alloy coolant channels are central to the design of Indian Pressurised Heavy Water Reactors (PHWRs) and form the individual pressure boundaries. These coolant channels consist of horizontal pressure tubes made of zirconium alloys, which are separated from cold calandria tubes using garter spring spacers. High temperature heavy water coolant flows through the pressure tube, which supports the fuel bundles. Figure 1 shows a typical coolant channel in a PHWR.

These pressure tubes are subjected to several life limiting degradation mechanisms like creep and growth, hydrogen pick-up, reduction in fracture toughness and delayed hydride cracking phenomena because of their operation under high temperature, high stress and high fast neutron flux environment. Considering the early onset of these degradation mechanisms in Zircaloy-2 pressure tubes used in the early generation of Indian PHWRs, the life management of these coolant channels becomes a challenging task, involving multidisciplinary R & D efforts in areas like analytical modelling of degradation mechanisms, evolution of methodologies for assessment of fitness for service and, tools and techniques for remote on line monitoring of integrity, maintenance and replacement.

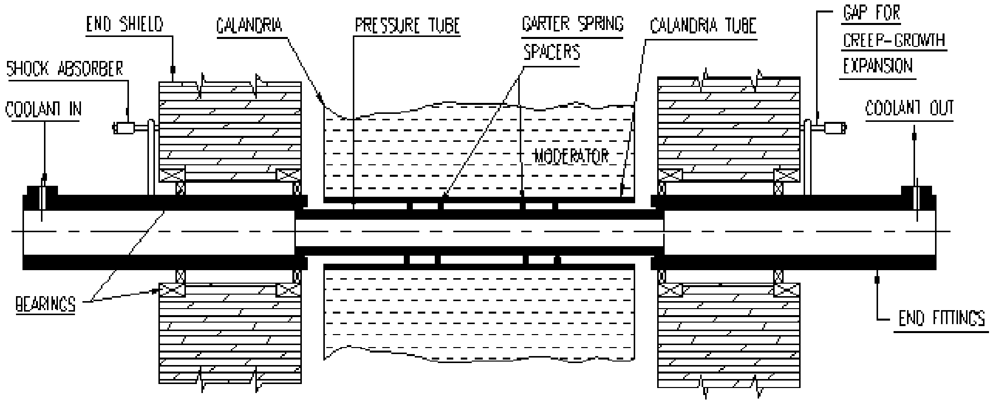

FIG. 1 Schematic of PHWR Coolant Channel

The degradation mechanisms have been modelled and incorporated into specially developed computer codes, such as SCAPCA for irradiation induced creep & growth deformation modelling, HYCON for hydrogen pick-up modelling, BLIST for hydrogen diffusion, blister nucleation and growth modelling and CEAL for assessment of leak before break behaviour. These codes have been validated with respect to the results of in-service inspection and post irradiation examination. Development of analytical models has helped in evolution of more refined methodologies for assessing the safe residual life of coolant channel. Information gathered from various experiments simulating the degradation mechanisms, results of post-irradiation examination of the coolant channels and various research publications in international journals formed the bases for the above safety methodologies. Today, the analytical models together with the safety evaluation methodologies have become an important tool for assessing fitness for service of individual pressure tube in different reactors.

Apart from developing the above analytical tools & methodologies for safety evaluation, several systems and tools have been designed and developed to cater for the activities like remote inspection, online integrity monitoring and life extension etc., of our life management programme of coolant channel. Systems developed for remote inspection of the coolant channel are Dry Visual Inspection System (DRYVIS), BARC Inspection System(BARCIS) and Hydraulic Remote Inside diameter Measurement (HYRIM) System. A non-destructive diagnostic technique based on the principle of vibration measurement has been developed to identify the contact between the pressure tube and calandria tube. This technique called 'Non Intrusive Vibration Diagnostic Technique (NIVDT)' is being used as a tool to screen the vulnerable pressure tubes so as to reduce the inspection load.

Sliver Sample Scraping Tool (SSST) has been developed to monitor the hydrogen ingres rate in an operating channel by measuring hydrogen in the sliver samples obtained by in-situ scraping of the channel. At present, scraping operation can be carried out in both the dry and the wet channel of the operating reactor. Hydrogen measurement carried out in the sliver samples obtained at regular interval gives not only the estimate of cumulative degradation taken place due to hydrogen ingres but also helps in improving the predicting ability of the analytical model for hydrogen ingres.

Extension of service life of a coolant channel is the end objective of the programme for the life management of coolant channel of Indian PHWRs. In the fresh reactor of earlier design, loose fit garter springs shift from their design locations during hot commissioning. Since the shift of garter spring from the design location is undesirable from the bending creep point of view, they need to be relocated to the extent possible. Mechanical Flexing Tool (MFT) has been developed to carry out the above task. To accomplish the similar task in an operating channel, Integrated Garter Spring Repositioning System (INGRES) has been developed.

With all these analytical models and methodologies on one hand and tools and techniques for inspection, online integrity monitoring and life extension on the other, the programme for life management of the coolant channels of Indian PHWRs has been successfully implemented

This paper gives the highlights of the above mentioned different R & D activities carried out for life management of coolant channels of PHWRs which have progressed hand-in-hand with the operation and inspection experience.

2. DEGRADATION MODELS AND THE ASSOCIATED COMPUTER CODES TO MONITOR FITNESS FOR SERVICE OF COOLANT CHANNELS

A number of computer codes have been developed to study the life limiting degradation mechanisms pertaining to coolant channel. These codes have been validated using data from coolant channels of Indian PHWRs and are being used regularly for taking safety related decisions relevant to the components of coolant channel assembly. The following paragraphs describe in brief the capabilities, validation and utilisation of these computer codes.

2.1 Computer Code for Static and Creep Analysis for Pressure tube and Calandria tube Assembly (SCAPCA)

A computer code SCAPCA has been developed in BARC to simulate the creep-growth behaviour of the coolant channel assembly. This code estimates the dimensional changes in coolant channels and their creep-growth limited life on the basis of channel specific design, operation, environment and material inputs. While the mathematical formulation and other relevant details are covered in ref. [1], following gives some of the important capabilities of SCAPCA:

a) This code has the capability to simulate different operating conditions of coolant channels such as channel operation under different power levels, quarantine mode of operation, ISI condition as well as capability to model loading imparted during INGRES operation.

b) Provision is there in the code to account for externally acting axial forces varying with time which may be present due to feeder stiffness, improper settings of creep gap margins at shock absorber, jamming of end-fittings in the bearing sleeves, growth arrest of calandria tube etc.

c) This code has an important ability to model the in-service shift of GS spacers. The code is also capable for estimating the spread of contact between contacting pressure tube and calandria tube by introducing pseudo support of very small height in the expected region of contact.

The module of SCAPCA that performs the static/elastic analysis has been validated using mechanics of structures approach for simple cases and using finite element method for complex cases. The module of SCAPCA that performs the creep-sag analysis was initially validated against data available for CANDU reactors. Subsequent to the availability of information from the in-service inspection (ISI) of coolant channels of Indian PHWRs, this module together with the creep laws have also been validated for Indian PHWRs. The SCAPCA predicted pressure tube calandria tube gap profile has been compared with the actual measured gap profile during ISI of ~324 Nos. of coolant channels of Indian PHWRs. FIG. 2 shows the comparison between SCAPCA estimated gap under ISI condition and gap observed during ISI. In 94% of the cases, the SCAPCA predictions have been found to be in good agreement with the observed gap profiles and in 88% cases, SCAPCA predictions have been conservative for channels having contact less than 10 FPYs.

In addition to its conventional role of estimating the creep-growth limited life of coolant channels, SCAPCA has also been extensively used to model the garter spring un-pinching behaviour of the garter spring repositioning system INGRES. The domain of relevance of the un-pinching module of INGRES can be assessed for channels selected for garter spring repositioning. This code predicts gap profile created between sagged pressure tube and calandria tube during un-pinching operation. FIG.2 also shows the gap created at target GS location by pressure tube flexing during INGRES operation. This gives an estimate of the distance by which the garter spring can be relocated and also about the extent to which creep-growth limited life can be improved

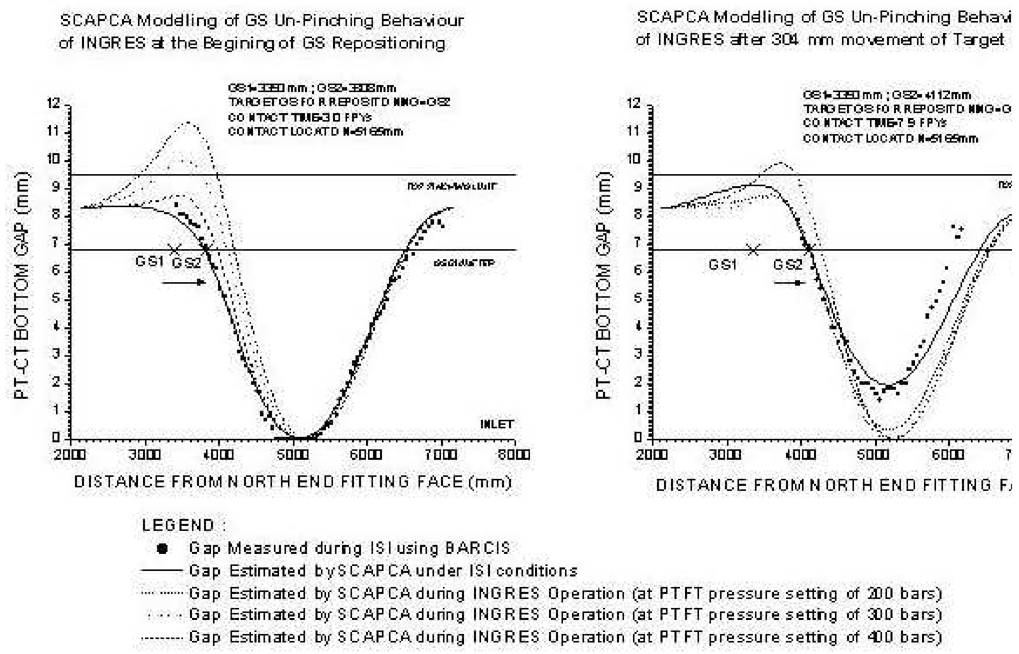

FIG.2 SCAPCA estimated pressure tube calandria tube bottom gap under various loading conditions

Using this methodology, SCAPCA has provided support during the repositioning work carried out at Indian PHWRs at Kalpakkam, Chennai.

2.2 Computer Code for Estimation of Hydrogen Concentration (HYCON) [2]

Apart from creep and growth, another life limiting mechanism in the case of Zircaloy-2 pressure tubes is hydriding. During their residence time in reactor, pressure tubes undergo corrosion with high temperature heavy water coolant flowing through them. An oxide layer is formed on the inside surface of the tube, thickness of which increases with reactor operation. Hydrogen is also evolved during the corrosion process, a part of which is absorbed by the pressure tube material. A computer code HYCON-95 was developed earlier to predict the hydrogen pick-up in Zircaloy-2 pressure tubes. This code has been used extensively for life management of all the Indian PHWRs with Zircaloy-2 pressure tubes.

The hydrogen pick up model used in HYCON-95 has been upgraded using the post irradiation examination (PIE) data on a number of Zircaloy-2 pressure tubes removed at different FPYs from Indian PHWRs. Using these data, new semi-empirical correlations have been developed for estimation of oxide thickness and hydrogen pick up. These correlations can be used to estimate oxide thickness and hydrogen pick up in the post transition regime of corrosion process as well. This revised version, christened as

'HYCON-99' has been used extensively during the recent garter spring repositioning campaign in unit-1 of Indian PHWR at Kalpakkam, Chennai.

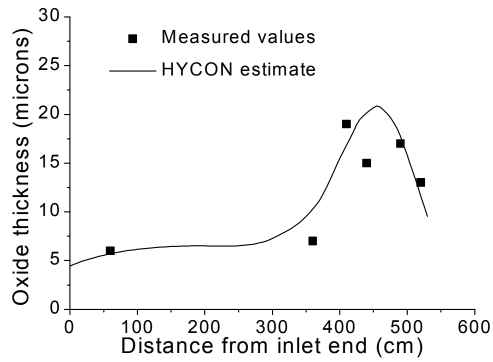

FIG.3. Comparison between measured oxide thickness and HYCON estimate for J07 channel of Unit-2 of Indian PHWR at RAJSTHAN (Now re-tubed)

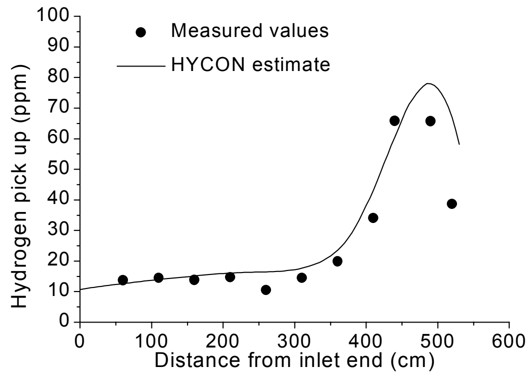

*FIG. 4. Comparison between measured hydrogen pick-up and HYCON estimate J07 channel of Unit-2 of Indian PHWR at RAJSTHAN (Now re-tubed)***.**

The comparison of estimated oxide thickness and hydrogen pick-up profiles with measured values for a typical high flux channel are shown in above FIGs. 3 and 4.

2.3 Computer Code for Estimation of Blister Depth (BLIST) [3]

When the hot pressure tube comes in contact with cold calandria tube, due to creep sag, a temperature gradient is established in and around small region, called contact spot, on the outside surface of pressure tube. By diffusion process, hydrogen present in the tube migrates to the cold regions and precipitates there as hydrides. These hydrides, when accumulated in a small region, lead to formation of hydride blister.

In order to assess the blister depth, mathematical models have been developed for the process of diffusion of hydrogen under thermal gradient. To study the phenomena of hydrogen diffusion and resulting blister growth for circular shape of contact between PT and CT, two computer codes BLIST1D and BLIST2D have been developed on the basis of axi-symmetric models using the finite difference numerical technique. The BLIST1D uses 1-D equivalent of the actual temperature distribution where as the BLIST2D uses the actual 2-D temperature distribution. While BLIST1D is faster and can be used to screen the entire core in short time, BLIST2D gives more precise estimate of blister depth. BLIST1D code has so far been used to assess the safety of more than 1000 un-inspected and inspected channels in all the Indian PHWRs having Zircaloy-2 pressure tubes. The assessment has so far been quite satisfactory in the sense that non of the assessed channels have gone to the other side of safety envelope drawn by this code. BLIST2D code has been validated with respect to the measured blister of J07 channel in RAPS-2, as shown in FIG. 5. This code is being used for more precise estimate of blister depth in channels declared unsafe by BLIST1D.

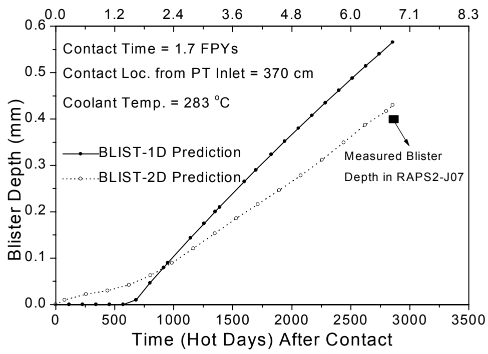

FIG.5. Comparison between measured blister depth and estimations using BLIST codes.

These computer codes have been used extensively during several campaigns of ISI and or INGRES carried out in different units of Indian PHWRs

2.4 Computer Code for Estimation of Assurance of Leak Before Break (CEAL) [4]

The fitness of pressure tube for service requires a demonstration of leak before break (LBB) capability. For pressure tubes of PHWRs, the LBB criterion is defined by the following expression;

$Ao + Lo + 2\ Vc * t < A_{max}$

Where,
Ao = through wall axial crack length which leads to minimum detectable leak rate
Lo = Length of the crack on the inside surface of the tube just as the crack front reaches the outside surface of the tube
Vc = delayed hydride crack propagation velocity
t = operator response time
A_{max} = maximum allowable crack length

To achieve LBB, 'A_{max}' must be less than the minimum of the values of critical crack length of the pressure tube under all possible conditions of pressure and temperature during normal operation, hot shut down, depressurisation and start-up.

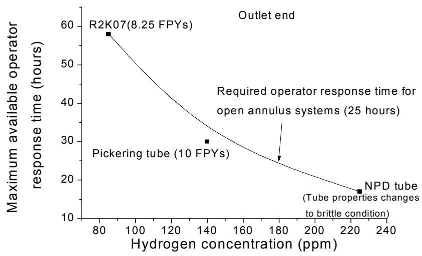

FIG.6. Estimated variation of operator response time with hydrogen concentration.

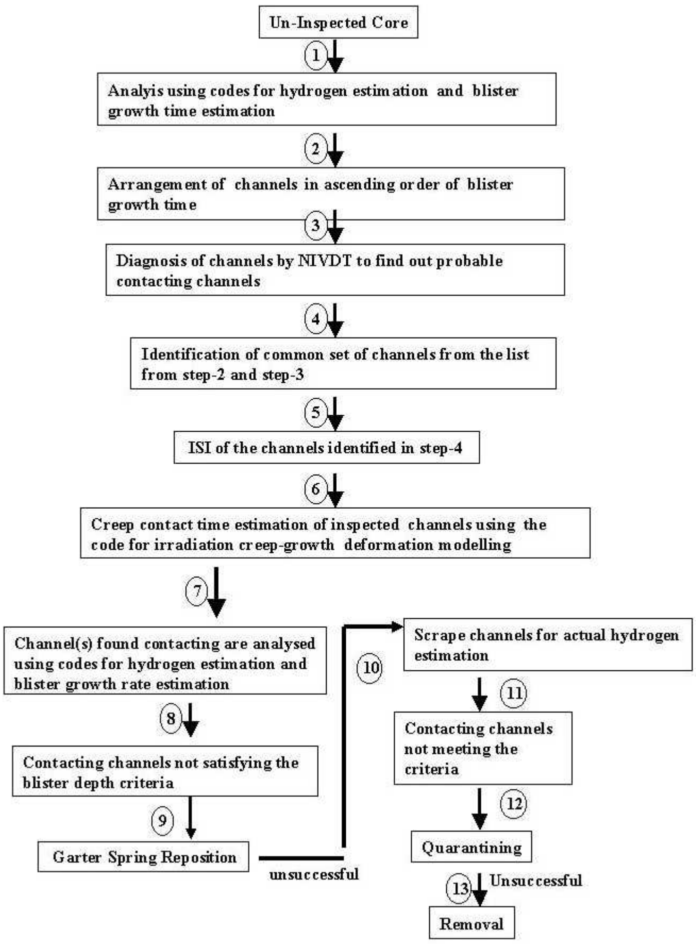

FIG.7. Coolant Channel Life Management Strategy

Using the aforementioned criterion for LBB, a computer code CEAL (Code for the Estimation of Assurance of LBB) has been developed. FIG.6 shows the estimated variation of operator response time with hydrogen concentration for Zircaloy-2 pressure tubes.

3. LIFE MANAGEMENT STRATEGY FOR THE ZIRCALOY-2 PRESSURE TUBES

Since the reactor core in the early generation of reactors is having a few contacting pressure tubes, a comprehensive life management strategy evolved with the operating experience over a period of time , as described below, is being followed for taking safety related decisions with regard to them.

The life management strategy, as described above, requires application of the analytical models and safety methodologies along with the tools and techniques for health diagnostics, inspection and life extension. These have been described in the subsequent paragraphs.

4. TOOLS AND TECHNIQUES FOR DIAGNOSIS AND INSPECTION

4.1 Identification of Contacting Channels using Non-Intrusive Vibration Diagnostic Technique (NIVDT)

A technique based on measurement of vibration at both ends of the coolant channel, caused by excitation from an electro-dynamic shaker mounted at one end, has been developed, and is being routinely used, to identify pressure tubes in contact with the calandria tubes. This procedure enables screening the entire reactor core in 3 to 4 days time. This technique has been used as a screening method to identify channels which are likely to be in contact so that further ISI efforts and subsequent garter spring relocation efforts, if necessary, could be applied more selectively on the short list prepared on the basis of application of NIVDT. Non intrusive screening technique for identification of coolant channels with PT/CT contact was carried out in all the units of early generation of reactors. Several refinements have been incorporated based on the feed back from channel inspection. Figure 8 shows the NIVDT being used at site.

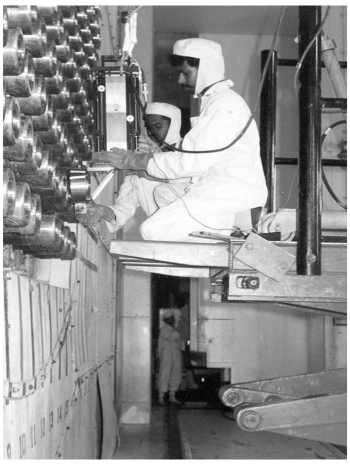

FIG.8. NIVDT in operation

It consumes less man-rem and also takes less time to examine the entire core. Both these features have helped it grow as an important diagnostic tool. The technique has been upgraded in order to be used in Indian reactors of latest design. Artificial neural network is being trained to discriminate pattern of contacting channels. This is expected to further improve the strike rate of identifying contacting channels.

4.2 Tools for Coolant Channel Inspection

In order to carry out ISI of coolant channels, identified to be in contact using NIVDT and not meeting the criteria for acceptable blister depth, a special inspection system called BARCIS (BARC Channel Inspection System) has been developed by incorporating several ultrasonic and eddy current based sensors for carrying out measurements [6]. This inspection using this tool is an important part of the coolant channels life management programme being followed in India.

4.3 Tool for Dry Channel Visual Inspection

A tool for dry channel visual inspection called DRYVIS (Dry Channel Visual Inspection System) has been developed based on a pneumatically operated tube walker, which can be remotely made to crawl in the desired direction within a tubular component. The device can carry any transducer for carrying out inspection. The DRYVIS system comprises a tube walker, a radiation resistance video camera, a separate illumination head and a grating for sizing of indications.

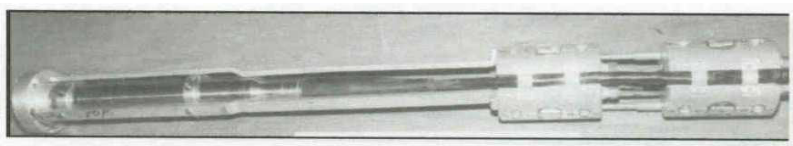

FIG.9 Dry Visual Channel Inspection System (DRYVIS)

This system has been used to carry out visual inspection of the drained and defuelled pressure tubes of RAPS-2. Fig. 9, given above, shows the photograph of DRYVIS.

4.4 Hydraulic Remote Inside Diameter Measuring (HYRIM) Tool

This is a hydraulically operated inside diameter (ID) measurement system suitable for applications where bore to be measured is inaccessible due to complicated assembly and/or severe operating environments. It is designed, primarily, for measurement of ID of operating pressure tubes of Indian PHWR under wet channel condition. It has measurable ID range of 82 to 94 mm and accuracy of the system is of the order of +/- 0.05 mm.

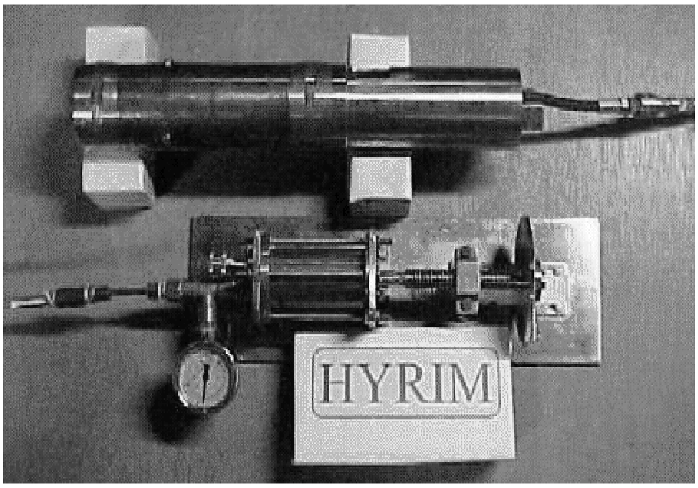

FIG. 10 Hydraulic Remote Inside Diameter Measuring Tool

The system consists of an inspection head and a measuring station hydraulically coupled by a long hose. The inspection head is placed inside the bore of the pipe to be measured and ID measurement is taken at the measuring station which is installed at a certain distance. The measuring station is having a hydraulic cylinders. Piston of this cylinder is connected to a hand wheel (Dial) through a lead screw. As the dial is rotated, the piston at measuring station moves forward, and produces an equivalent displacement of the radial legs at the inspection head. Dial is rotated until the radial legs touches the ID of the tube being measured, Thus, the rotation of dial gives direct reading of ID of tube

4.5 Tool For Removal Of Sliver Scrape Samples From Pressure Tubes

Sliver **S**ample **S**craping **T**ool (SSST) has been developed for obtaining in situ scrape samples in a pressure tube of an operating PHWR. Two different versions of the tool exist for carrying out the above operation in dry and wet pressure tubes respectively. The version of tool used in wet channel has been named as **WE**t **S**craping **T**ool (WEST).

The SSST incorporates mainly the Scraping tool and hydraulic, pneumatic and mechanical sub-systems. This technique is used remotely to obtain metal samples from a desired axial location from the bore of the pressure tube at 12’O clock position. The system was successfully used for the first time in Unit-1 of Madras Atomic Power Station during

April 1998 to remove five samples from two pressure tubes. Later on, the system was modified to reduce the man-rem consumption and down time of reactor by incorporating remotisation of tool. The modified SSST has been used in the same reactor for obtaining 36 sliver samples from eight pressure tubes.

In contrast to the SSST, which uses its own dedicated systems for axial positioning and feeding of cutting tool, the WEST is operated using fuelling machine both for axial positioning and feeding of cutting tool. FIG.11 shows the photograph of Sliver Sample Scraping Tool.

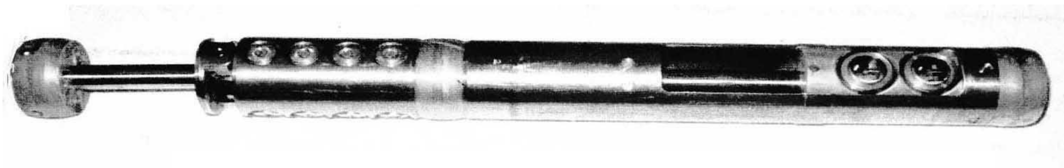

FIG. 11 Sliver Sample Scraping Tool

5. LIFE EXTNENSION OF COOLANT CHANNELS USING GARTER SPRING REPOSITIONING

Systems have been developed for extending service life of coolant channels by carrying out the task of repositioning of garter springs in new as well as operating reactors.

5.1 Garter Spring Repositioning in New Reactors

The garter springs of loose fit design used up to Unit-1 of Kakrapara Atomic Power Station at Gujrat undergo shift from their design locations after initial hot conditioning and therefore require re-location. In-house developed Mechanical Flexing Technique (MFT) helps in accomplishing this task. It employs a specially developed hydraulically actuated Pressure Tube Flexing Tool (PTFT) which, by causing gentle cyclic flexing of the pressure tube, enables the displacement of garter spring. A personal computer based computer programme RESET has also been developed to simulate the operation of PTFT.

5.2 Garter Spring Repositioning in Operating Reactors

For repositioning of garter springs in operating reactors, a multi-functional system called INGRES (**IN**tegrated **G**arter spring **RE**positioning **S**ystem) has been developed along with its associated software. The photograph of the system is shown in FIG. 11. Different modules of the system are capable of performing following functions.

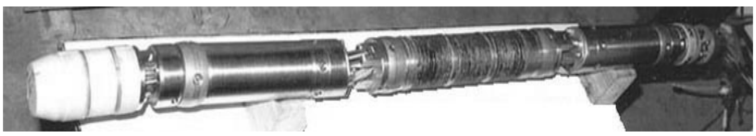

FIG. 12 INGRES (Version 2S) Tool

a) Eddy current principle based Garter Spring Detection Probe (GSDP) detects axial position of garter spring spacers.

b) Eddy current principle based Concentricity Detection Probe (CDP) checks PT-CT concentricity.

c) Hydraulically operated Pressure Tube Flexing Tool (PTFT) corrects the PT-CT eccentricity at garter spring position.

d) Linear induction motor principle based Garter Spring Relocation Device (GSRD) relocates garter springs.

The 1st and 2nd versions of INGRES tool called as INGRES-1S and INGRES-2S respectively are suited for operation in dry channel only. In INGRES-1S, the integral assembly of GSDP, CDP and GSRD modules is required to be inserted from one end of a defuelled and drained coolant channel while the PTFT is to be inserted from the other end. This system had been used to extend the life of 6 coolant channels in the Unit-2 of Madras Atomic Power Station at Kalpakkam during July-August 1995.

INGRES-2S incorporated following modifications over its predecessor.

a) Integration of the electromagnetic device (for providing movement to garter springs) and the hydraulic bending module (to un-pinch garter spring) into a single assembly helped its usage from single vault only.

b) Higher mechanical advantage of modified design of bending module helped in achieving un-pinching of garter spring at a comparatively smaller force.

c) Computer based control for carrier of the system was incorporated.

d) Operating parameters were modified so as to cause a large movement of garter spring during each trial operation.

e) Modifications in the insulation etc., were incorporated to improve system reliability.

f) For operation of bendng module, D_2O was used as hydraulic fluid.

This system is operated through computer interface located in a special control room outside the reactor building. This system was used in Unit-1 of Madras Atomic Power Station at Kalpakkam during 1998 for repositioning of garter springs in five coolant channels and recently during March 2000 in the same Unit to extend the life of 28 coolant channels by around 1.5 FPYs .

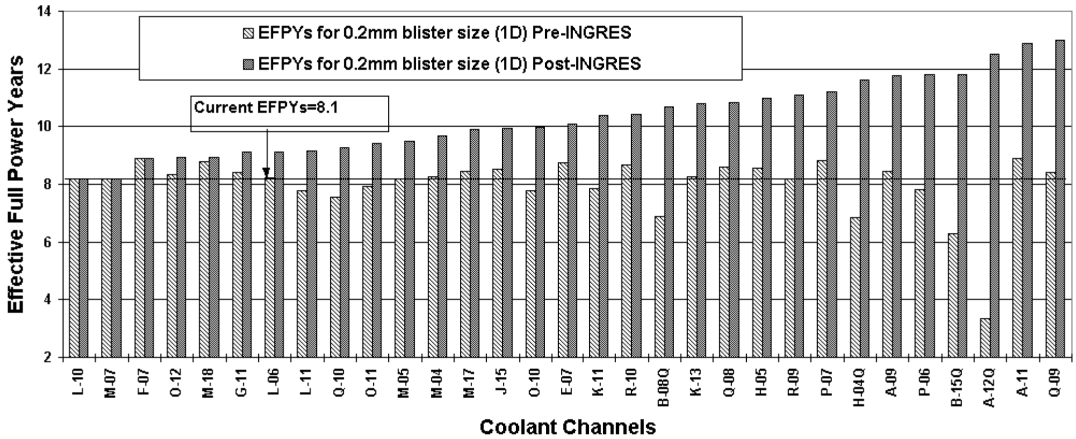

FIG.13. Performance of INGRES at Unit –1 of Madras Atomic Power Station

In its latest incarnation as INGRES-3S, this tool is capable of carrying out the relocation of garter spring spacers in the wet channel. It is also fitted with in-house developed eddy current based transducer for accurate detection of garter springs. This transducer, called 'Garter Spring Detection Probe (GSDP)' is capable of operating at high temperature (100 ^{o} C) under radiation environment [7].

6. CONCLUSIONS

India has now acquired strength in all aspects of life management of PHWR coolant channels. The task has been quite challenging considering the early onset of the degradation mechanisms in the Zircaloy-2 pressure tubes in the early generation of Indian PHWRs. The R & D activities encompassing development of analytical models for degradation mechanisms and their validation, development of safety methodologies and development of life management technologies, have progressed hand-in-hand with the operation and inspection experience. The solutions have been delivered in time to enable taking mature as well as conservative measures in time.

REFERENCES

[1] Sharma Avaneesh, Malshe U.D.& Sinha R.K., "Analytical Modelling for Simulation of Creep and Growth Behaviour of PHWR Coolant Channel"; proceedings of workshop on Creep, Fatigue and Creep-Fatigue Interaction, Kalpakkam, India, Feb 15-17, 1995.

[2] Madhusoodanan K., Sinha S.K. and Sinha R.K., "A Computer Code for Estimation of Hydrogen Pick up and Oxide Thickness in Zircaloy-2 Pressure Tubes", International Symposium on Materials Ageing and Life Management, Sarabhai Auditorium, IGCAR, Kalpakkam, Tamil Nadu, India, October 3-6, 2000.

[3] Sinha S.K. and Sinha R.K., "Numerical Modelling of Hydride Blister Using Finite Difference Technique", Technical Discussion meet on Life Management Methodologies for Coolant Channels of Indian PHWRs, Anushaktinagar, Mumbai, August 2, 1999.

[4] Madhusoodanan K., Rupani B.B. and Sinha R.K., "Comparative Assessment of Zirconium Alloy Pressure Tubes for Leak Before Break", Technical Discussion meet on Life Management Methodologies for Coolant Channels of Indian PHWRs, Anushaktinagar, Mumbai, August 2, 1999.

[5] Sinha S.K., Sharma Avaneesh, Madhusoodanan K., Sinha R.K., and Kakodkar Anil, “Optimum Design of Tight Fit Garter Spring Spacers and Design Evaluation of Existing Spacer”, Internal Report No. RED/RCCS/AS/GSD-R0/891102
[6] Sing Manjit, “BARCIS (BARC Channel Inspection System*)*”, BARC Newsletter, no.189, October 1999, BARC Founder’s Day Special Issue.
[7] Shyam T.V., Srivastava A.K, Apraj S. K.,”Design and Development of Eddy Current Systems to Locate Garter Spring Spacers in Highly Radioactive Coolant Channels of Indian PHWRs Type of Nuclear Reactors”, paper selected for S.N. Seshadri Memorial Award for year 1998.

IN-SITU RAMAN SPECTROSCOPIC INVESTIGATION OF ALLOY 600 IN SIMULATED PWR WATER

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Abstract

In-situ Raman spectroscopy has been used in order to characterize the surface oxide film of nickel-base alloy (alloy 600) in simulated primary water conditions of pressurized water reactors. Specimens were exposed to deaerated and deionized water with 1000 ppm boron, 2 ppm lithium and 30 cm^3(STP)/kg hydrogen concentration at a pressure of 18 Mpa and temperatures ranging up to 320°C in an optically accessible cell. For measurements of ex-situ spectra in the air, the NiO and $NiFe_2O_4$ powders were used to obtain reference spectra for aqueous corrosion of alloy 600 at high temperature and high pressure water environment. In-situ Raman spectra were collected at different temperatures as the specimens were heated and then cooled. During heating, alpha chromium oxide hydroxide(α-CrOOH) phase was detected on the specimen surface at relatively low temperature ranging from 220°C to 250°C. Also, Cr^{VI} or crystalline Cr^{III}/Cr^{VI} species identified with their tentative spectra were observed. At higher temperature, nickel chromium spinel ($NiCr_2O_4$) features are observed.

1. INTRODUCTION

Intergranular stress corrosion cracking (IGSCC) is one of major degradation modes that occur in Ni-base structural materials for steam generators or nozzles in nuclear power plants. The IGSCC has been caused by specific combinations of materials and environmental conditions. [1] The IGSCC of alloy 600 in PWR primary water has been extensively studied to establish theoretical and empirical models [2-5] for predicting the crack initiation time in field components. Even though there is no general agreement on the origin of IGSCC, one common assumption of these approaches is that the damage to the alloy substrate can be related to some transport or repair properties of overlaying oxide film. Recently, it was shown that the oxide film structure, intergranular stress corrosion crack initiation time and crack growth rate were systematically varied as a function of hydrogen partial pressure in high temperature water. [6] This is the evidence that the oxide film on the surface of the alloy 600 in the primary water plays a key role in the process of IGSCC.

This study is aimed to characterize the oxide film by the in-situ Raman spectroscopy under PWR primary water conditions to understand how the oxide film can participate in the cracking process as a key variable. This can be done by analyzing the oxide film properties obtained in various conditions leading to different IGSCC susceptibility.

The in-situ Raman spectroscopic information has been obtained through a hermetically-sealed window onto the oxide film of metal surfaces in simulated primary water. The system is capable of examining the effect of temperature, hydrogen partial pressure, and impurity concentration on chemistry and kinetics of film formation at temperature of up to 320°C and pressure of up to 18 MPa.

Table 1. Nominal composition of alloy 600 for PWR nozzles.

Element	C	Mn	Fe	S	Si	Cu	Ni	Cr	Al	Ti	Cb	P	B	N
Comp.	0.06	0.26	8.31	0.001	0.3	0.12	75.12	15.25	0.16	0.36	0.04	0.009	0.002	0.001

Table 2. Temperature and alloy 600 exposure time prior to in-situ Raman spectra measurements.

Temperature (°C)	Total exposure time prior to first measurement at	Hold time at each temperature prior to first
20	8	8
220	26	18
250	30	2
290	33.5	3
320	37	3

2. EXPERIMENTAL

2.1. Materials

The alloy 600 used in this study for analysis of the oxide films have been used as a reactor pressure vessel (RPV) head penetration materials for control rod drive mechanisms in PWRs. Test material with nominal composition given in Table 1 was solution-annealed at 1050^{o}C for 2 hours and then water-cooled. The specimens is a round disk with 7 mm diameter and 1 mm thickness. All the specimens were mechanically polished down to 1μm Al_2O_3 powder, then rinsed with ethanol, and finally with deionized water prior to exposure.

2.2. Environment

For the surface oxide formation, the alloy 600 specimen was placed in an optical cell and exposed to a deionized and deaerated aqueous environment. The same chemical condition of water system as PWR primary water was used. Water with a resistivity of 18 MΩ-cm was mixed with additional chemicals of 1000 ppm boron in the form of boric acid and 2 ppm lithium in the form of lithium hydroxide. Hydrogen gas with 30 cm^3(STP)/kg hydrogen concentration was injected into the water and it made a dissolved hydrogen concentration of 2.68 ppm in this study [7]. The operating pressure was about 18 Mpa and the temperature was varied from room temperature to 350^{o}C. Temperatures at which Raman spectra were collected with the remained time length are summarized in Table 2.

2.3. Apparatus

The optical cell for in-situ observation at high temperature and pressure condition was constructed with a custom-made 1 liter-capacity autoclave. To reduce corrosion products from the flow system at high temperature, the cell was made of alloy 690. For the same reason, the head of a water charging pump was made of titanium and other components exposed to high temperature water including compression fittings and tubings were made of alloy 600. The cell was machined with two penetrations in the wall of cylinder to place optical windows for the access of Raman spectroscopy. The window assembly, composed of 4.5 mm thick sapphire with gaskets to insulate test water, was designed after Maslar and Hurst's work [8-

10]. In this study, gaskets were made of alloy 718 electroplated with gold to 25 μm thickness. Figure 1 shows the assembly drawing of the sapphire window system used in this study for in-situ Raman spectroscopic investigation at high temperature and high pressure aqueous environment. The alloy 600 specimen was held in the recess of an oxidized zirconium metal by a washer made of alloy 718 with the gasket of an oxidized zirconium sheet so as to electrically insulate the specimen for the cell.

The Raman spectroscopy system is consisted of an excitation source, a spectrometer and optical components including mirrors and filters. Figure 2 shows the layout of optical system with near backscattering geometry including optical cell and water chemistry loop used in this study. An argonion laser operating at 514.5 nm was used for excitation of in-situ Raman spectra. A holographic laser bandpass filter was used to remove unwanted plasma lines. Laser radiation polarized perpendicular to the plane of incidence was focused onto the specimen with cylindrical lens having a 150 mm focal length producing a rectangular spot approximately 0.2×2 mm and a power density of less than 15 W/cm^2 on the specimen. Raman-scattered radiation was collected through the window and collimated with a 50 mm diameter, 500 mm focal length plano-convex lens. A holographic notch filter was used to reduce Rayleigh scattering entering spectrometer. With a 50 mm diameter, 350 mm focal length plano-convex lens, collimated Raman-scattered radiation was coupled into the spectrometer. A 0.55 m imaging spectrometer equipped with a 1200 grove/mm grating blazed at 500nm was used to disperse the light onto a 1250×300 pixel array, back-illuminated, liquid nitrogen cooled, charge coupled device(CCD) camera system. The band width was about $5cm^{-1}$ and the instrumental reproducibility was $\pm 2cm^{-1}$. All spectra were obtained for an integration time of 600 sec or less.

For the measurements of ex-situ spectra in the air, same optical system and excitation source as those of in-situ were used and specimens were placed in the optical cell without heating and water flow. The NiO and $NiFe_2O_4$ powders were used to obtain reference spectra for aqueous corrosion of alloy 600 at high temperature and high pressure water environment. Powder samples were placed in the recess of sample holder, mixed with deionized water, and then dried to be deposited on the sample holder.

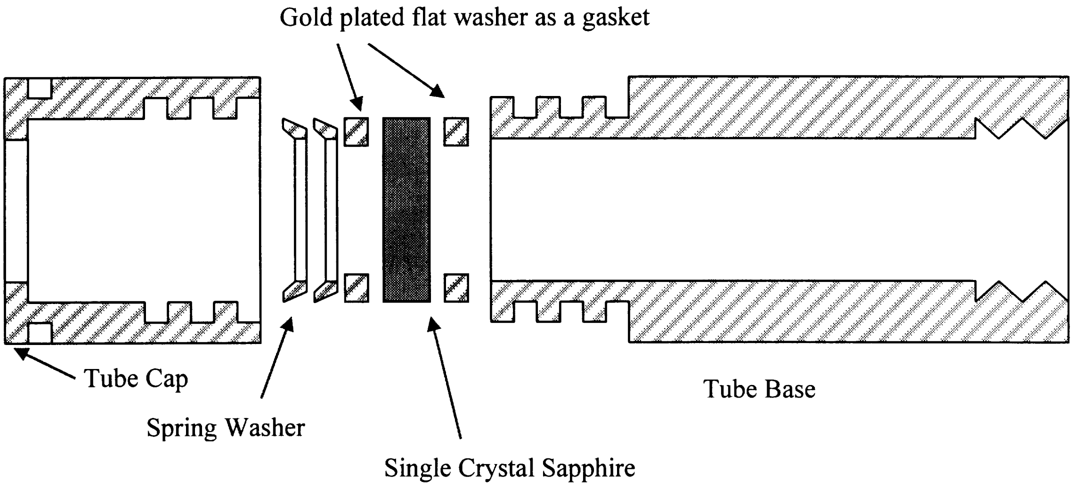

FIG. 1 Assembly drawing of the sapphire window system.

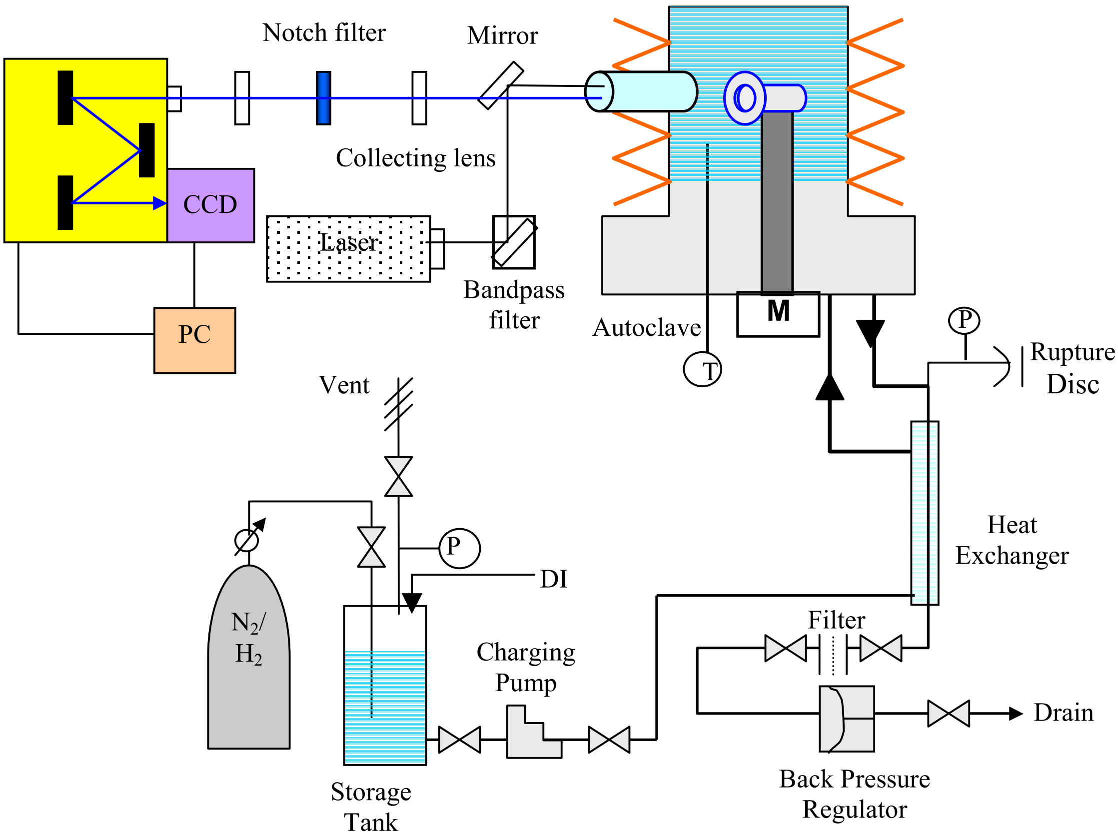

FIG. 2 Layout of in-situ Raman spectroscopic system with optical cell and water chemistry loop developed for this study.

3. RESULTS AND DISCUSSIONS

3.1. Ex-situ characterization

The initial experiment included nickel oxide and spinel, and results were used for verifying the developed system against literature data. The results of experiments with oxide and spinels at room temperature and air environment are shown in Fig. 3 and 4.

Figure 3 shows the ex-situ spectra obtained with NiO powder sample compared with literature data [11]. The respective spectral intensities were scaled for comparison. The NiO powder reference spectrum in this study exhibited features at ca. 400cm^{-1}(weak), 530 cm^{-1}, 725 cm^{-1}, 915 cm^{-1} and 1074 cm^{-1}. This results are well matched with green NiO spectrum in literature. Features at ca. 400 cm^{-1}(weak), 530 cm^{-1} are attributed to disorder-activated, symmetry-forbidden, first-order scattering, and those at 725 cm^{-1}, 915 cm^{-1} and 1074 cm^{-1} are attributed to second-order phonon scattering [12]. A peak at ca 470 cm^{-1} was attributed to the optical lense and that at ca. 1110 cm^{-1} attributed to the background noise from 546.1 nm glow bar.

Figure 4 shows the ex-situ Raman spectra of the $NiFe_2O_4$ powder both in this study and in literature. The wavenumbers of the peaks in the powder spectrum are summarized in Table III with reported nominal Raman wavenumbers in literature. As shown in this result, two most intense peaks were observed at ca. 489 cm^{-1} and 704 cm^{-1} in this experiment and the reference spectrum of $NiFe_2O_4$ powder agree well with those reported in Ref. [13].

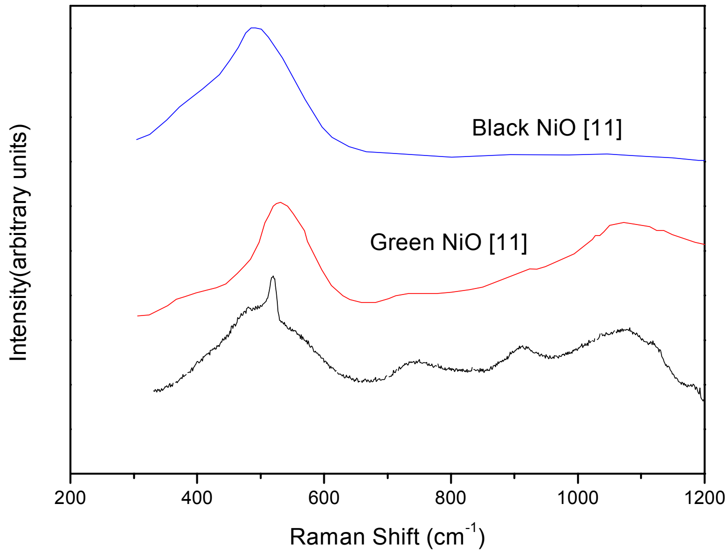

FIG. 3 Ex-situ Raman spectra of the NiO powder and literature data [11].

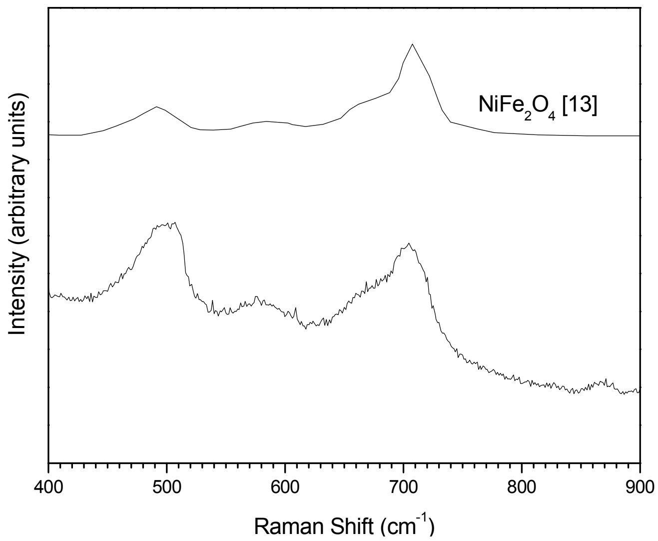

FIG. 4 Ex-situ Raman spectra of the NiO powder and literature data [13].

Table 3. Nominal Raman shift of $NiFe_2O_4$ spinel[a]

$NiFe_2O_4$ (This work)[b]	$NiFe_2O_4$ (Maslar et al. [13])	$NiFe_2O_4$ (Graves et al. [13])
<u>704</u>	<u>705</u>	700
654 sh	655sh	655 sh
595	592	
574	570	579
<u>489</u>	488	<u>490</u>
460 sh	457 sh	483 sh

[a]The most intense peak(s) in each spectrum is underlined

[b] The wavenumber of a well-resolved peak has an associated uncertainty of ±2 cm^{-1}, and a shoulder(^{sh}) ±5 cm^{-1}

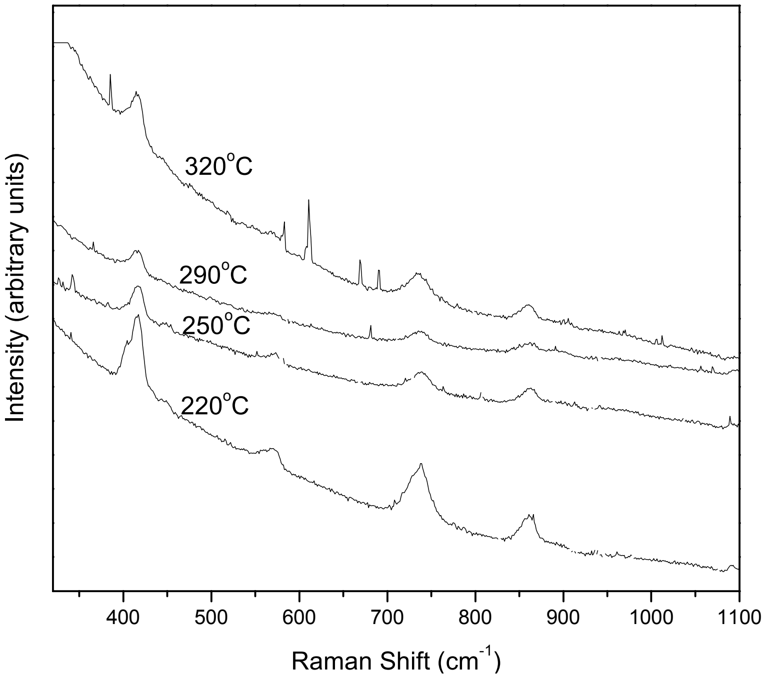

FIG. 5 In-situ Raman spectra of alloy 600 specimen. The spectra are offest on the vertical scale for clarity.

3.2. In-situ characterization

Figure 5 shows the in-situ Raman spectra of alloy 600 specimen obtained while temperature increased. Peaks observed at ca. 417 cm^{-1}, 643 cm^{-1} (weak) in the in-situ Raman spectra are due to the sapphire window. A features in ca. 546-587 cm^{-1} range in the spectrum acquired at 220°C was attributed to α-CrOOH as described in Ref. [10]. The intense peaks are observed in the 840-880 cm^{-1} range. According to the extensive discussion by Maslar et al. [14], this feature would be attributed to Cr^{VI} or crystalline Cr^{III}/Cr^{VI} species assuming a chromium species is responsible for this feature. This explanation can account for the weak signature observed in the 340-350 cm^{-1} range in the spectra of the specimen at 220°C. Chromium oxide features in this wavenumber range are generally attributed to Cr^{VI}–oxygen terminal stretching modes or mixed Cr^{III}/Cr^{VI} oxide vibrational modes. Hydrated surface chromate species have been reported to exhibit a vibrational mode at ca. 865 cm^{-1}. The most intense peak in the aqueous $HCrO_4^-$ Raman spectrum is reported from ca. 880 to 899 cm^{-1}. A number of mixed Cr^{III}/Cr^{VI} oxides have been identified during the thermal decomposition of various chromium-containing materials. Crystalline compounds such as Cr_3O_8, Cr_2O_5, and XCr_3O_8 (X = Na, K, Rb) exhibit their most intense Raman spectral features in the ca. 820-904 cm^{-1} range at room temperature. The most intense Raman spectral feature of an amorphous Cr^{III}/Cr^{VI} compound has been observed at 859 cm^{-1}. All features observed in the spectrum at 250 °C were same as those in 220 °C spectrum.

In the spectra obtained at 290°C and 320°C, features of pure chromium oxide including α-CrOOH, Cr^{VI} and crystalline Cr^{III}/Cr^{VI} compounds become weaker, while nickel chromium spinel ($NiCr_2O_4$) features are firstly observed and become more apparent in subsequent spectra. The features at ca. 682 cm^{-1} and 510 cm^{-1} (weak) were attributed to $NiCr_2O_4$ phase [15]. The $NiCr_2O_4$ is one of thermodynamically stable phases in a reducing aqueous environment with the range of hydrogen overpressure at temperatures of about 300°C.[16]

4. SUMMARY AND CONCLUSION

From the in-situ Raman spectroscopic investigation with a nickel-base alloy (alloy 600) exposed to deaerated and deionized water with 1000 ppm boron, 2 ppm lithium and 30 cm^3(STP)/kg hydrogen concentration at a pressure of 18 Mpa and temperatures ranging up to 320°C in an optically accessible cell, following conclusions are made:

1. The window assembly composed of 4.5mm thick sapphire with gaskets made of alloy 718 with 25 μm thick electroplated gold layer to seal high pressure water provided reliable and robust characteristics at high temperatures and high pressures.
2. During a heating, alpha chromium oxide hydroxide(α-CrOOH) phase was detected on the specimen surface at relatively low temperature ranging from 220°C to 250°C.
3. The intense peaks are observed in the 840-880 cm^{-1} range. According to the extensive discussion by Maslar et al. [14], this feature can be attributed to Cr^{VI} or crystalline Cr^{III}/Cr^{VI} species. The interpretation can account for the weak signature observed in the 340-350 cm^{-1} range in the temperature range of 220°C to 250°C.
4. The features at ca. 682 cm^{-1} and 510 cm^{-1} (weak) attributed to $NiCr_2O_4$ phase were observed at 290°C or higher.

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REFERENCES

[1] J.M Gras, "Stress corrosion cracking of steam generator tubing materials - review and assessment", Parkins Symposium on Fundamentals Aspects of Stress Corrosion Cracking, TMS, 1992, p. 411.

[2] T. Magnin, J-M. Boursier, D. Noel, R. Rios, and F. Vaillant, "Corrosion-deformation interaction during stress corrosion cracking of alloy 600 in primary water", Proceedings of the 6th International Symposium on Environmental Degradation of Materials in Nuclear Power System – Water Reactors, San Diego, TMS, 1993, p.669.

[3] P.M. Scott and M. Le Calvar, "Some possible mechanisms of intergranular stress corrosion cracking of alloy 600 in primary water", Proceedings of the 6th International Symposium on Environmental Degradation of Materials in Nuclear Power System – Water Reactors, San Diego, TMS, 1993, p. 657.

[4] G. Was, T.M. Angeliu, and J.K. Sung, "Deformation and intergranular cracking behavior of Ni-Cr-Fe alloys at high temperature", Proceedings of the "Specialist Meeting on Environmental Degradation of Alloy600", Airlie House, VA, April 1993, EPRI Report TR-104898, 1996, p. 24-1.

[5] M.M. Hall, “Thermally activated low temperature creep and primary water stress corrosion cracking of NiCrFe alloys”, Proceedings of the “Specialist Meeting on Environmental Degradation of Alloy600”, Airlie House, VA, April 1993, EPRI Report TR-104898, 1996, p. 6-1.
[6] C. Soustelle, M. Foucault and P. Combrade, “PWSCC of alloy 600: a parametric study of surface film effects”, Proceedings of the 9th International Symposium on Environmental Degradation of Materials in Nuclear Power System – Water Reactors, TMS, 1999, p. 105.
[7] D.M. Himmelblau, “Solubilities of inert gases in water 0 °C to near the critical point of water”, J. Chem. Eng. Data **5** 1 (1960) 10.
[8] W.S. Hurst, M.S. Hodes, W.J. Bowers Jr., V.E. Bean, J.E. Maslar, and K.A. Smith, “Optical flow cell and apparatus for solubility, salt deposition and Raman spectroscopic studies in aqueous solutions near the water critical point”, Journal of Supercritical Fluids **22** (2002) 151.
[9] J.E. Maslar, W.S. Hurst, W.J. Bowers Jr., J.H. Hendricks, and M.I. Aquino, “In situ Raman spectroscopic investigation of aqueous iron corrosion at elevated temperatures and Pressures”, Journal of Electrochemical Society **147** (2000) 2532.
[10] J.E. Maslar, W.S. Hurst, W.J. Bowers Jr., J.H. Hendricks, M.I. Aquino and I. Levin, “In situ Raman spectroscopic investigation of chromium surface under hydrothermal conditions”, Applied Surface Science **180** (2001) 102.
[11] J.E. Maslar, W.S. Hurst, W.J. Bowers Jr., J.H. Hendricks, and M.I. Aquino, “In situ Raman spectroscopic investigation of nickel hydrothermal corrosion”, Corrosion **58** (2002) 225.
[12] R. E. Dietz, G. I. Parisot*, and A. E. Meixner, “Infrared Absorption and Raman Scattering by Two-Magnon Processes in NiO”, Phys. Rev. B, Condens. Matter **4** (1971) 2302.
[13] J.E. Maslar, W.S. Hurst, W.J. Bowers Jr., and J.H. Hendricks, “In situ Raman spectroscopic investigation of stainless steel hydrothermal corrosion”, Corrosion **58** (2002) 739.
[14] J.E. Maslar, W.S. Hurst, W.J. Bowers Jr., and J.H. Hendricks, “In situ Raman spectroscopic investigation of zirconium-niobium alloy corrosion under hydrothermal conditions”, J. Nucl. Mat. **298** (2001) 239.
[15] M. D. Cunha Belo et al., “Composition, structure and properties of the oxide films formed on the stainless steels 316L in a primary type PWR environment”, Corrosion Science **40** 23 (1998) 447.
[16] B. Beverskog and I. Puigdomenech, “Pourbaix diagrams for the ternary system of iron-chromium –nickel”, Corrosion **55** (1999) 1077.

NEUTRON FLUX AND FLUENCE DETERMINATION FOR BWR REACTORS

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Abstract

The embrittlement of pressure vessel steel is one of the most serious problems facing the nuclear industry. The vessel surveillance program for BWR reactors normally includes a dosimeter intended for removal after the first fuel cycle. Two BWR reactors were studied, the first one completed its first fuel cycle in 1991, and the second one in 1996. During the outage that followed, the flux wire dosimeters attached to the surveillance capsule at vessel 30º azimuth were removed for each unit. The dosimeters were tested and the results were analyzed. Test results and the associated determination of peak vessel flux, fluence, and fluence per EFPY (effective full power years), and comparisons for each unit are presented in this work.

1. INTRODUCTION

A reactor vessel is continuously being bombarded by radiation. The effects of such radiation result in embrittlement in the vessel materials, specifically in the core region or beltline. The norm for licensing a nuclear power plant requires that the reactor coolant pressure boundary be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing and possible accident conditions, the boundary behaves in a nonbrittle manner and also that the probability of a rapidly propagating fracture is minimized. This requires the calculation of changes in the fracture toughness of the reactor vessel caused by neutron irradiation. A surveillance program for a BWR reactor vessel requires the assurance that a brittle fracture is prevented [1]. The development of Pressure-Temperature curves needed for the reactor operation takes into account the irradiation embrittlement, therefore these curves have to be adjusted in accordance with the specifications established by the norm (10CFR50 appendix G). Three types of operations are considered when setting pressure and temperature levels. One is the hydrostatic pressure and leak tests, another is the non-nuclear heat up and cool down, and another is the reactor core critical operation. The vessel regions that affect these limits are divided into the closure flange region, the core beltline region and the non-beltline regions (nozzles, closure flanges, shell head plates and the control drive penetrations). It is generally assumed that key chemical elements in the vessel material influence the embrittlement and have been used for predicting the shift in nil ductility reference temperature RTNDT as a function of neutron fluence in the vessel beltline region. In the past, copper and phosphorus were used as indicators but currently phosphorus has been substituted by nickel.

2. DOSIMETERS

The reactor vessel materials surveillance program for these reactors includes the analysis of three capsules and a neutron flux wire dosimeter. The capsules contain Charpy samples of beltline base metal, welding and HAZ materials and several neutron flux wire dosimeters. These capsules are periodically withdrawn during the reactors operational life. Besides these dosimeters a flux wire dosimeter positioned at 30° azimuth is removed after concluding the first operation cycle. The neutron flux is proportional to the fission rate and this is nearly proportional to the reactor thermal power, therefore the results from the neutron flux wire dosimeters are used to establish a point in the relation between vessel neutron fluence and thermal power. With an extrapolation from this point, predictions can be made for the fluence during the reactor´s operational lifetime.

Table 1 Neutron flux wire dosimeters used to calculate neutron flux and fluence.

Dosimeter (Element)	Diameter (mm)	Purity (%)	Weight (g)	Activity (dps)
Iron	0.508	99.5	0.1338	$1.514 \times 10^4 \pm 3.02 \times 10^2$
Copper	0.762	99.99	0.4464	$1.964 \times 10^3 \pm 0.451 \times 10^2$

The wire dosimeters subsequently extracted will be used for the correction of the previous prediction given that these dosimeters will have a longer irradiation history.

The neutron flux wire dosimeters considered in this work were extracted in the first operational cycle of a BWR type nuclear plant after 547 working days at a 0.75 average capacity, or 412 days at full power or 1.13 effective full power years (EFPY). Three copper and three iron dosimeters were measured for the calculation of the neutron flux (>1MeV); the diameter and purity are shown in Table 1. The bare dosimeters are allocated in pairs in a stainless steel tube welded at both ends and filled with helium. A handle is welded to the tube for easy handling.

3. DOSIMETERS DISINTEGRATION RATE MEASUREMENTS

3.1 Experimental

The detector used was a 52.0 mm diameter and 44.1 mm length high purity germanium detector, (Ortec Poptop GEM 15910) with a 90 cm^3 active volume. The energy resolution for the detector system was 0.6 keV and 1.8 keV at energies of 136 keV and 1332 keV respectively. The associated electronics consists of an amplifier with the time constant set to 3μS, a high voltage supply and a multichannel analyzer card installed in a personal computer.

The data acquisition software was Ortec Maestro II recorded in the 4096 multichannel mode with the spectra being measured in the 40-2000 keV energy range. The spectra analysis was performed with the computer code Omnigam Ortec. An energy calibration for gamma photons between 60 keV and 1332 keV was obtained using the least squares method to fit energies and corresponding channel positions for certified gamma point sources.

3.2 Detector Efficiency Calibration

The full energy peak efficiency for a germanium detector was obtained experimentally measuring a set of reference standard gamma sources with well isolated peaks (Table 2). The efficiency is calculated using Eq. (1):

Table 2 Experimental and fitted efficiencies for a HPGe detector.

Isotope	Energy (keV)	$\in$ *measured* 10^{-3}	Uncer-tainty 10^{-4}	$\in$ 6 *Par. function* 10^{-3}
^{241a}m	59.53	.08157	.09266	.08145
^{133}Ba	80.09	.15420	.33870	.14891
^{109}Cd	88.03	.16108	.11616	.16121
^{57}Co	122.06	.18121	.12463	.18261
^{57}Co	136.47	.18024	.22610	.18165
^{133}Ba	276.38	.12920	.23060	.12909
^{133}Ba	302.85	.12040	.20760	.12049
^{51}Cr	320.08	.11446	.09312	.11537
^{133}Ba	356.02	.10540	.17330	.10582
^{133}Ba	383.85	.09900	.01859	.09935
^{113}Sn	391.7	.09994	.08463	.09766
^{137}Cs	661.66	.06145	.05130	.06189
^{54}Mn	834.84	.05169	.04051	.05104
^{65}Zn	1115.54	.04024	.03644	.04116
^{60}Co	1173.23	.04040	.02370	.03979
^{22}Na	1274.2	.03810	.07245	.03774
^{60}Co	1332.28	.03610	.03379	.03673

$$\varepsilon = \frac{A}{N_0\, P_\gamma\, \exp^{-\lambda t}} \tag{1}$$

where

ε	is the full energy peak efficiency
A	is the corrected net peak area rate (counts/s)
N_0	is the activity at the time of standardization
P_γ	is the absolute gamma ray emission probability
λ	is the decay constant
t	is the elapsed time since standardization.

The uncertainties can be calculated by the Propagation of Errors Equation [2] and assuming t is known precisely.

$$(\sigma \in)^2 = \left(\left[\frac{\partial \in}{\partial A}\right]^2 (\sigma A)^2 + \left[\frac{\partial \in}{\partial N_O}\right]^2 (\sigma N_O)^2 + \left[\frac{\partial \in}{\partial P\gamma}\right]^2 (\sigma P\gamma)^2 + \left[\frac{\partial \in}{\partial \lambda}\right]^2 (\sigma \lambda)^2 \right) \tag{2}$$

The experimental data was fitted to an analytical equation that fits the detector efficiency against the energy for a given distance [3,4]:

$$\in (E,P) = \left(P_1 + P_2(\ln E) + P_3(\ln E)^2 + P_4(\ln E)^3 + P_5(\ln E)^5 + P_6(\ln E)^7\right) / E \tag{3}$$

where

E is the gamma ray energy (MeV)
P_1 to P_6 are the fitting parameters.

The error in the calculated efficiency is calculated by:

$$\sigma_{\in}^2 = \sum_{i=1}^{n} \sum_{i=1}^{n} \frac{\partial Ep^*}{\partial p_i} * \frac{\partial Ep}{\partial p_j} * \rho(i,j) * \sigma_i \sigma_j \tag{4}$$

where

E is the gamma ray energy in MeV
σ is the error in the parameter
$\rho(i,j)$ is the correlation between σ_i and σ_j
n is the number of parameters.

3.3 Nuclear Reactions

The reactions of interest in the iron and copper dosimeters when bombarded with a neutron are described as:

$^{54}Fe(n,p)^{54}Mn$, $^{63}Cu(n,\alpha)^{60}Co$

The ^{54}Mn half life is 312.21 ± 0.05 days and the ^{60}Co is 5.2708 ± 0.0008 years . The threshold cross section for fast neutrons (E>1 MeV) were of 0.180 ± 0.036 barns for Fe and 0.00314 ±50.00078 for Cu obtained using an initial spectrum calculated at the inner vessel surface for a typical BWR (by vendor) [5,6,7].

3.4 Saturation Activity and Reaction Rate

The saturation activities are calculated for each radionuclide for the operation history and for a specific reference power using Eq. (5):

$$A_{si} = \frac{A_i}{e^{-\lambda_i t_d} \sum_{k \equiv 1}^{n} \frac{p_k}{P_o} \left(1 - e^{-\lambda_i t_k}\right) e^{-\lambda_i t_{wk}}} \tag{5}$$

where

A_i is the corrected measured activity
P_0 is the reference power (1931 MWt)
P_k is the average power in the time interval k
t_k is the irradiation time in the time interval k
t_{wk} is the waiting time from the end of the interval k to the end of irradiation.

Eleven time intervals covering 547 days of operation were taken.

The reaction rate (R_i) for each nuclide is given by Eq. (6):

$$R_i = \frac{A_{si}}{mN_m P_r P_{ri}} \tag{6}$$

where
m is the sample mass in milligrams
N_m is the number of atoms per milligram of the isotope
P_r is the concentration in percent of the target isotope
P_{ri} is the isotopic abundance.

The reaction rates for both types of dosimeters are shown in Table 3.

4. NEUTRON FLUX AND FLUENCE DETERMINATION

The neutron flux having an energy >1MeV is calculated by Eq. (7)

$$\Phi_i = R_i/\sigma_i \tag{7}$$

where

σ_I is the microscopic threshold (>1MeV) integral cross section (^{54}Fe=0.180 ± 0.036 barns, ^{59}Cu=0.00314 ± 0.00078 barns) [5].

The average of the calculation of the two types of dosimeters gives a neutron flux value of $1.46\text{x}10^9 \pm 2.19\text{x}10^8$ n/cm^2-s .

The >1MeV fluence determination (number of neutrons per square centimeter having energies greater than 1 MeV) is normally given for a period of 32 effective full power years (EFPY) and is carried out in two steps, first the dosimeters are analyzed and the flux and fluence determined at the dosimeter location, then lead factors are calculated which relate the flux at the location of the vessel wall exposed to the peak exposure.

The flux (n/cm^2-s) is calculated using the dosimeter gamma disintegration data and the reactor daily power history for the equivalent period. The methods used to obtain the dosimeter lead factor are described in the Final Safety Analysis Report [7]. A lead factor value (LF) of 0.96 was used for the peak location inside surface. The flux and fluence results for each dosimeter are shown in Table 3.

Table 3 Flux and fluence determinations for each dosimeter for the first cycle of a BWR reactor.

Dosimeter (Element)	Reaction Rate (dps/nucleus)	Flux (n/cm^2-s)	Fluence (n/cm^2)
Iron	$2.54x10^{-16}$	$1.4x10^{9}$	$5.1x10^{16}$
Copper	$4.77x10^{-18}$	$1.5x10^{9}$	$5.4x10^{16}$

The peak fluence (fp) is the calculated by

fp=(dosimeter fluence*32 EFPY)/(Dosimeter EFPY* LF)

The dosimeters measured fluence given at 1.13 EFPY was $5.2x10^{16}$ n/cm^2. Extrapolating this value to calculate the fluence at 32 EFPY gives a fluence of $1.48x10^{18}$ n/cm^2 . This is the fluence measured at the vessel inside surface and is used to calculate the fluence at ¼ T depth; that is at 1.25 inches in accordance with the relationship f ¼ = f surface(e-0.24*thickness) obtaining $1.12x10^{18}$ which is considerably lower than that of the fluence design value of $5.5x10^{18}$ n/cm^2 at ¼ T depth. The fluence value is used for impact analysis which in turn provides input data for the construction of the pressure-temperature curves. A computer code SPECTRIN was developed at the ININ for fluence and neutron spectra determinations. This code has the advantage that it does not require a trial spectrum as other codes do, because it is based on an algorithm developed by Sudar [9]. The results given by SPECTRIN and by the program SANDII [10] are given in Table 4.

Table 4 Comparison of flux and fluence determinations using SPECTRIN and SANDII.

Dosimeter (Element)	Flux (n/cm^2-s)	Fluence (n/cm^2)
SPECTRIN	$1.45x10^{9}$	$5.2X10^{16}$
SANDII	$1.39X10^{9}$	$4.9X10^{16}$

The results for the RT_{NDT} and ART (adjusted reference temperature) calculated using regulatory guide 1.99 [8] are given in Table 5.

Table 5 Beltline evaluation for plate lower shell containing 0.13% copper and 0.64% nickel.

EFPY (years)	□RT_{NDT} (ºF)	ART (ºF)
1	4.633	9.267
8	19.162	38.325
16	28.295	56.591
24	34.860	68.128
32	40.074	74.074

Table 6 Comparison of theoretical and experimental fast neutron flux at the foil location.

Experimental Fast (EF) Flux, n/cm^2-s (E > 1 MeV)	Theoretical Fast (CF) Flux, n/cm^2-s (E > 1 MeV)	Difference, Per Cent [(CF-EF)/CF]x100
2.39×10^9	2.75×10^9	13

5. NEUTRON FLUENCE EXTRAPOLATION FOR THE SIXTH CYCLE.

This section will review discuss a computational method used to reproduce the experimental results. The choice of neutronic methods chosen to reproduce the experimental flux was precipitated by the desire to run many future survey calculations for the prediction of the vessel flux in relatively short times. Thus the transport code DORT-4 [11], a two-dimensional Sn code, was chosen for the calculations. The results of the comparison and conclusions are presented in Table 6.

Using this calculation a prediction of the neutron fluence can be made for the sixth cycle of one of the BWR reactors under study and the obtained results are presented in Table 7. The input data used was the real operational time and the reactor´s capacity factor and taking into account that this reactor increases 5% its thermal power during the last 3 cycles.

6. CONCLUSIONS

The flux wire test results show a nominal peak fluence on the vessel at 32 EFPY of $1.48 \times 10^{18.}$. The fluence determined by dosimetry is significantly lower than the original design fluence value of 5.5×10^{18} n/cm^2 at ¼ T (which includes a safety factor of 2 on lead factor). This lower trend is consistent with the results of dosimetry at other BWR plants.

The results of the calculations are in agreement with the experimental data, thus validating the surveillance computer program developed.

Table 7 Neutron flux and fluence at the wall of a BWR vessel calculated by DORT

Cycle	Operational time (days)	Capacity Factor %	Flux (E > 1MeV) (n/cm2s)	Fluence (n/cm2)	EFPY
Thermal Power 1931 MW					
1	514	73.13	2.75E+09	8.93E+16	1.02983068
2	391	90.18	2.75E+09	8.38E+16	0.96603781
3	452	87.92	2.75E+09	9.44E+16	1.08876274
Thermal Power 2027 MW					
4	534	72.03	2.88E+09	9.57E+16	1.05380877
5	566	78.9	2.88E+09	1.11E+17	1.22349041
6	522	83.5	2.88E+09	1.08E+17	1.19416438
			Total	**5.83E+17**	**6.56E+00**

REFERENCES

[1] ASTM E 185-82 Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear. Power Reactor Vessels. ASTM Committee E-10 on Nuclear Technology and Applications (1982).

[2] BEVINGTON, P.R., Data Reduction and Error Analysis for Physical Sciences, (1967)

[3] GRAY P.and AHMAD, A. Nucl. Instr. and Meth. 227(1985)577

[4] LONGORIA, L.C. and BENITEZ, J.S., Appl. Radiat. Isot. 47(1996)339

[5] ASME F-261-90 “Determining Neutron Fluence Rate Spectra by Radioactivation Techniques”(1990).

[6] INTERNATIONAL REACTOR DOSIMETRY FILE (IRDF-90) IAEA-NDS-42 (1990).

[7] FINAL SAFETY ANALYSIS REPORT (FSAR) UNIT1, Comision Federal de Electricidad Vol. VI Chapter 4 (1979).

[8] USNRC REGULATORY GUIDE 1.99, REVISION 2, Radiation Embrittlement of Reactor Vessel Materials, (1988)

[9] SUDAR, S., A Solution for the Neutron Spectrum Unfolding Problem without using Input Spectrum, Report INDC(HUN)—026/L (1989)

[10] SANDII, Neutron Flux Spectra Determination by Multiple Foil Activation-Iterative Method (Manual) RSIC Oak Ridge National Laboratory, (1970)

[11] “TORT-DORT-PC: Two and Three Dimensional Discrete Ordinates Transport Version 2.7.3,” CCC-543, Oak Ridge National Laboratory, Oak Ridge, Tennessee, (June 1996).

IMPROVEMENT OF REGULATORY PROCEDURES FOR RPV INTEGRITY AND LIFETIME ASSESSMENT FROM THE VIEWPOINT OF BRITTLE FAILURE RESISTANCE

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Abstract

This paper presents the main approaches of new Russian regulatory procedures for reactor pressure vessel integrity assessment: “Procedure for WWER reactor pressure vessel lifetime assessment during operation”, МРК-СХР-2000 and “Procedure for WWER-1000 RPV nozzle area brittle failure resistance calculation”, МПКР-2002. The results of calculations with the use of the new procedures for WWER-1000 and WWER-440 reactor pressure vessels under pressurized thermal shock are also presented.

1. INTRODUCTION

The integrity of reactor pressure vessel must be assured during the whole service lifetime since the modern designs of reactors (both in Russia, and abroad) are not provided with technical facilities that allow mitigating the consequences of a catastrophic destruction of reactor pressure vessel. The adequate approach to reactor pressure vessel integrity assessment is the basis for the safe operation during the whole operational lifetime. It gives the capabilities to estimate the operational lifetime of reactor pressure vessel, which allows, in case of necessity, realizing preventive and mitigating measures directed on reduction of brittle failure risk.

The present paper presents the regulatory procedures recently developed in the Russian Federation for integrity assessment of WWER reactor pressure vessels. The main reasons for development of such procedures are as follows:

– to take into account the new data on RPV material behavior under irradiation and the new data on RPV material fracture toughness;
– to take into account the capabilities of non-destructive testing in reactor pressure vessel integrity assessment;
– to take into account the international experience in reactor pressure vessel integrity assessment;
– to remove extra conservatism from reactor pressure vessel integrity assessment using valid and well founded approaches.

The following procedures are considered in this paper:

- “Procedure for WWER reactor pressure vessel lifetime assessment during operation”, МРК-СХР-2000 [1] developed by CRISM "Prometey", with the participation of NIKIET, ICP MAE and OKB “Gidropress”, which establishes the rules for determination of reactor pressure vessel residual lifetime. This procedure is applied to cylindrical part of reactor pressure vessel, exposed to neutron irradiation.

- “Procedure for WWER-1000 RPV nozzle area brittle failure resistance calculation”, МПКР-2002 [2] developed by NIKIET, with participation of CRISM "Prometey", ICP MAE and OKB “Gidropress”.

2. MAIN APPROACHES OF THE CONSIDERED REGULATORY PROCEDURES

2.1. Postulated defects

According to procedure МПКР-2002 [2] the postulated defect for WWER-1000 reactor pressure vessel nozzle area is a crack in the base metal with the depth a_0=0,07S (S is the wall thickness of reactor pressure vessel) and length $2c_0=6a_0$ (Fig 1), its contour being formed by two lines:

a) The first line is the line “base metal - cladding ”;
b) The second line is a part of the arc of an ellipse.

According to procedure МРК-СХР-2000 [1] the postulated defect for unclad reactor pressure vessel is the surface semi elliptical crack with depth a_0=0,07S and aspect ratio a_0/c_0=1/3 (Fig 2).

The scheme for the choice of postulated defects for clad reactor pressure vessel according to procedure МРК-СХР-2000 [1] is presented in Fig 3. Proceeding from Fig 3, the choice of a postulated defect depends on the performance of non-destructive testing of the cladding. In Fig 3: S is the wall thickness of reactor pressure vessel, and S_C is the cladding thickness.

For all the above mentioned defects the cyclic growth during operation is taken into account and the calculational sizes of defects are finally established.

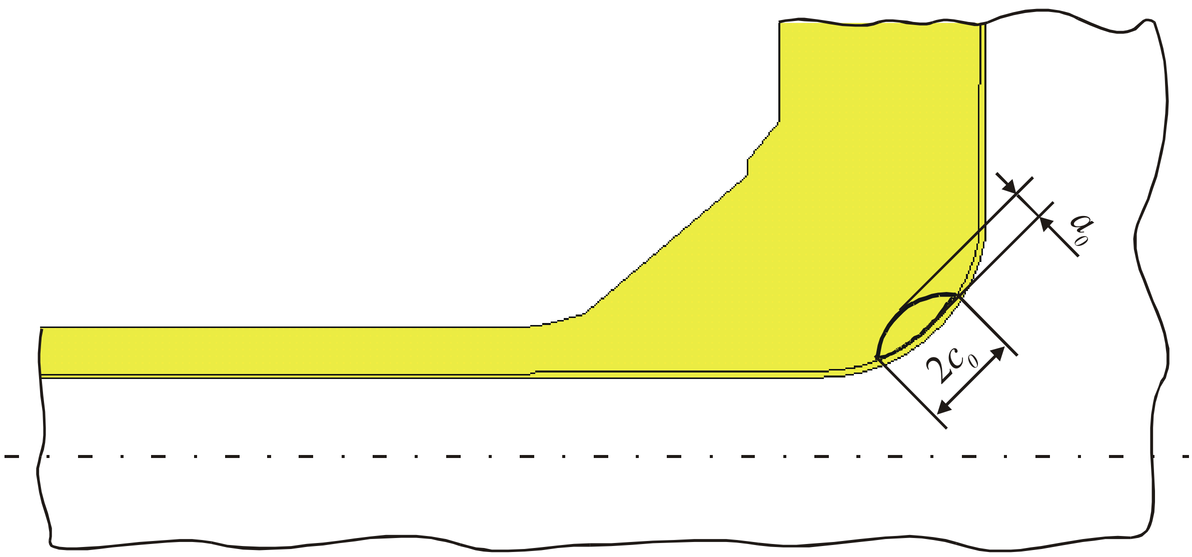


FIG. 1 - Postulated defect for WWER-1000 reactor pressure vessel nozzle area

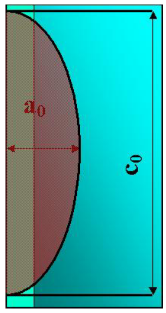


FIG. 2 - Postulated defect for unclad reactor pressure vessel

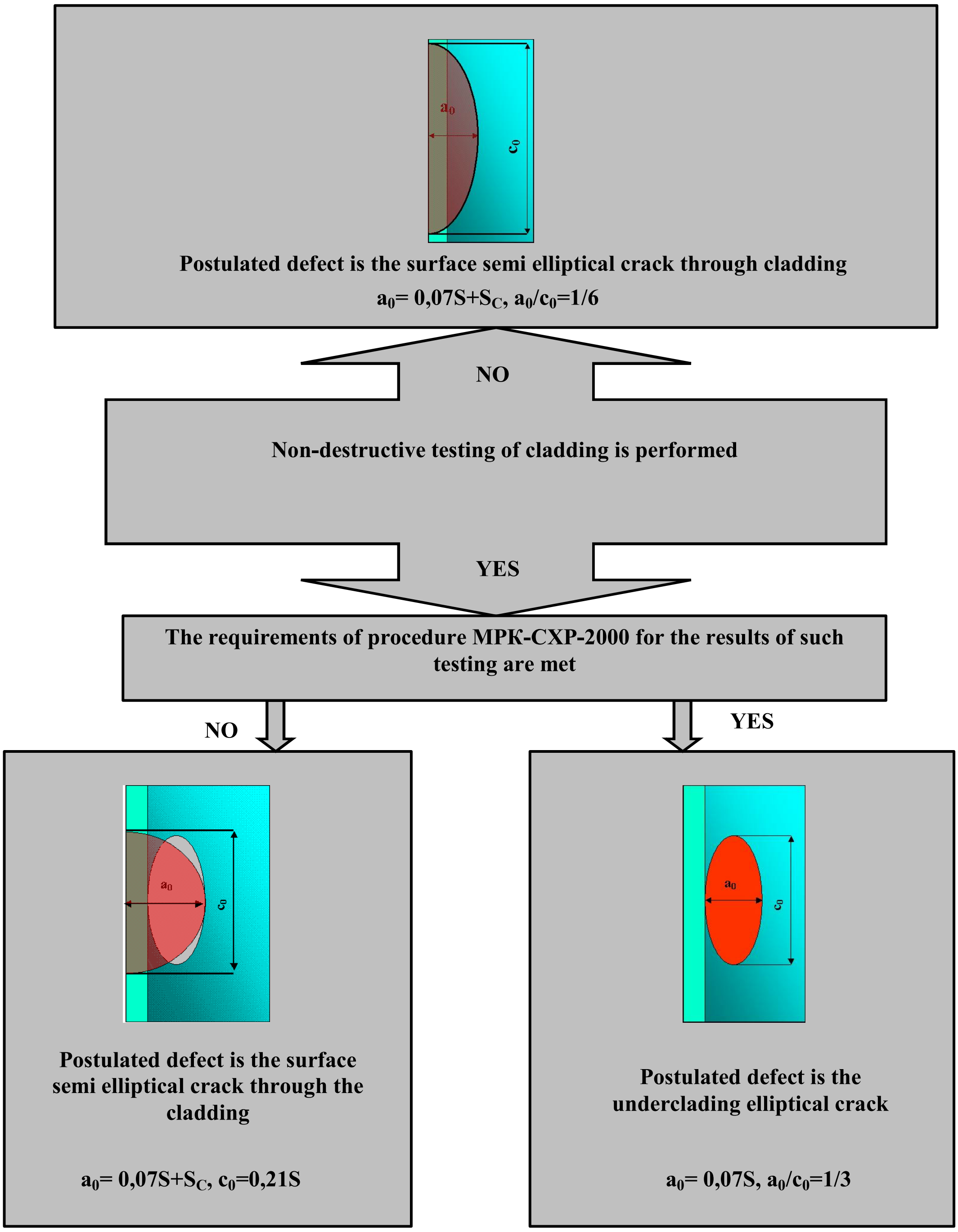


FIG. 3 - Choice of postulated defects for the clad reactor pressure vessel

2.2. Calculational characteristics

According to both procedures the initial information on the fracture toughness is the base curve $K_{IC}(T-T_K)$ for the specimen thickness B=150 mm that corresponds to probability P_f=0,05 of finding of fracture toughness value being less than K_{IC}. One base curve for all WWER pressure vessels materials is proposed:

$$\mathbf{K_{IC} = 23 + 48 \cdot exp[0,019(T-T_к)]}$$

Allowable values of stress intensity factor for crack with the crack front length B_i can be determined from the base curve by the following formula:

$$[K_{IC}] = \left(\frac{\overline{B}}{B_i}\right)^{1/4} (K_{IC} - K_{\min}) + K_{\min}$$

Where

$\overline{B}$ =150 mm,
$K_{\min}$ = 20 MPa $m^{1/2}$.

2.3. Strength criteria

According to both procedures МРК-СХР-2000 and МПКР-2002 the strength criteria for any point at the crack front in the base metal, can be written down as follows:

$$K_I = n_k K_{IP} + K_{IS} \le [K_{IC}(T-T_k)]$$

$$K_I = K_{IP} + K_{IS} \le [K_{IC}(T-T_k-\Delta T)],$$

where

K_{IP} is stress intensity factor due to primary stresses (MPa $m^{1/2}$),K_{IS} is stress intensity factor due to secondary stresses (MPa $m^{1/2}$).

The safety margins for different categories of conditions are presented in Table 1.

Table 1

Condition	**n_i**	**ΔT, ^{O}C**
Normal operation, i=1	**2**	**30**
Hydrotests, i=2	**1,5**	**30**
Anticipated transients, i=3	**1,25**	**30**
Postulated accidents, i=4	**1,1**	**0**

According to both procedures МРК-СХР-2000 and МПКР-2002 (for normal operation, hydrotests and anticipated transients), the strength criteria for any point of the crack front in the cladding or contacting with cladding can be written as follows:

$(K_I)_i = n_i \cdot K_{IP} + K_{IS} \leq K_{JC}$,

where

$K_{JC} = \sqrt{\frac{J_C E}{1-\nu^2}}$, J_C is the critical value of J-integral under brittle failure (if the crack ductile growth before brittle failure was less than 0,2 mm) or J-integral value, determined on cladding J_R-curve for the crack ductile growth 0,2 mm.

For postulated accident according to procedure МПКР-2002, for any point of the crack front in the cladding or contacting with cladding the following conditions shall be met:

$(K_I)_i = n_J \cdot K_{IP} + K_{IS} \leq K_{J1}$,

where $K_{J1} = \sqrt{\frac{J_1 E}{1-\nu^2}}$, J_1 – J-integral value, determined on cladding J_R-curve for the crack ductile growth Δl=1,0 mm.

and

$\Delta a_C + \Delta l_{PA} \leq S_C/2$,

where

Δa_C is the crack cyclic growth in the cladding during considered operation time (mm),
Δl_{PA} - crack ductile growth in the cladding during considered transient (postulated accident) (mm).

2.4. Precise analysis of strength criteria

The precise analysis can also be performed to obtain more adequate results. The criterion for precise analysis can be written down as follows:

$$\frac{1}{B}\int_B \frac{(K_I(\varphi) - K_{\min})^4}{(K_{IC}(\varphi) - K_{\min})^4}\, dB < 1$$

where
$K_{IC}(\varphi)$ - fracture toughness distribution through the crack front (MPa m$^{1/2}$),
$K_I(\varphi)$ - stress intensity factor distribution through the crack front (MPa m$^{1/2}$).

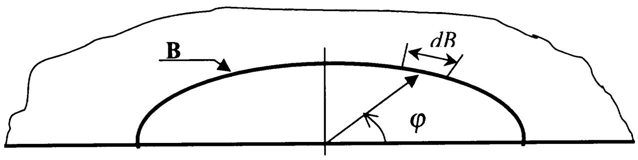


FIG. 4 – Precise analysis

3. CALCULATION RESULTS WITH THE USE OF THE CONSIDERED REGULATORY PROCEDURES

3.1. Calculation results with the use of procedure МРК-СХР-2000

Transients (postulated accidents) with the formation of ECCS cold water plumes in the reactor pressure vessel downcomer were considered («LOCA» for WWER–1000 reactor pressure vessel, «Secondary LOCA» for WWER–440 reactor pressure vessel with cladding, «SG steamline break» for WWER–440 reactor pressure vessel without cladding).

The governing location of reactor pressure vessel from point of veiw of brittle failure resistance was considered (weld №4).

Water mixing in the reactor pressure vessel downcomer was taken into account in boundary conditions in temperature fields calculations.

Residual stresses in the welds were considered. For reactor pressure vessels with cladding residual stresses due to cladding manufacturing were considered (according to procedure МРК-СХР-2000 recommendations).

Temperature and stress calculations were performed with the use of FEM Code MARC on 3D FEM models. FEM discrete models are presented in figs 5-7. Elastic – plastic stress calculations were performed.

Stress intensity factor calculations were performed by weight functions method. Additionally for WWER–440 reactor pressure vessel without cladding the crack was simulated directly (numerical calculation of J-integral values was performed).

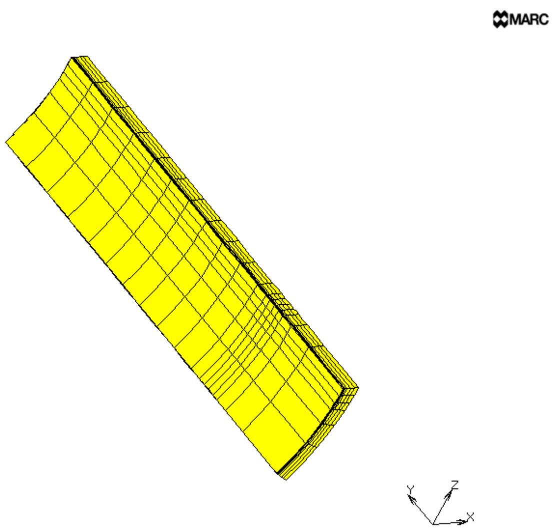


FIG. 5 – Discrete FEM model for WWER-1000 reactor pressure vessel

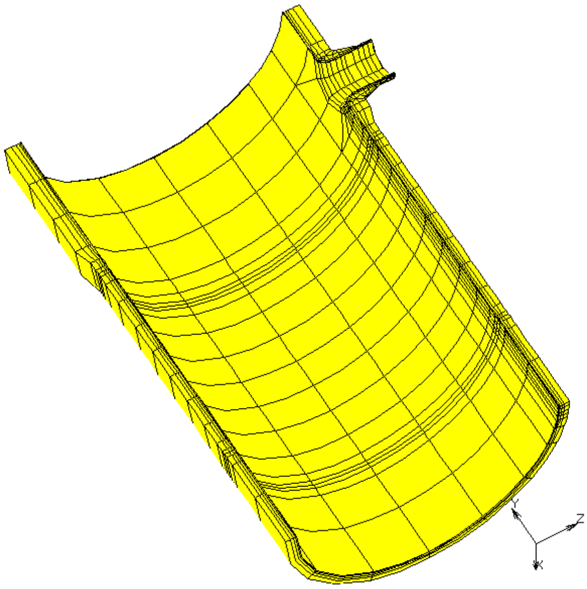


FIG. 6 – Discrete FEM model for WWER-440 clad reactor pressure vessel

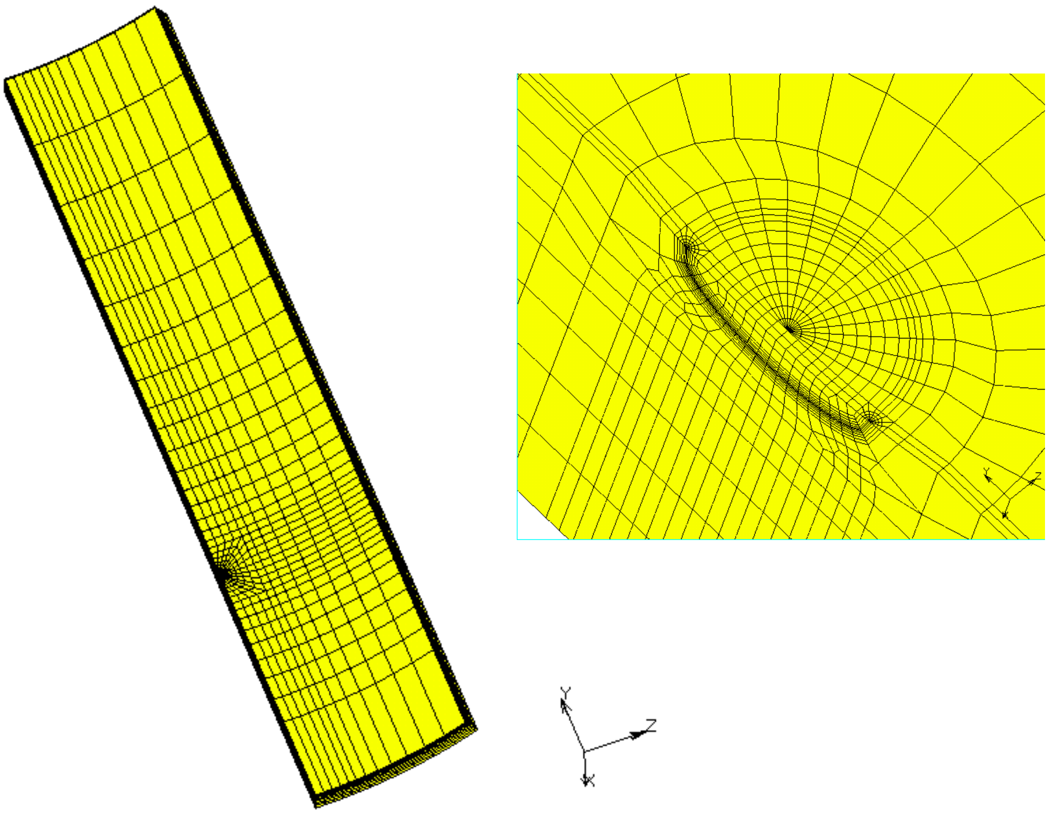
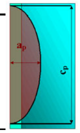
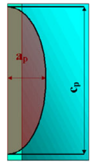
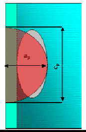
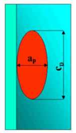
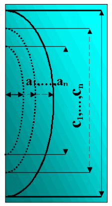
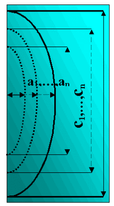


FIG. 7 – Discrete FEM model for WWER-440 unclad reactor pressure vessel (the crack was simulated directly)

Calculation results for WWER reactor pressure vessels according to procedure MPK-CXP-2000 are presented in the Table II. The results are presented in the form of maximum allowable transition temperature $T_{ка}$. Calculation results according to design procedure [3] are also presented in Table II.

Table 2 Results of temperature T_{ka} calculation for WWER reactor pressure vessels

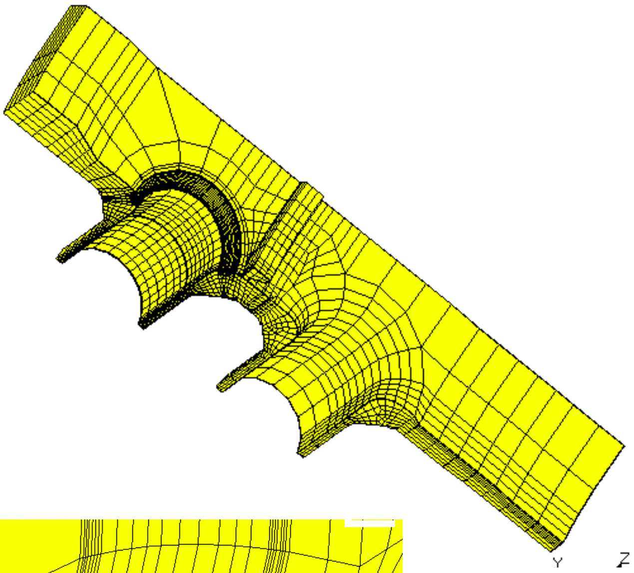
	T_{ka} °C							
Calculation procedure	**Postulated defects**	**WWER-1000 (LOCA)**		**Clad WWER-440 (Secondary LOCA)**		**Postulated defects**	**Unclad WWER-440 (SG steamline break)**	
		Strength criteria	**Precise analysis**	**Strength criteria**	**Precise analysis**		**Strength criteria**	**Precise analysis**
Procedure МРК-СХР-2000		**48**	**52**	**151**	**151**		**188**	**194**
		**60**	**84**	**172**	**190**			
		**70**	**85**	**200**	**216**			
Design procedure [3]		**80**		**168**			**188**	

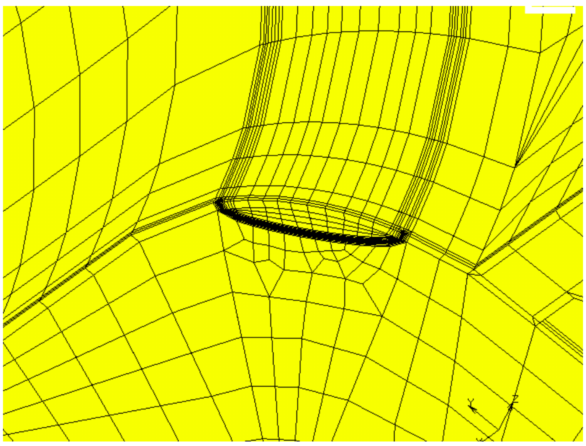
3.2. Calculation results with the use of procedure МПКР-2002

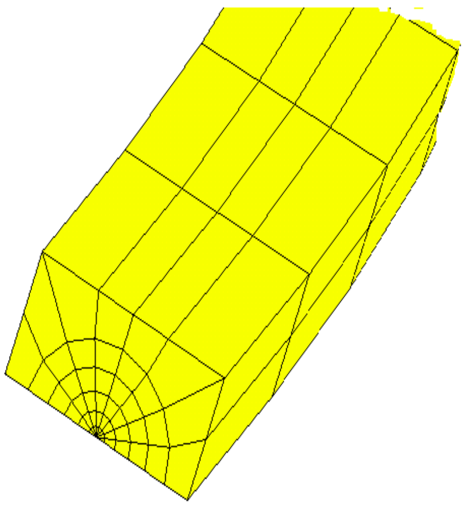
Calculation was performed for outlet nozzle Dnom 850. This nozzle is the most loaded component of WWER-1000 reactor pressure vessel nozzle area under conditions (postulated accidents) with injection of ECCS cold water in the reactor upper chamber. Postulated accident “Dnom 850 leak” with injection of ECCS cold water in the reactor upper chamber was considered.

Residual stresses due to cladding manufacturing were taken into account. Temperature and stress calculations were performed with the use of FEM Code MARC on 3D FEM models. FEM discrete model is presented in fig 8. Stress intensity factor was determined by recalculation of J-integral values obtained by FEM numerical solution. Stress intensity factor was determined by the following formula:

$$K_I = \sqrt{\frac{JE}{1-\nu^2}}$$







front

FIG. 8 - Discrete FEM model for WWER-1000 pressure vessel nozzle area

Results of stress intensity factor calculation $(K_I)_4=1,1\cdot K_{IP}+K_{IS}$ (the maximum values for all crack front points in the base metal) under condition “Dnom 850 leak” are presented in Fig 9. Proceeding from the Fig 9 the strength criteria of procedure МПКР-2002 are met for the base metal (transition temperature value is $T_K=20\ ^OC$).

Results of stress intensity factor calculation $(K_I)_4=1,1\cdot K_{IP}+K_{IS}$ (the maximum values for all crack front points contacted with the cladding) under condition “Dnom 850 leak” are presented in Fig 10. Proceeding from the Fig 10 the strength criteria of procedure МПКР-2002 are met.

Results of value Δl_{PA} calculation under condition “Dnom 850 leak” are presented in Fig 11. Proceeding from the Fig 11 Δl_{PA} not exceed 0,4 mm and therefore $\Delta a_C+\Delta l_{PA} = 3,6 + 0,4 =4$ мм $\leq S_C/2 = 5,5$ mm.

4. CONCLUSIONS

4.1. Procedure МРК-СХР-2000

Analysis of results obtained for clad WWER pressure vessels shows that these results essentially depend on the following:
- the performance of non-destructive testing of the cladding;
- the results of such testing (requirements of procedure МРК-СХР-2000 must be satisfied);
- the availability of irradiated cladding K_{JC} data.

If non-destructive testing of the cladding is not performed the calculation according to procedure МРК-СХР-2000 gives more conservative results than the calculation according to the design procedure [3]. If non-destructive testing of the cladding is performed and the results of such testing meet the requirements of procedure МРК-СХР-2000 the calculation according procedure МРК-СХР-2000 gives more optimistic results than the calculation according to design procedure [3].

Analysis of results obtained for unclad WWER reactor pressure vessel shows that the calculation according to procedure МРК-СХР-2000 gives less conservative results than calculation according to design procedure [3].

Procedure МРК-СХР-2000 corresponds to the up-to-date insight of the problem. The results obtained according to procedure МРК-СХР-2000, at observance of some conditions (the presence of non-destructive testing of the cladding and the availability of irradiated cladding K_{JC} data), are more optimistic, than results obtained according to the design procedure [3]. In a lot of cases the use of procedure МРК-СХР-2000 allows proving a longer operational lifetime of reactor pressure vessel, than the use of the design procedure [3]. It is necessary in every possible way to stimulate the introduction of the non-destructive testing of the cladding at NPPs.

4.2. Procedure МПКР-2002

Procedure МПКР-2002 corresponds to the up-to-date insight of the problem. The strength criteria of procedure МПКР-2002 are met for WWER-1000 nozzle area under pressurized thermal shock.

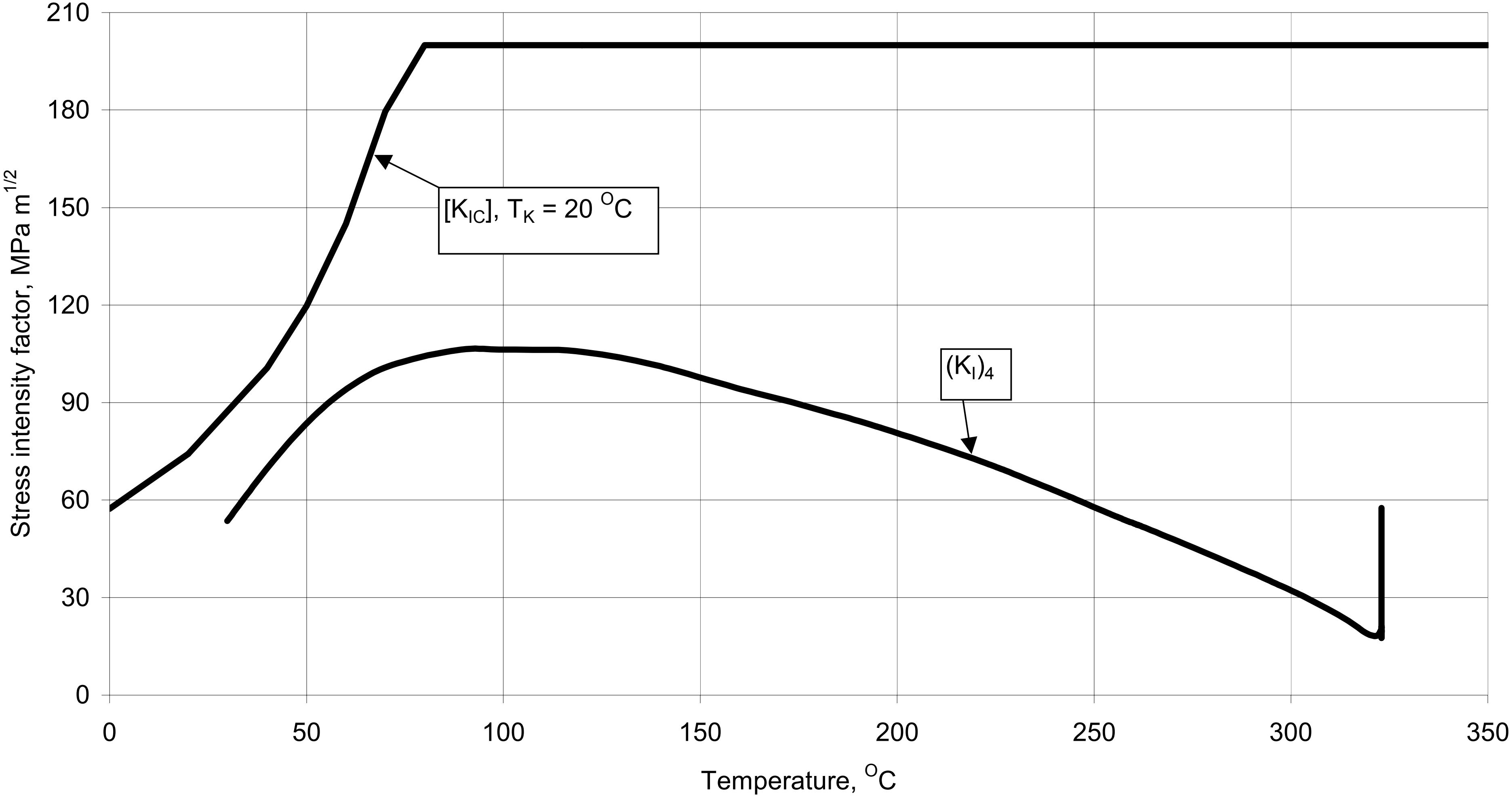


Figure 9 - WWER-1000 reactor pressure vessel. Outlet nozzle Dnom 850." Dnom 850 leak".
Stress intensity factor calculation results.

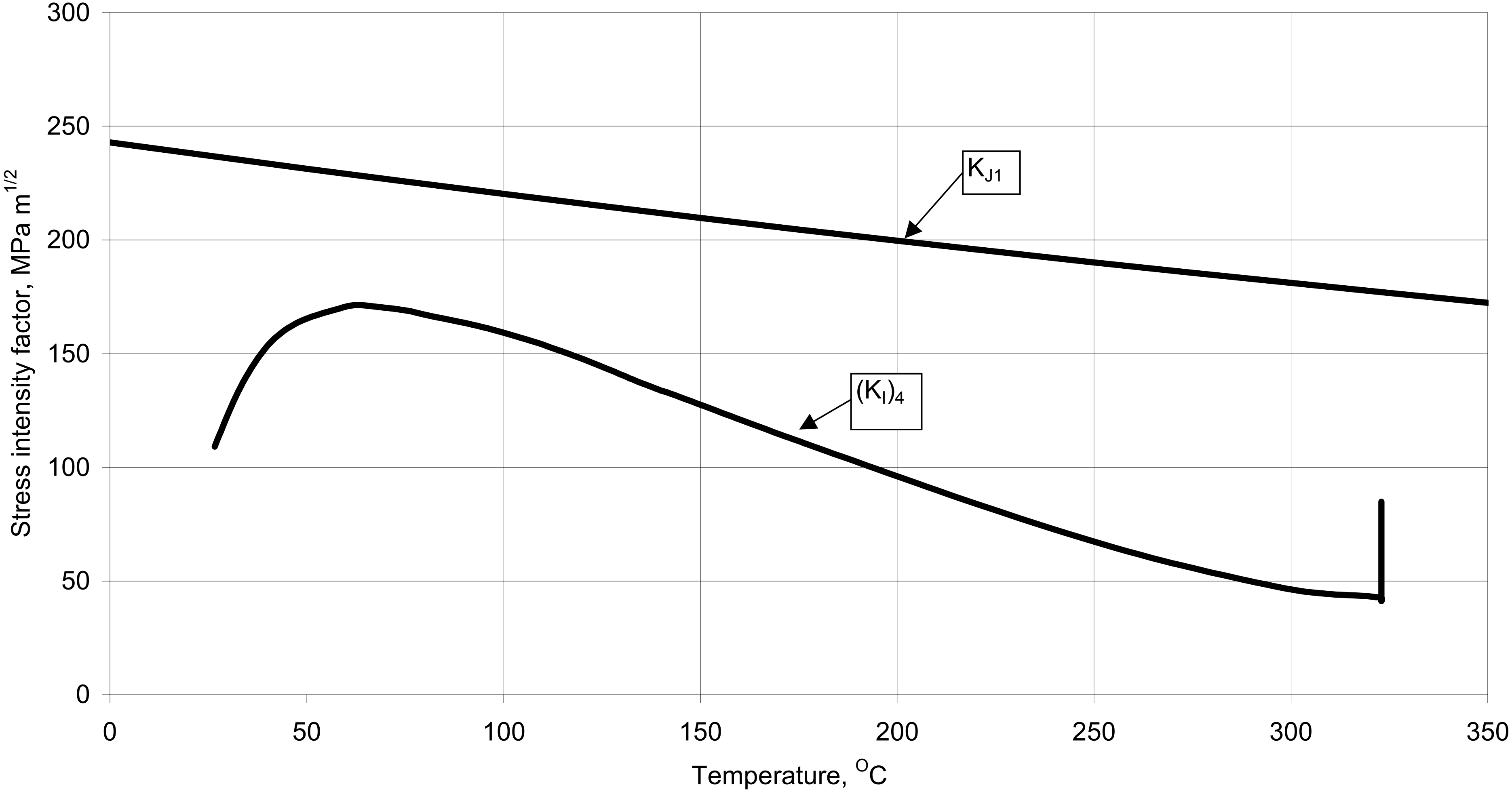


Figure 10 - WWER-1000 reactor pressure vessel. Outlet nozzle Dnom 850." Dnom 850 leak".
Stress intensity factor calculation results.

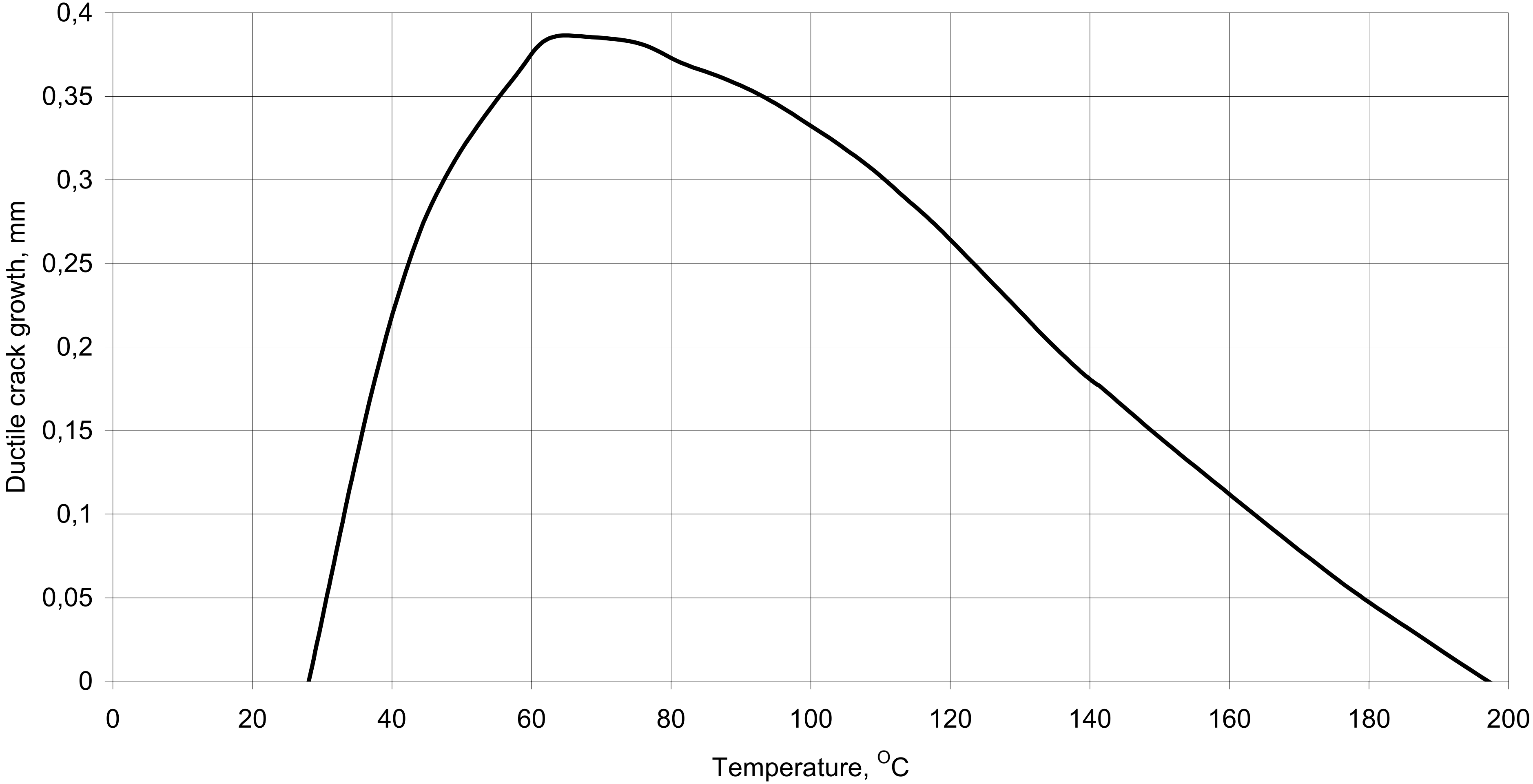


Figure 11 - WWER-1000 reactor pressure vessel. Outlet nozzle Dnom 850." Dnom 850 leak".
Ductile crack growth Dl_{PA} calculation results.

REFERENCES

[1] "Procedure for WWER reactor pressure vessel lifetime assessment during operation", МРК-СХР-2000, St.-Pitersburg-Moscow, 2000

[2] "Procedure for WWER-1000 RPV nozzle area brittle failure resistance calculation", МПКР-2002, Moscow, 2002

[3] Regulations for strength calculation of equipment and pipelines of atomic power plants, PNAE G - 7-002-86, Moscow, Energoatomizdat, 1989

NPP COMPONENT MAINTENANCE AND LIFE MANAGEMENT IN THE RUSSIAN FEDERATION

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Abstract

This report represents the conceptual strategies (ideas) on life management programs for nuclear power plants. Use of the optimum programs for NPP's NDE, maintenance service, operation and service life can provide the best economic benefit for the utilities. The paper presents general approaches to life management, maintenance service, and risks of operating and service life of NPPs in Russia. The report offers some optimized ways for the solution of these important tasks.

1. INTRODUCTION

The package of the new Federal Laws of Russian Federation in the industrial safety and licensing of operation determines basic principles, methodology and requirements in the field of life management and risks of operating and service life of potentially dangerous objects in the industry and power generation. A main principle of these requirements stipulates that at any moment of NPP operation the quantitative indexes of risk should not exceed acceptable social risk. On the other hand for gaining the best profit from operation of the equipment, system or component the second principle should be applied as a whole: the best possible commercial use of capabilities and service properties of the equipment. This principle reflects an essence of the conceptual purpose of savings in energy and money. It declares desire of the power utility shareholders to use the potential profit most effectively. Implementation of this principle assumes transition from the becoming obsolete concept of maintenance, repairs and replacements of the equipment, pipelines or systems to the up to date economic concept of the NPPs operation by using condition monitoring and maintenance of the SSC in accordance with the assessment of their real state.

2. LIFE MANAGEMENT RE-EVALUATION OF MAJOR COMPONENT, PIPING SYSTEM AND FRAMEWORK OF NPPs IN RUSSIA

To assure high reliability and safety of NPP during its transition to a new principle, it is necessary to assure excellence in NDE and maintenance service. Therefore it is necessary to select new strategies and programs of life management and service procedures from the viewpoint of technical and financial assessment of efficiency and profitability of NPP operation. Another task arising from this approach is the optimization of costs of life management based on the analysis of the basic strategies of NPP operation with the purposes of:

- having the component service life as long as at least the design life is;
- operating the plant till the failure;
- operating the plant till the pre-failure condition.

This problem may be represented as a chain of various tasks for NDE and maintenance service, repair and replacement of main NPP components. The solution of these tasks could be a program of component maintenance and repair (CMRP). Draft for life management assessment is shown in fig 1

The structure of the normative documentation of CMRP, as a rule, consists of several groups of the documents of a different legal significance. However further improvement and development of the new CMRP normative documents is still necessary. Developing and implementing the guidelines for planning life management should be the scope of guidelines includes rules for service inspection of Nuclear Power Plant Components. These rules include means for assessing an inspection programs and its plan, conceptual strategies on life management programs. The approach is to emphasize safe and reliable operation through cost effective life management program. Two risk based approaches are presented with each successive level requiring more data analysis to reduce reliance on conservative assumptions, there are – qualitative risk and quantitative risk for nuclear power plants operation in Russia.

Thus the formulation of the mandatory norms (standards) should be based on the substantial practice, being a balanced generalization of this practice and operational feedback. The new documents should not contain in their essence new regulatory requirements connected with the assurance of NPP safety. Definition of Qualitative or /and Quantitative risk is shown in fig 2. Methodology for different levels of the calculation lifetime and risk is shown in fig. 3.

Modern information technologies assume the use of integrated systems of supporting databases for the analysis of the feedback experience of NPPs operation. The development and maintenance of such databases (ODB) is the second main task in the implementation of CMRP. The fig 4 show as OBD will be used for life management service.

Today there is already logically valid philosophy on ODB based on a modular principle and its conversion ideology to managing systems of a higher hierarchical decision-making level. It will provide a capability for administrative and operating staff to receive quickly the optimum decision on particular managerial implications of CMRP in technology and conditions of safe operation of components NPP on the basis of comprehensive information support. In general the process of management and maintenance of the ODB for the purposes of CMRP, also is as good as for the purposes of NPP load factor increase, and is defined by the methodological algorithm: "Planning - Preparation - Checking - Action”. The key unit for the ODB is “data banks” where the information about processes of aging and accumulated knowledge and concepts about the mechanisms of these processes are stored. The latter emphasizes in particular a vital necessity of continuing research in all directions connected with study, analysis and systematization of aging factors at NPP, implementation of measures mitigating those factors negative for NPP safety. In databases there should be information on capabilities of CMRP technologies and means for their implementation. It is a way of improving the existing databases on NPP materials and designs.

The guideline of all CMRP measures is the improving current condition and capability of components for safe NPP operation as well as the assurance of its service life. The structure of CMRP measures is corporate with other operating NPP's programs. Thus the ODB should be understood as one of the managing systems of NPP PLIM which is supported by the whole complex of technological, organizational and financial measures ensuring mitigation or elimination of influence of aging mechanisms affecting engineering and piping systems and technological NPP equipment with the aim of reaching reliable, safe and economically effective operation of the whole NPP.

The approach to creation of the list of primary CMRP measures is presented in the fig 5. Based on the ranking (fig 6) of CMRP measures and their importance the value of operational risk (Ro) for elements of the equipment (systems) could be calculated. The evaluation of risk is based on calculation of probability (Pd) of NPP components destruction. In general the quantitative risk (Ro) can be determined as a multiplication of probability of destruction (failure or refusal)) of a component (Pd) and the value of integral losses (SS) arising from emergency situations, costs of liquidation of emergency and its consequences or indemnification of the caused damage.

$$Ro = Pd \times SS$$

The quantitative and qualitative factors of risk can be unified in certain units and represent statistical risk model of a type:

$$Cijkm = m + Rk + Ai + B(i)) \ldots. + Dm + A(i))\ im + B(i))\ m(i)) \ldots \ldots. + \ldots\ Eijkm,$$

Where

Cijkm - value of operational risk (or other incidental variable describing efficiency of measures CMRP).

The element of the risk factor, for example from particular conditions of NDE can be reckoned on the basis of known criteria of mathematical statistics, and then the optimum version of NDE in co-ordinates “Risk – NDE Condition - NDE cost - Increase of load factor- Increase of lifetime” could be selected.

The expediency of scientific search for the solution of CMRP goals for ageing NPPs is determined by economic profit and tendency to more effective utilization of radioactive waste. The general goal of the CMRP program is the increase (optimization) of NPP operating lifetime. The achievement of this main goal is connected with the obtaining positive outcomes of research activities in some scientific areas. Among them there are: introduction of new technologies for determination of safe service life of the equipment and systems and component maintenance based on the actual condition and individual parameters.

SUMMARY

1. NPP component maintenance and life management in Russia is one of the major issues that determine the lifetime of a framework component and piping.
2. The approach described in the paper demonstrate that option of life management service can be used three main idea:
3. Having the component service life as long as at least the design life is;
   - Operating the plant till the failure;
   - Operating the plant till the pre-failure condition.

4. Diagnostic, Understanding and Modeling Material Damage Mechanisms Is Essential for Option of Life Management Service and for Prediction of Remaining Safe Life.

5. Need to Predict Remaining Life to Extend Component and Piping Operating Life. If Remaining Lifetime and Risk Cannot Be Predicted Component and Piping Must Be Monitored or repair (replacement) to Ensure Safe and Profit Future Operation.
6. Use of the Optimum Programs for NPP's NDE, Maintenance Service, Operation and Remaining Life Can Provide the Best Economic Benefit for the Utilities.

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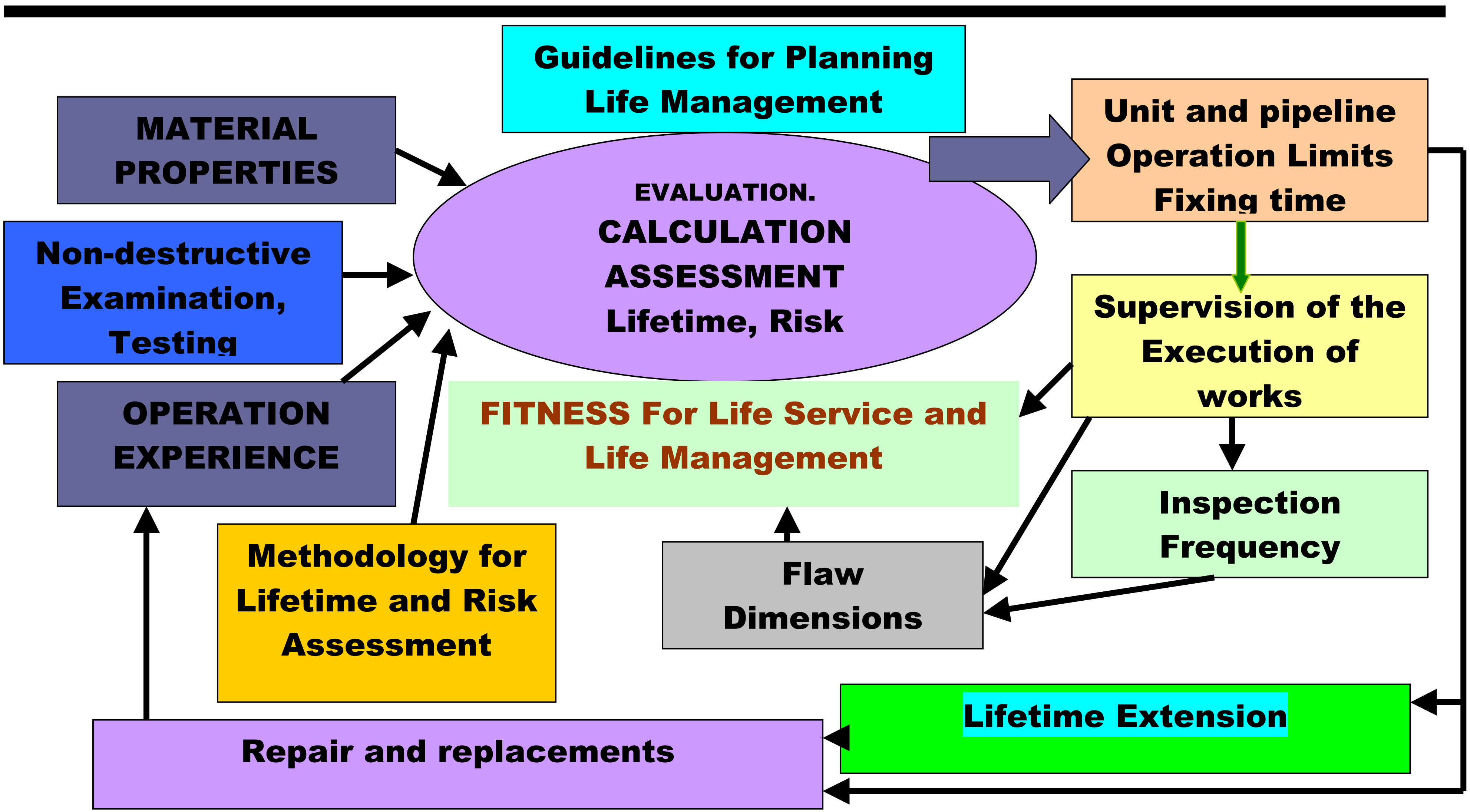

FIG. 1 Life Management Assessment Methodology

$$\textbf{Risk} = \boxed{\begin{array}{c}\textbf{Consequence of}\\ \text{Failure}\end{array}} \times \boxed{\begin{array}{c}\textbf{Probability of}\\ \text{Failure}\end{array}}$$

The quantitative and qualitative factors of risk can be unified in certain units and represent statistical risk model of a type:

$$\mathbf{RISK_{ijkm} = m + R_k + A_i + B_{(i)} \ldots + D_m + A_{(i))\ i,m} + B_{(i))\ m(i))} + \ ..+E_{ijkm}}$$

FIG. 2 Definition of Qualitative or/and Quantitative RISK

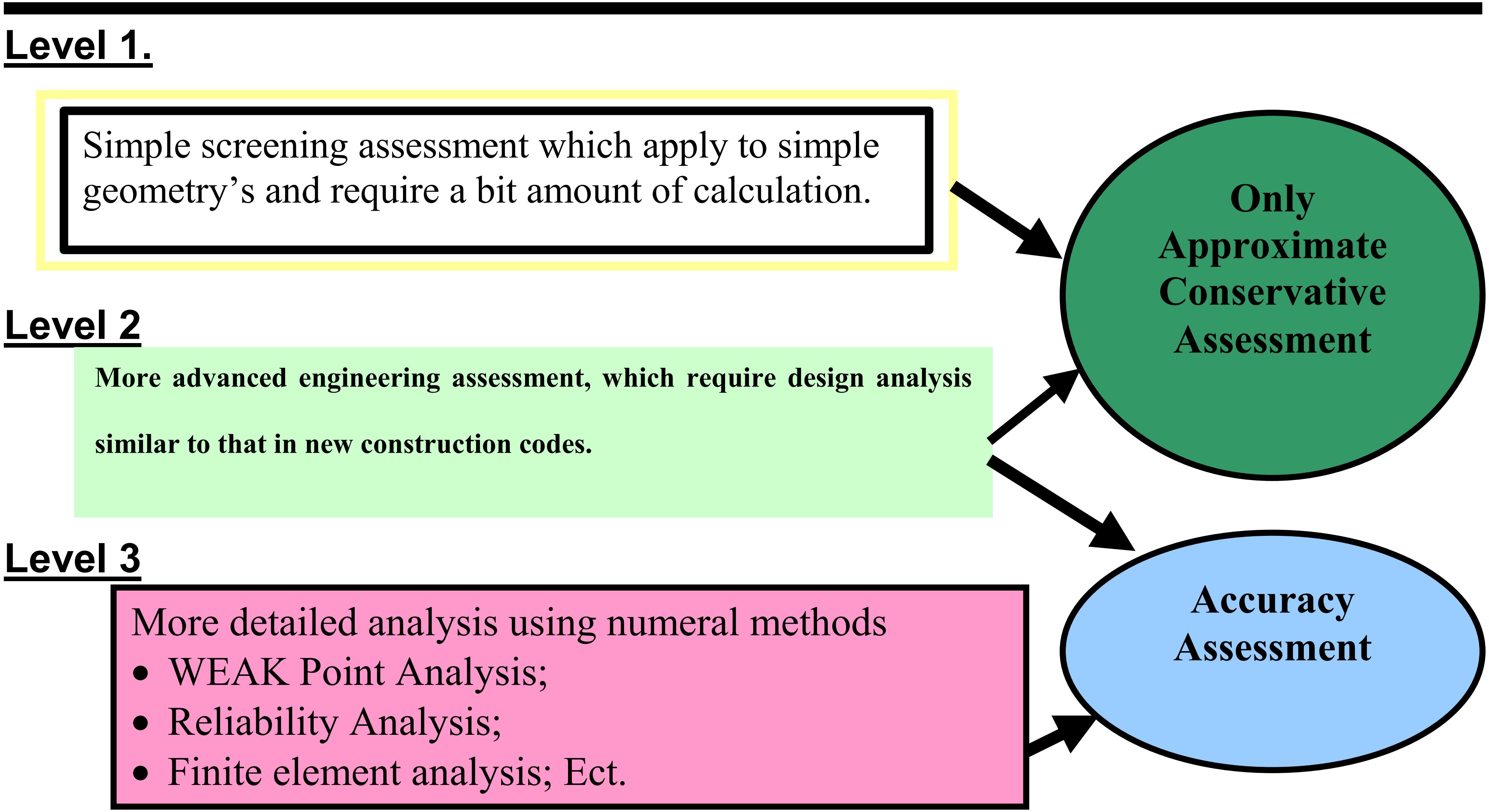

FIG.3 Methodology for different levels of the calculation lifetime

- Preventive maintenance service;
- Diagnostic service (inspection and NDE, test, monitoring);
- Collecting, systematization and archival storage of the information about individual parameters and characteristics of a history of operation NPP (OBD);
- Probabilistic analysis of safety of the equipment and other components NPP;
- Operating operations of staff;
- Water-chemical preparation;
- Other regulated programs on operation NPP.

FIG. 4 *General Measures on a Program Life Management*

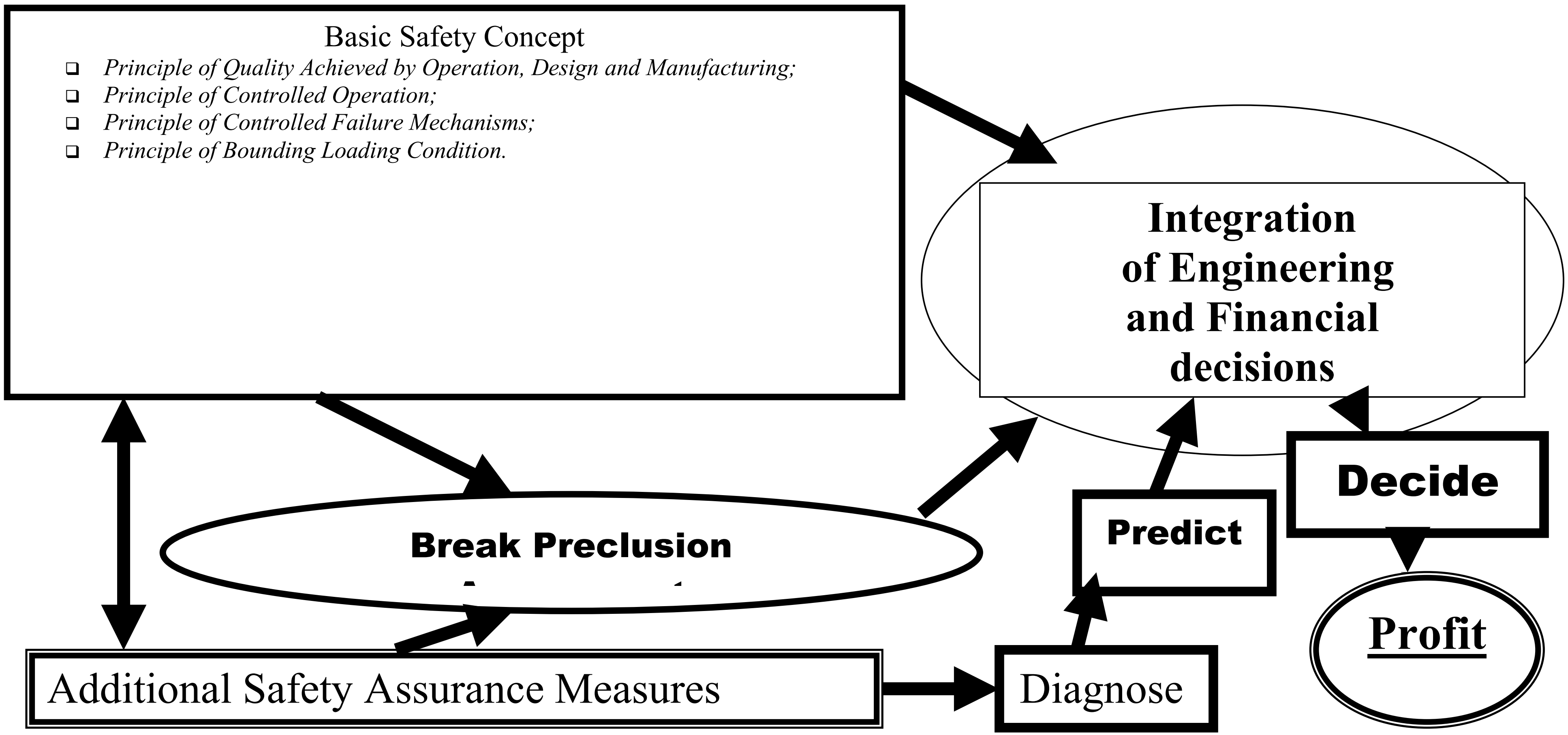

FIG. 5 Programme Life Management Recommended Optimum Concept

Failure defined as
Probability of Failure/Cost of Failure:

1

I. Event of Explosion;
II. Flammable Event;
III.Event of Damage;
IV. Radioactive Release;
V. Radioactive Damage
2 **V. Major Environmental Damage;**
VI. Event of Break;
VII. Event of Leak;
Public opinion;
Stadart; Low;
Etc.

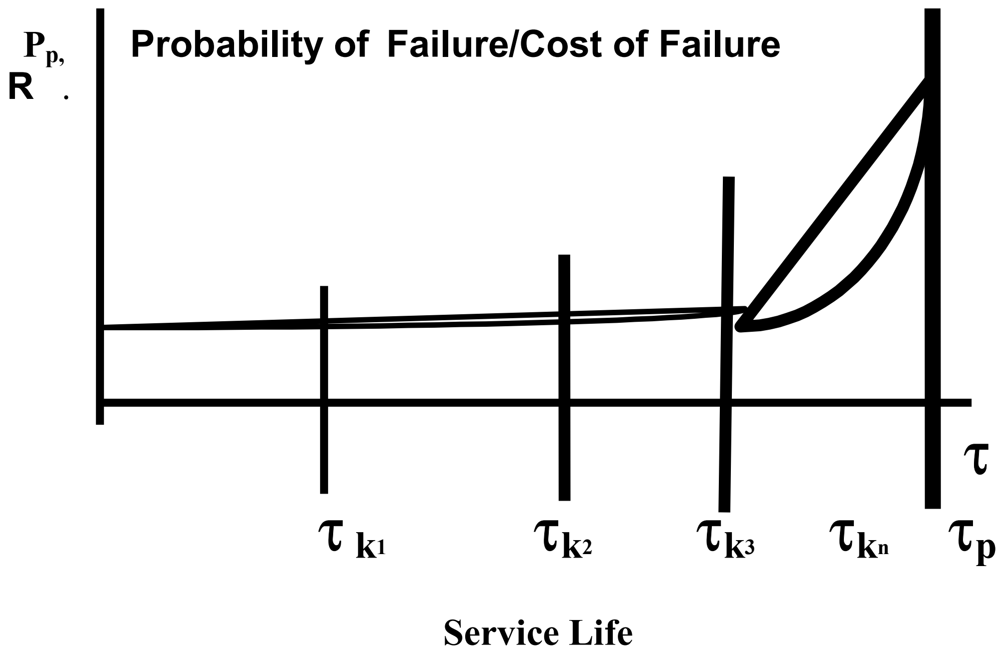

FIG. 6 Criteria for the Evaluation of Risk

RISK INFORMED IN-SERVICE INSPECTION AND TESTING IN SPAIN

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Abstract

In-service inspection (ISI) in nuclear power plant components provides assurance of structural integrity. ISI has a key role in NPP life management. Nowadays a revision of ISI scope is being considered. The accumulated experience demonstrates that, generally speaking, the inspected areas did not have relevant cracks and that the problems appear in areas outside the in-service inspection scope. As in other countries, in Spain, risk informed in-service inspection and testing programs are being applied to optimise them. Reduce the ISI scope without reducing safety levels is possible.

1. INTRODUCTION

In-service inspection (ISI) in nuclear power plant components has a double function. It provides assurance of structural integrity, but also it allows taking corrective measures before degradation becomes severe and structural margin is eroded. Therefore, periodic in-service inspection offers effective means of detecting the effects of the degradation mechanisms and to verify structural integrity.

NPP are committed to establish ISI programs to assess the structural integrity of their components to timely detect and prevent unacceptable degradation that may impair their safety functions.

This way it's cleared the key role of ISI in NPP life management.

Nowadays a revision of ISI scope is needed. The accumulated experience demonstrates that, generally speaking, the inspected areas did not have relevant cracks and that the problems appear in areas outside the in-service inspection scope.

More and more the regulation that governs the practices for O&M of NPP is considering criteria based on risk information.

The in-service inspection and in-service testing programs are also following this way and thus and as an example, in the case of Spain, risk informed in-service inspection (RI-ISI) and testing (RI-IST) programs are being applied.

Why?. To optimise in-service inspection. Optimise means to reduce inspection costs without impacting in the operational safety levels of the plant.

Is it possible?

Only as an example, in the Almaraz Unit 2 NPP, the initial scope of the ISI was 283 areas in class 1 piping (127 requiring volumetric inspection). After applying the RI-ISI methodology it becomes 71 areas (57 requiring volumetric inspection). And safety margin (core damage frequency) being 1.23 E-6 instead of initial 1.25 E-6.

2. RISK INFORMED IN-SERVICE INSPECTION & TESTING IN SPAIN

The Spanish nuclear regulatory authority, the Consejo de Seguridad Nuclear (CSN), requires the use of codes and standards in force in the country of origin of the plant technology.

For this reason, the in-service inspection and testing programs applied at Spanish nuclear power plants basically adhere to the requirements of the ASME XI and ASME OM Codes.

Concerning RI-ISI and RI-IST two clearly differentiated lines of work got under way:

- On the one hand, and as regards risk informed in-service testing programs, the American standards were used as the sole reference.

- In the case of risk informed in-service inspection programs, the Spanish nuclear power plant-owning utilities and the Consejo de Seguridad Nuclear decided to draw up a Spanish guideline that, although using the ASME developments as a reference, would have its own specific characteristics.

In relation to RI-ISI and RI-IST, the activities performed to date in Spain have been as follows:

- ALMARAZ NPP (PWR)

RI-ISI: Class 1 Piping of Unit 2 (At the end of licensing process)
Class 1 Piping of Unit 1 (Being developed)

- ASCO NPP (PWR)

RI-IST: Check Valves (Licensed)
Pneumatic Valves (In licensing process)

RI-ISI: Spanish Guideline for Piping (PWR Pilot Plant)
Class 1 Piping of Unit 1 (Licensed)
Class 1 Piping of Unit 2 (In licensing process)

- COFRENTES NPP (BWR)

RI-IST: Valves and Pumps (In licensing process)
RI-ISI: Class 1 Piping (Being developed)

- STA Mª GAROÑA NPP (BWR)

RI-ISI: Spanish Guideline for Piping (BWR Pilot Plant)

- VANDELLOS 2 NPP (PWR)

RI-IST: Check Valves (Being developed)
RI-ISI: Class 1 Piping (Being developed)

2.1 Risk informed in-service inspection Spanish guideline for piping

Taking into account the US developments regarding Risk Informed ISI, the Spanish utilities and the Regulatory Body have shown an increasing interest in any possible optimization of the ISI programs. With this framework, a pilot study on risk-informed ISI was arranged in a co-operative R&D project between the CSN and UNESA (Spanish Utilities Group).

The objectives of the project were:

- To define the principal characteristics of a suitable methodology in order to define a risk-informed ISI program for piping, using, as reference, the US NRC rules (R.G. 1.178, S.R.P. 3.9.8)

- Apply the developed methodology to a Spanish NPP, defining the scope of systems to be include in the program, and analyzing the different cases that could occur in order to define the necessary steps to be followed and to identify and solve the potential problems that could come up during the definition of a full scope RI-ISI program for piping.

- To define the minimal requirements for the documentation to be submitted and the basic steps of the CSN staff evaluation process in order to allow an agile implementation process for future applications.

The project had three main activities:

- The revision and analysis of all the applicable and available documentation, in order to define the framework for the project, for the Spanish approach. This activity finished in December 1998.

- The definition of a generic guideline, applicable to Spanish NPPs, for the establishment of an In-service Inspection program for piping using risk information and having as reference the US NRC Regulatory Guide and the experience of US pilot plant applications. The guideline has a main body with the following structure:

  - A general part defining the basic steps and principles of any Risk-Informed ISI program.

  - A methodological part defining the general approach to be followed taking into account both approaches (the quantitative and the qualitative one).

  - A documentation requirement part defining the documentation that should be maintained at the plant and the documentation that need to be sent to the Regulatory Body as the “Final Report”.

  - An Evaluation part which includes the process that has to be followed by CSN Staff in order to review and approve a RI-ISI program. All the activities that should be reviewed and the acceptance criteria for each of them are included in this part of the document.

- Attachments: One with the details of the quantitative methodology, other with the detail of the qualitative methodology and a third attachment with the type of report that has to be submitted. Each of these attachments develops the details of the general body of the guideline.

This guideline was issued in its first consensus draft in May 1999 and revised to include the lessons learned from the pilot studies. Final document was issued during first quarter of year 2000.

- The application of the guideline defined to one or two pilot plants. The objective of this activity was to check the technical consistency of the guideline.

The approach for the Spanish guideline is not to limit the scope of the program to only Section XI and to try to include all the ISI programs (ASME XI, IGSCC, ...) in place at the NPP, in a voluntary basis.

Other characteristic of the Spanish guideline is that it includes the technical approach and the evaluation process in the same document due to it has been developed under a co-operative action.

2.2 Risk informed in-service inspection for class 1 piping of Almaraz Unit 2 NPP

The most outstanding elements of the RI-ISI process are described hereafter.

Scope and definition of segments. The scope was limited to all Class 1 piping. The piping included within the aforementioned scope was divided into segments for which a failure at any point would have the same general consequences (86 segments were finally defined).

Identification of failure consequences. The consequences of failure of each segment were obtained from the PSA and were classified as direct effects (LOCA, reactor scram, loss of mitigation systems, etc.) and indirect effects (whiplash effect, jet impingement effect, flooding, etc.). Only direct effects are applicable to Class 1 piping.

Assessment of failure potential. The first step consisted of identifying the possible failure mechanisms for each segment: thermal stratification (6 segments), high cycle fatigue (19), water hammer (2) and without known degradation mechanism, except fatigue (59). The probabilities of failure of each segment were subsequently calculated using the WinPRAISE calculation program.

Risk calculation. The risk of a segment is calculated multiplying the failure probability of the segment by the consequences of its failure, expressed in terms of Core Damage Frequency (CDF).

Categorization by the experts panel. The revision by the Experts Panel of the calculations of risk, importance measures, sensitivity analysis and deterministic considerations gave rise to a categorization of the segments in relation to risk, containing 24 high safety significance (HSS) segments and 62 low safety significance (LSS) segments.

Selection of areas to be inspected. The areas of the HSS segments most likely to be affected by the postulated degradation mechanism were selected for overall inspection. For the rest of the HSS segments, a statistical selection method was used, based on the failure probabilities calculated using WinPRAISE. The selection of areas for inspection in the LSS segments was accomplished by sampling in segments of the same size and function.

In-service inspection program. The following table includes a comparison between traditional in-service inspection programs, in accordance with the requirements of ASME Code XI, and the new Risk Informed In-Service Inspection program, for Class 1 piping at Unit 2 of the Almaraz NPP.

	Class 1 ISI areas ASME XI	**Class 1 ISI areas RI-ISI**	**Percentage of Reduction**
Volumetric inspection (UT)	127	57	55%
Surface inspection	156	14	91%
TOTAL	283	71	75%

Re-evaluation of risk. New CDF was established in 1.23 E-6 instead of initial 1.25 E-6.

In concluding and looking into the figures, the interpretation of the results is both straightforward and direct: in addition to significantly reducing the number of areas to be inspected, a reduction in risk is achieved, as a result of which the optimization required by the CSN-UNESA Guideline is fulfilled.

2.3 Risk informed in-service testing program for check valves of Asco NPP

As mentioned before, the USA standards were use as reference.

This way, the ASCO NPP risk informed IST process, consists on the elements that follows.

Scope definition. For the RI-IST program, the scope of check valves to be analysed consists of all the valves included in the traditional IST program and all the valves included in the PSA. This scope includes valves not considered in the traditional IST program and that may be important to safety, particularly those modeled in the Asco NPP PSA but which are not currently within the actual test program.

Check valves risk ranking. This step of the process consists of a preliminary classification of the valves taking into account risk information obtained from PSA. The results of this risk classification brought 21 check valves classified as HSSC (5 check valves not included in the traditional IST program) and the others classified as LSSC. This last group includes 24 check valves to be classified by the expert panel because they are not included in the PSA scope.

Integrated decision-making: expert panel. Before establishment of the expert panel the activities considering development of expert panel administrative procedure, decision-making worksheets and training were performed. Once the expert panel was made up, the final classification of the check valves was established. 39 valves were classified as HSSC and 157 as LSSC. Within the HSSC group 2 valves are not included in the traditional ASME OM IST program.

Test strategies. The objectives of the test strategies differ depending on the safety classification of the valve. For the HSSC group of valves the objective is to identify and to trend the degradation that could lead to the failure mode that resulted in the HSSC. For the LSSC group of valves the objective is to verify the operational readiness of the valve. For doing this it is necessary to define a "performance based" program.

> *Ri-ist program for HSSC check valves.* Each valve will be full stroke tested with the frequency defined in the current program (3 months, cold shutdown or refuelling outage). For the check valves that are CIV or PIV, the results of the seat leak test will be used as an indicator of possible valve degradation. All the valves will be included in a disassembly and visual inspection program. This program will be performed on a sample basis in each group. All valves in a group should be tested in a 6 years or 4 refuelling outage period. In areas of high radiation zones and big valves, the use of non-intrusive diagnosis is an appropriate substitute for identifying degradation.

> *Ri-ist program for LSSC check valves.* Each valve will be full stroke tested with a 10 year frequency. For the check valves that are CIV or PIV, the results of the seat leak test will be use to analyze if the full stroke frequency should be modified. If there is any valve which safety function is the opening and it is not possible to perform a full stroke test, then a partial stroke will be performed and the valve will be included in a disassembly and visual inspection program.

Implementation and monitoring program. A sampling strategy is applied to determine if the scope of testing needs to be expanded to other similar check valves when unacceptable degradation is discovered. When a functional failure or unacceptable degradation is found, a failure mode, effects analysis is used to determine if other similar check valves, not included in the original sample, need to also be included in the test sample. For LSSC check valves, particularly those with extended test intervals, it is possible to take credit for normal plant operation and maintenance tests in the verification of the operability of the valve. There is a transition plan before extending test interval to maximum defined values in order to allow for experience to be gained before a component test interval may be extended too far. It is essential for the RI-IST process the use of a feedback process to validate the effectiveness of test methods and appropriately modifying the test strategy. For LSSC check valves where a functional failure has been found after the test interval has been extended, it is necessary to return the component to a shorter interval, such as each re-fuelling outage, and only extending he interval after acceptable performance has been achieved for 2 consecutive tests.

3. CONCLUSIONS

In Spain, both the NPP's and the regulatory authority have opted for the development of risk informed in-service inspection and testing programs as an alternative to those conventionally applied.

The cost reductions achieved through the implementation of these alternative programs, underlined by the pilot studies carried out in the United States and corroborated by those already performed in Spain, serve to promote these developments from the point of view of the plant owners.

The non-reduction of safety levels in operation of the plants, even with the implementation of these alternative programs, also make them acceptable to the regulatory authority.

IAEA-CN-92/P15

NEW CONTROL ROOMS IN SWEDEN

F. REISCH

Royal Institute of Technology
KTH
SWEDEN

International Conference on
Nuclear Power Plant Life Management
4 - 8 November 2002, Budapest

The participation in this Conference is supported by the Swedish Electrotechnical Commission and its committee for Nuclear Instrumentation, national mirror committee of Technical Committee TC 45, of the International Electrotechnical Commission, IEC.

Sweden

- Two nuclear power plants: Ringhals 2 (PWR, started in 1974) and Oskarshamn 1 (BWR, 1971) are in the process of modernizing their main control rooms.
- On Swedish initiative and under the direction of a Swedish project leader the IEC has prepared a report: IEC 62096, Nuclear Power Plants, Instrumentation and Control. Guidance for the Decision on Modernization.
- Many of the most severe accidents of the nuclear industry have occurred due to insufficient instrumentation leading to lack of important information about the status of the process and making the operators unable to make the correct decisions.
- These experiences have been applied during the modernization work.

Ringhals

- Ringhals 2 has chosen a step by step approach. At each step the operating personnel will carry out exercises and tests of the facility, give their opinion about it and suggest improvements.

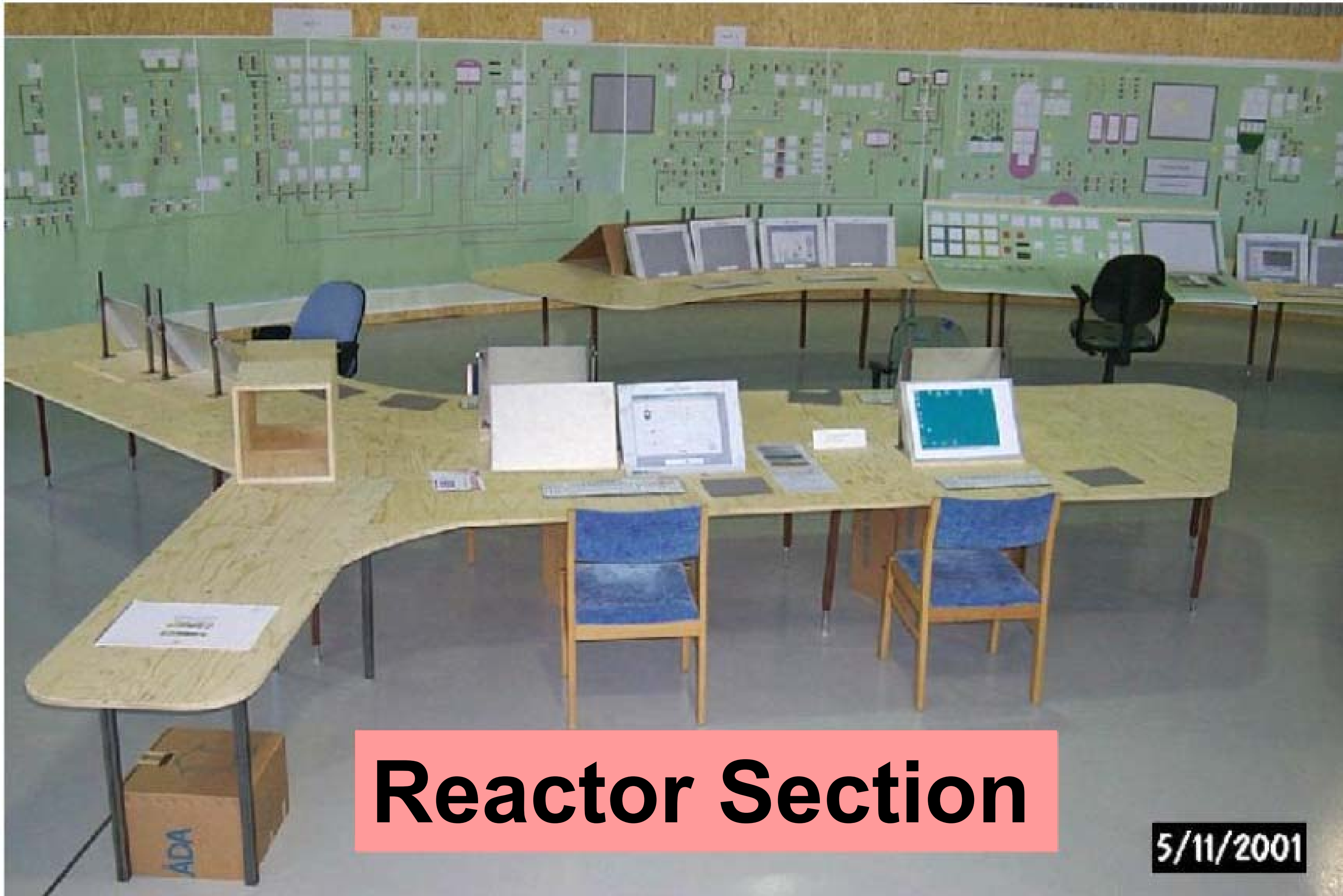
- The first step is a wooden model of the control room with drawings showing the instrumentation.

- The next step will be a simulator (to be completed in 2004).

- A year later, after extensive tests, the new control room will be installed.

- The newest state of the art computerized presentation will be applied.

- A hard-wired back up system with traditional instrumentation will also be in place for safe shutdown and cooling of the reactor.


Reactor Section
5/11/2001


Turbine Section
5/11/2001


Overview
5/11/2001

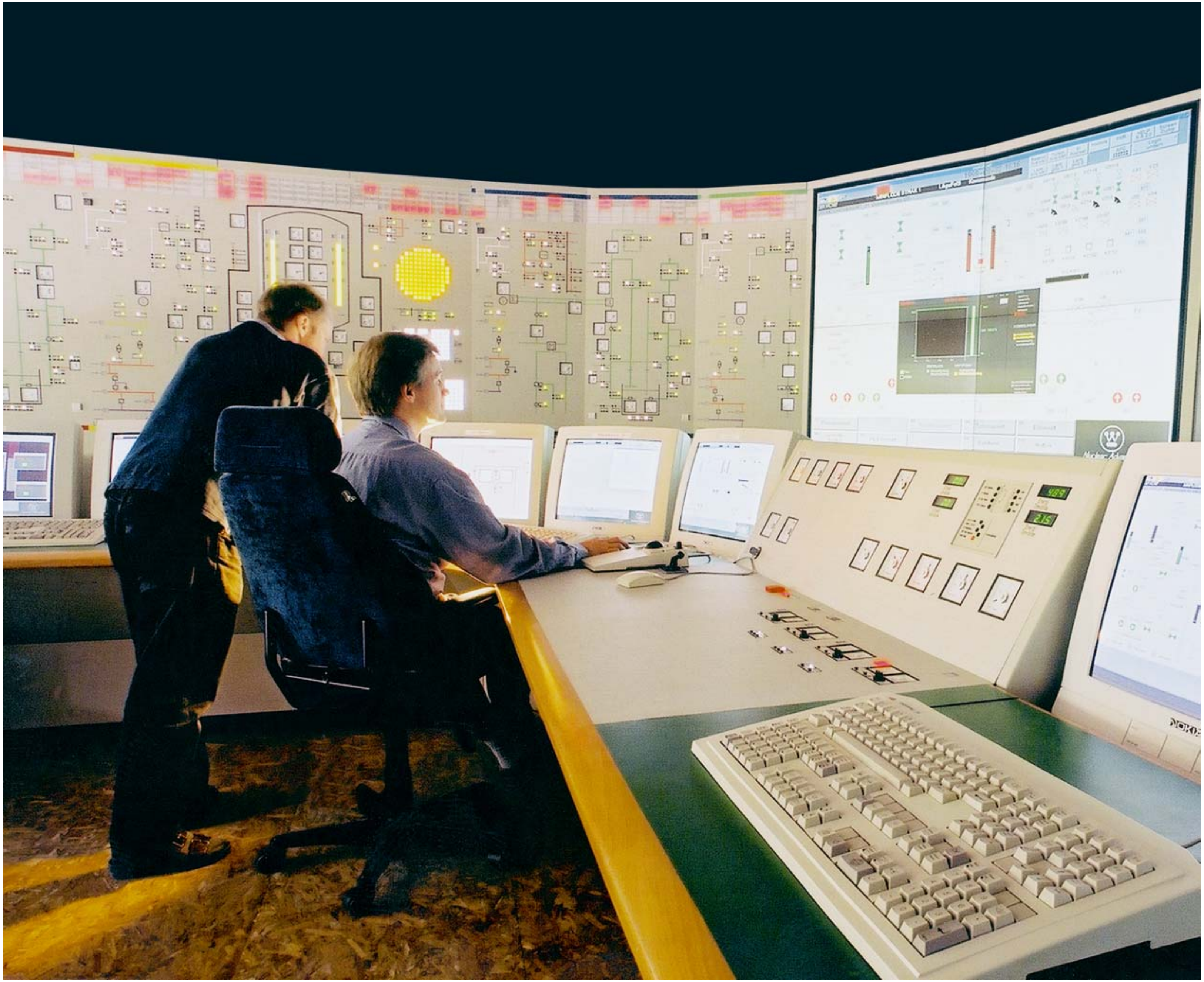
Electrical Section

Oskarshamn

To secure 20 years of future operation, Oskarshamn 1 will introduce:

- A safety concept with 4 train separated electrical power supply and I&C with physically separated cable routes.
- An electrical building for safety systems with two new diesel generators.
- A new emergency control room.
- A soft-ware based reactor protection system.
- A new control room design with safety panel, large screen and screen based operator interface.

In addition to the state of the art computerized presentation, there will be a hard-wired back up system with traditional instrumentation in place for safe shutdown and cooling of the reactor.



IEC 62096 – Nuclear Power Plants – Instrumentation and Control - Guidance for the decision on modernization - Technical report of Type 3

Scope and object

The purpose of this technical report is to support owners of an NPP in the decision-making process and preparation of partial or complete modernization of the I&C. For this it provides:

- a summary of the motivating factors for I&C modernization
- the principal options for the elaboration of different scenarios for I&C modernization
- the technical and economic criteria for the selection of a long term I&C strategy
- the principal aspects to be taken into account for a detailed technical feasibility study.

Special attention is paid to the improvement of the reactor safety and of the human machine interface.

The report does not provide I&C design requirements. For these it is assumed that the IAEA Codes and Guides are used as top level documents while IEC Publications mainly will be used for system design, requirements on equipment's and some work methods. IAEA Reports and other documents are referenced to give information that is more detailed on specific areas.

THE KORPUS FACILITY IN DETERMINATION OF RESIDUAL LIFE AND VALIDATION OF POSSIBLE PROLONGATION OF THE VVER-1000 VESSEL OPERATION BEYOND THE DESIGNED SERVICE LIFE

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FSUE “SSC RF RIAR”
Dimitrovgrad
RUSSIAN FEDERATION

Abstract

The paper presents the results of radiation embrittlement of base metal through the whole thickness of the VVER-1000 vessel under the nominal and increased neutron flux density and nominal temperature of the VVER-1000 vessel operation. These results were compared to the known ones of metal radiation embrittlement of the witness specimens from the Ukraine NPPs. The material science models are observed, by means of which the critical temperature change can be described; this change is insignificant ($\Delta T_f \sim 0$) at a neutron fluence F of (E >0.5keV) $F<2*10^{19}cm^{-2}$ and less than normative change at a neutron fluence of $3*10^{19}<F<11*10^{19}cm^{-2}$.

INTRODUCTION

The KORPUS facility [1, 2] was designed to irradiate vessel materials under conditions that meet all the contemporary requirements [3]. Before the facility development and nowadays irradiation of specimens is carried out in reactor channels (option 1), in consequence elements of the irradiation rig suspended behind the shaft and partitioning of the VVER-400 reactor (option 2), above the VVER-1000 reactor core (option 3) and in cavities located on the inner surface of the VVER vessel (option 4). Metal is tested, which was cut off from the VVER-1000 vessels operated during certain time including the design service life (option 5). These options have some disadvantages. The most important of them are:

- Limited capacity for location of specimens irradiated under identical conditions (options 1, 2, 3);
- Lifetime change of metal shielded with specimens (option 4);
- High radiation burden on the personnel and expenses on the vessel maintenance (option 5);
- Non-adequacy of the irradiation conditions in the energy spectrum and neutron flux density and in their distribution in the space including distribution through a set of specimens (options 1, 2, 3);
- Limitation in choice and maintenance of the irradiation conditions (options 1-5)
- Limitation in immediate and continuous condition control in the course of metal irradiation

The following conditions were specified in the facility design: range of the neutron flux density change is (E>0,5 MEv) 10^{13}-10^{10}cm^{-2} c^{-1}, error of the neutron fluence determination is ±15%, irradiation temperature range is 200 – 350^{0}C, error of the temperature assignment and determination is ±10^{0}C (confidence probability - p=0,95) and other Standard requirements [3].

1. BRIEF DESCRIPTION OF THE KORPUS FACILITY THIRD ASSEMBLING

During the last seven years (1995-2002) vessel material irradiation has been performed in the KORPUS facility third assembling (Figs 1, 2) in the modernized capsules (Fig. 3). There is a lead shielding (1+12+1mm, steel, lead, steel) located between the facility partitioning and the first row of capsules. Between the core, partitioning (steel, 3mm thick), shielding and capsules of the first, second, third and fourth rows there are water layers

4-5mm, 6-8mm, 4-6m, 5-12mm, 65-75mm and 5-15mm thick, respectively.

Up to five capsules with material science specimens, up to six methodical and four dosimetric capsules were installed in the cells of the first, second, third and fourth row during different loadings. Capsule locations in the row cells were determined by the experiment objectives and contract conditions.

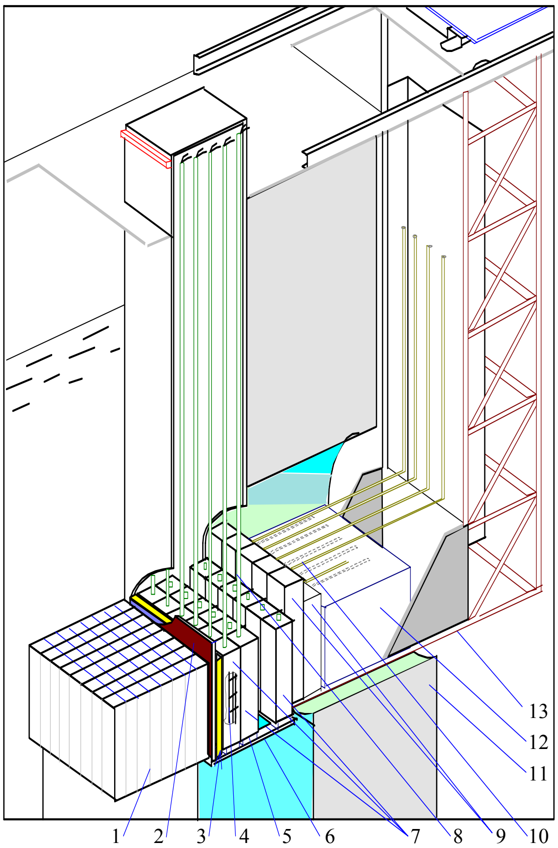


FIG. .1. – KORPUS facility layout:

1- core,

2 - partitioning,

3 - shielding,

4 - specimens,

5 – supporting plate to install a row of capsules,

6 - guides, along which supporting plates with capsules move,

7 – 1st type capsules with specimens,

8 – communication outlets выводы from the 1st type capsules,

9 – 2nd type capsules with specimens,

10 – communication outlets выводы from the 2nd type capsule

11 – concrete partitioning with a cylindrical niche,

12 – block for out-of-vessel space simulation,

13 – mobile platform

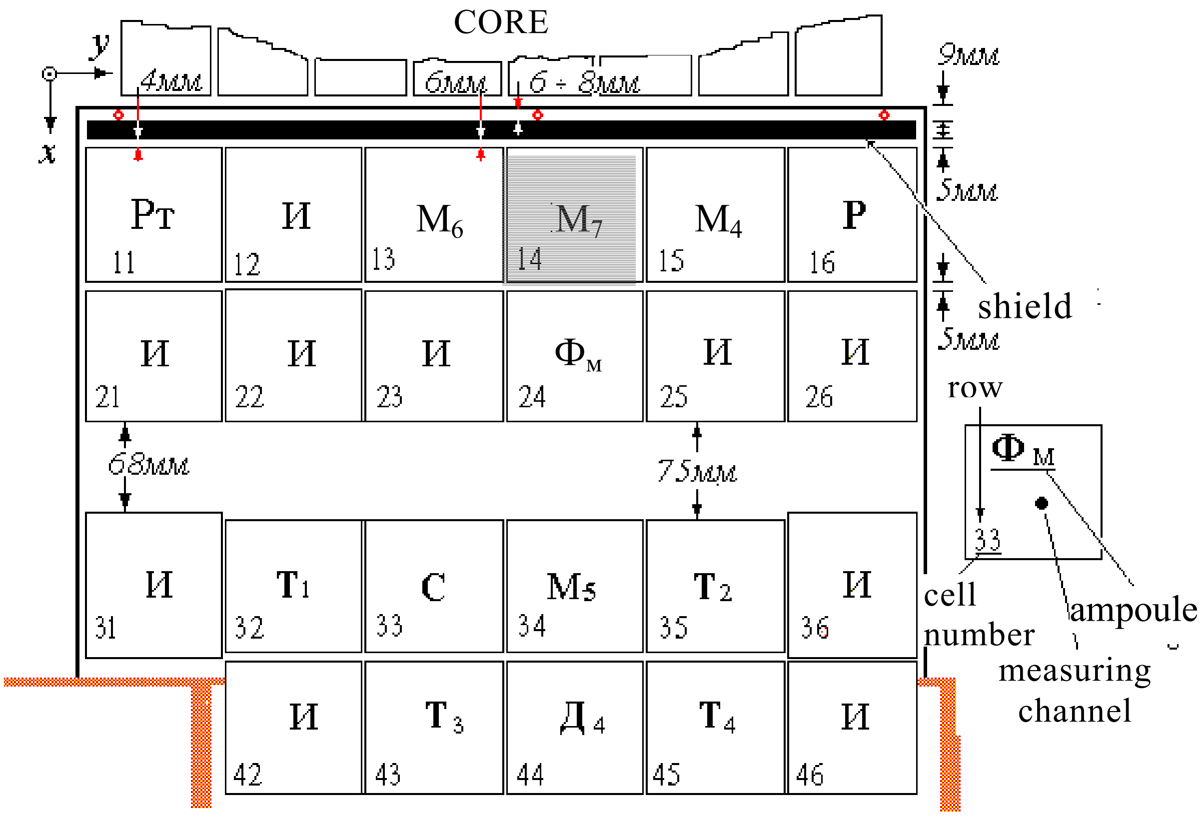


FIG. 2 - Capsule location in the KORPUS facility third assembling. M_1, M_2, $Ф_1$, $Ф_2$ – capsules with vessel material specimens; M_4, $Ф_M$, T_1 - T_4 – methodical capsules; C, P, PT, $Д_4$ – dosimetric capsules; 1 – core; 2 – facility partitioning; 3 – lead shielding; 4 - 7 – the first - fourth facility rows

.

A capsule of a new design (Fig. 3) differs from the ones tested previously in that the specimens are clamped between flat heaters located on steel plates. Three heaters are located on the specimen block side facing the reactor core. The other three heaters are located on the opposite side. Six or seven specimen layers (10*10*55mm) can be installed between the heaters. Six levels of specimens can be installed along the capsule height. Levels 1 and 6, 2 and 5, 3 and 4 are located symmetrically about the plane of capsule vertical symmetry (Fig.3 plane C-C). A measuring channel (inner diameter – 8mm) passes through the specimen array (from the capsule bottom). The channel is intended for the periodical control of temperature and neutron flux density (after reactor core reloading) as well as for routine determination of neutron fluence before dealing with capsules (rotation in a cell, replacement to another cell, unloading for annealing, completion of irradiation). At the 3d and 4th levels there are six (plane D-D) and ten (plane B-B) thermocouples in the cross-sections passing through Charpy specimen notches

Capsules with NAS can be installed in the cylindrical slots drilled in the plates located under the 3d level of specimens (plane E-E), between the specimens of the 3 and 4th levels (plane C-C) and above the 4th level of specimens (plane A-A).

In the period from 1995 to 2002 the facility technical conditions provided for performance of vessel material irradiation under contracts. Here:

- Reloading mode of the RBT-6 core was implemented allowing for a permissible change of neutron flux density during the contract execution; the possibility of neutron flux density increase by 10-12% was considered;
- 40 heaters were tested; eight heaters were operated under nominal conditions and eight ones were operated in the temperature control mode within 100-180 days; two heaters were operated under nominal conditions and two heaters were operated in the temperature control mode within 3,5 years; heater service life is 3,5,years;
- about 200 thermocouples were tested as well as the procedure for thermocouple inserting into the specimen array and cable output to data-measuring system DMS; one measuring line "thermocouple-cables in capsule" failed in the course of testing, the signal attenuated by ~15° at 265°C and by 60° at 474°C;
- The data-measuring system was certified; the main reduced error for measuring channels is (0,11-0,87)% in the temperature ranges (0-400)°C and (0-360) °C; DMS failures were eliminated and there was no any significant loss of information; some deviations in the signal value up to 5° were observed in several channels.

2. IRRADIATION CONDITIONS

Examinations of neutron filed in the KORPUS facility were performed in the periods from 1995 to 1996 and from 1999 to 2002. Table 1 presents average values of the neutron flux density in the centers of capsule cross-sections made by the horizontal plane of the RBT-6 reactor core for the capsules located in cell 13 (14), 23 (24) and 33 (34). Upper values correspond to the first measurement cycle and lower ones - to the second one. Deviations (%) are given under the neutron flux density values [2,4-6]. They are related to the reactor core reloading and taken into account during specimen irradiation.

Neutron flux density (E>0.5MEv) in the center of the first row capsule is 1.6 times more than the neutron flux density φ_{cc} on the surveillance specimens at a standard loading of the VVER-440 reactor core. The neutron flux density in the 2nd row is 3.5 times lesser than the neutron flux density φ_{cc} on the surveillance specimens at a standard loading of the VVER-440 reactor core. It is 3 times more than the neutron flux density φ_{cc} on the surveillance specimens at a non-standard loading of the VVER-440 reactor core and it is 5-6 times more than the neutron flux density in the VVER-440 vessel shell layers adjacent to the reactor core.

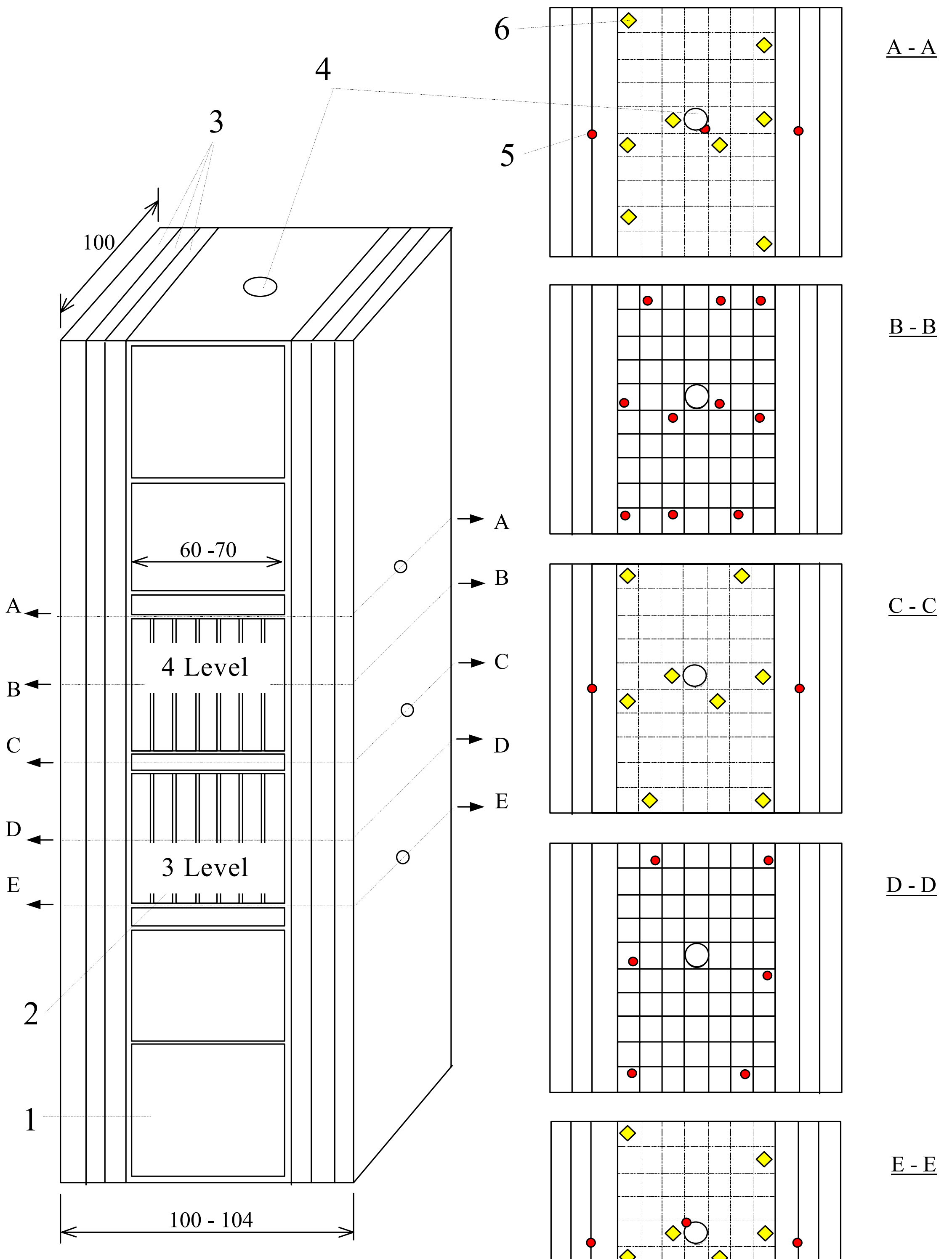


FIG. 3 – Location of specimens, heaters, thermocouples and neutron fluence monitors in the modernized capsule. 1 - blocks of capsules and simulators, 2 - specimens, 3 – heater plates, 4 – measuring channel, 5 - thermocouples, 6 – neutron fluence monitors

Table 1 – Neutron flux density in the center of the capsule measuring channels

Row	Neutron flux density, $10^{17}cm^{-2}day^{-1}$ / $10^{12}cm^{-2}s^{-1}$, With energy MEv			
	E>3,0	E>1,0	E>0,5	E>0,1
1	0,38 / 0,44, 0,44 / 0,508 +12%	2,3 / 2,7, 2,63 / 3,08 +13%	4,0 / 4,6, 4,7 / 5,46 +17%	8,1 / 9,4
2	0,042 / 0,049, 0,040 / 0,0463 -6%	0,38 / 0,44, 0,335 / 0,387 -13%	0,97 / 1,12, 0,72 / 0,83 -30%	1,5 / 1,73
3	0,0016/0,0019, 0,00183/0,00212 +13%	0,0139/0,0161	0,032/0,037	0,069/0,080

In the facility third row (cell 24) the neutron flux density in the capsule cross-section center is lesser than that on the capsule inner surface ($5,7{\cdot}10^{10}cm^{-2}s^{-1}$) and is equal to the neutron flux density at the ¼ of the VVER-1000 vessel shell thickness ($3,9{\cdot}10^{10}см^{-2}c^{-1}$)

The neutron flux density change in the operating capsule capacity is described by an empirical relation based on calculations and experiment results. In the course of irradiation the neutron flux density is maintained at the specified level for a long time.

The Standard requirement [3] "the neutron flux density change at the specimen locations necessary to determine the brittleness temperature at a specified neutron fluence value should not exceed 15%" is implemented on 16 specimens of i-layer on the 3d and 4th levelsin case of capsule irradiation in cells 14 and 13. The arrays satisfying the Standard requirements [3] in **Δ φ/φ<15 %** can be formed from the specimens irradiated in other cells.

The **neutron fluence** distribution **F(x, y)** through the specimen block capacity in the capsule corresponds to neutron flux density distribution tables 1 and 2) provided the capsule irradiation is performed in one facility cell. Capsule irradiation can be performed sequentially in the specified cell or in one cell with the capsule rotation by 180°around the vertical axis; thus, the fluence distribution along the specimen layer can be equalized. When the capsule is irradiated sequentially in position 0° and 180°, the most and the lowest neutron fluence values **E>3.0 MEv, E>1.0 MEv** and **E>0.5 MEv** in the specimen cross-section centers differ by 14%, 9% and 6%, respectively. Rot-mean-square deviations for a layer, column and the whole array of specimens do not exceed 0.8%, 6% and 65, respectively. According to Standard requirements [3], the specimen array (~110 pcs) irradiated at the 3d and 4th levels can be considered as "irradiated up to the specified fluence value". The error of the threshold fluence determination is ±9,5% at a confidence probability of p=0.95 (for 110-specimen array). The error of the neutron fluence determination (energy - E>05 MEv) is more or equal ±13,7%. Standard requirements on neutron fluence determination (E>05 MEv) are satisfied.

Gamma-quanta spectrum is calculated in the control points of the capsule capacity in the energy range 0-12MEv. Radiation energy release in iron is calculated on the basis of the gamma-quanta spectrum. The values of gamma-quanta flux density and radiation energy release are interpolated into the centers of the operating specimen cross-sections.

The temperature measurement is performed continuously at all control points during the whole irradiation period; one cycle takes 1 minute. 90-98% of the measurement results satisfie the specified condition. Root-mean-square deviations from the average temperature values do not exceed 2.9°C at the control points. The Standard requirement [3] on the temperature maintenance during irradiation $\Delta T < |\pm 5|^0C$ is satisfied. The difference in the temperature values of specimen blocks located at the 3d and 4^{th} levels does not exceed 3-5°C. According to Standard requirements [3], each specimen layer and the whole specimen array located at two capsule levels can be considered as an array irradiated at specified temperature $T_d - 10 \le T \le T_d + 10^0C$.

Table 2 presents the main characteristics of the irradiation conditions in the KORPUS facility. Also, the main Standard requirements on irradiation conditions are presented. The irradiation conditions in the KORPUS facility comply with standards and they are not worse than conditions, under which the main results were obtained on the radiation embrittlement of the VVER vessel materials. Here, the irradiation conditions are provided for large specimen arrays.

Table 2 – Comparison of Standard requirements with irradiation conditions at the KORPUS facility.

Parameter,	Parameter value	
	According to Standards	At the facility
1. Neutron flux density change,%:		
- on the specimen array for temperature determination $T_к$;	< 15	–
	–	3.7, 8.5,
- in the specimen layer in cells 14, 13, 15, 12;	–	18, 26
- through the specimen thickness, E >3.0, E >1.0, E >0.5 kv;	–	21, 16, 14
- at a control point within the irradiation period		< 15
2. Error of neutron fluence (transfer) determination, %:		
- with energy E >3.0, E >1.0 MEv;	–	< $\lvert\pm10\rvert$
- with energy E >0.5 MEv	< $\lvert\pm15\rvert$	< $\lvert\pm14\rvert$
3. Deviation of temperature from the specified one under stationary irradiation conditions °C:		
- for specimen array with $T_{mass.}$265, 295°C;	≤ 20	≤15 и 20
- at each control point	< $\lvert\pm10\rvert$	≤ $\lvert\pm3\rvert$
4. Error of sensor readings during recording the irradiation temperature, °C.	±5	≤ $\lvert\pm2,5\rvert$
5. Error of average absolute temperature value determination within the irradiation period in cells 13, 14, 12 and 15, °C, for:		
- specimen,	–	≤ 5 -6
- layer of 18 and array of ~ 110 Charpy specimens.	–	≤ 8 - 8,5
6. Specimen quantity	≥ 12	16 - 20

The period of specimen irradiation at the reactor power close to the nominal one is
t_N = (0,95 - 0,98) · $t_{designed}$ of designed irradiation period t_d. The irradiation period in the specified temperature mode is t_T = (0,95 - 0,98) · t_N In the course of contract execution the reactor operation period at a nominal power is ~0.87 of the calendar irradiation period. The designed irradiation period implemented at the facility is 3.5 years.

The irradiation conditions are described in more detail in the proceedings of Inter-branch Conference on Reactor Material Science.

3. INCREMENTAL ELEMENTS OF ERROR FOR CRITICAL BRITTLENESS TEMPERATURE DETERMINATION

According to the known results [6], 9 error compounds can be distinguished. The main icrement is the value spread $T_{critical}$ through the main and auxiliary capacity of the shell base metal: from <15^0C up to <25^0C for steel 15X2HMФA(A). The increment of each irradiation parameter (fluence, irradiation temperature, neutron flux density, irradiation period) to the error is lesser than $(4 - 8)^0$C and can be compared to the error methodical compound ~6^0C including three incremental elements.

The error of the brittleness temperature determination on a series of irradiated and non-irradiated specimens includes four compounds and it is up to (16-25) °C for steel 15X2HMФA(A). The determination error of the critical brittleness temperature change $\Delta T_{critical}$ can be estimated equal to (9 - 12) °C provided the irradiated and non-irradiated specimens were cut off from the volume of one shell.

It is possible to determine effectively the coefficient of empiric functions used for prediction of vessel material embrittlement using a large quantity of the irradiated specimens or specimens cut off from the reactor vessel. Irradiation of hundreds of specimens can be performed at the facility under conditions complying Standard requirements [3].

The KORPUS facility has all necessary conditions to reproduce the radiation-damaged sate of metal to the extent that the property change, which occurs due to the task error, maintenance and determination of the irradiation temperature, fluence and neutron flux density, does not exceed the error of the brittleness temperature determination for the irradiated and non-irradiated specimens.

4. THE KORPUS FACILITY IN EXAMINATION OF THE VVER VESSEL METAL

4.1. Radiation embrittlement of base metal of the VVER-1000 standard shell through the whole reactor vessel thickness at the increased density of the reactor emission flux.

The experiment was carried out to simulate the radiation damage of base metal through the whole thickness of the VVER-1000 vessel standard shell. The simulation was provided by:

- specimen cut off from base metal (steel 15X2HMФAA) of the standard shell
- specimen location in the 1^{st} and 2^{nd} row capsules of the KORPUS facility (the relative specimen location in the shell was retained)
- irradiation at a nominal temperature(295 ± 15^0C) of the VVER-100 vessel operation
- retained relative change of the neutron flux density through the reactor vessel shell.

For the specimens located in the 1st and 2nd row capsule the neutron flux density (E>0,5MEv) is (7,2-3,5)·1012cm-2s-1 and (1,5-0,7)·1012cm-2s-1 and radiation energy release - (0,80-0,30)·10-1W/year and (0,15-0,08)·10-1W/year. The blow-bending tests were carried out fo rthe specimens. The critical brittleness temperature $T_{critical}$ was determined.

Experimental values $T_{critical}$ (large points) are lesser than values $T_{critical}$ allowed by Standards for strength calculation (Fig. 4) (upper line, model 1). The critical brittleness temperature increases more rapidly than normative relation parameter T_0. The greatest radiation brittleness coefficient AF=13,80C determined on the basis of the results for the 4th layer (the distance from the inner shell surface is ~50mm) is lesser than the normative one. The pessimistic service life estimations according to neutron fluence exceed the designed ones.

The lower line (Fig.4a) is plotted by all the experimental points and normative dependence. It deviates up to $20\text{-}40^0C$ from the experimental values $T_{critical}$. Values $T_{critical}$ for the specimens irradiated in the facility 2^{nd} row deviate up to 50^0C form the central line (Fig. 4a) plotted by the normative dependence and values $T_{critical}$ for non-irradiated specimens and specimens irradiated in the facility 1^{st} row. Deviations exceed errors.

Values $T_{critical}$ for the specimens irradiated in the 1^{st} and 2^{nd} rows can be interpolated by the normative dependence, on introducing a shift along the axis $T_{critical}$ relative to values $T_{critical}$ for **non-irradiated** specimens (lower line is for low fluence values (Fig.4b)). The shift can reflect a thermal process that proceeds with time and decreases the critical brittleness temperature up to -110^0C (model 2)

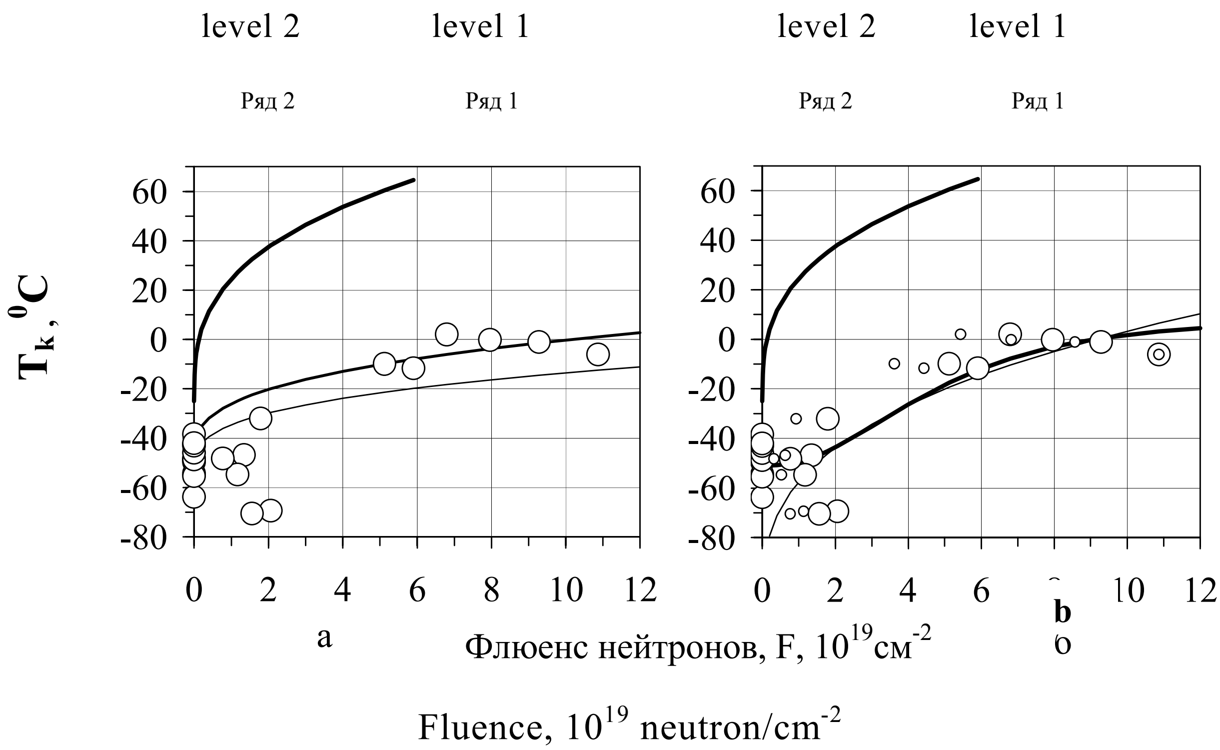


FIG. 4 – Change of critical brittleness temperature of the VVER-1000 vessel base metal (standard metal, steel 15X2HMFA, as a result of irradiation in the 1^{st} and 2^{nd} rows of the RBT-6 reactor KORPUS

The experimental results for non-irradiated specimens and specimens irradiated in the 1^{st} and 2^{nd} row of the facility can be interpolated by the relation describing the substitution of initial state T_{c0} with limiting one T_{cl} at n-fold repetition (n=3–5) or concurrence of damaging events (model 3) [6]. The effective neutron fluence (E>0.5MEv) is F_0=(2.9-2.1)*10^{19} $cm^{-}2$ The rate of T_c change tends to zero at low (Fig. 4b and 5) and high (Fig. 4b) fluence values.

The change of critical brittleness temperature can be described by relations, which present the radiation damage and radiation annealing of metal using neutron flux density φ^n (E>3; 1; 0.5 MEv) and radiation energy release q^m when n≥1 and m≥1 (model 4). The change of average irradiation temperature from layer to layer is not included into regression relations. The factor effect is slight as compared with the spread of the experimental result.

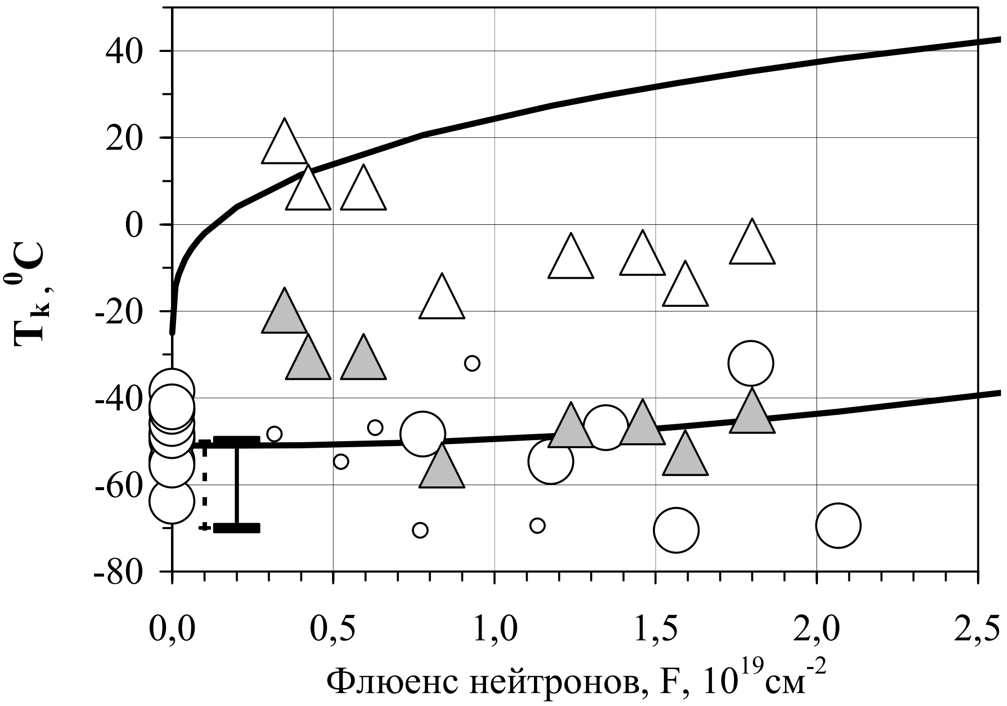
Figure 5 presents the results (results 1, circles) that are compared with those obtained on surveillance specimens from the Ukraine NPPs [7] (results 2, triangles). Results 2 belong to guaranteed value T_{c0}= –25^0C (unshaded triangles) and to value T_{c0}= – 64^0C obtained on the non-irradiated specimens of the first inner layer of the shell (shaded triangles).

Surveillance specimen changes of T_{c0} ±12^0C have the same range as the critical temperature change through the shell thickness in the initial state (from –43°C up to -64°C) (Fig. 5, fluence is equal to zero). Changes of T_{c0} around the perimeter and along the height of the shell exceed 5-10°C. Results 1 and results 2 are spread equally relative to average ones (confidence probability - p=0,95): T_c= -50,2±10,0^0C and T_c= T_{c0} + 23,6±12,3^0C. There are no releases at p=0,95. Only one value T_c [7] exceeds the corresponding (according to fluence) normative value when T_{c0}=-25^0C. At the lower critical brittleness temperature all the values [7] are lower than normative ones.

Results 1 and 2 can be described by a linear (three-dimensional) relation (model 5)

$$T_c=T_{c0}+a_0+a_1\cdot F+ a_2\cdot\varphi+ a_3\cdot t,$$

Where F, φ, t – fluence, neutron flux density and irradiation period; a_0 – a_3 – parameters. T_{c0}= - 48,5^0C was taken for results 2 as well as for results 1. The irradiation period is the most significant factor. The result description can be better if the neutron flux density and fluence are included into the relation. The radiation energy release increment was not estimated due to the absence of its value in this work [7]. We can assume that theincrement of the period and temperature of irradiation contribute is the most important. Here, the neutron flux density (and radiation energy release) increment is negative.



Fluence, 10^{19} neutron/cm^{-2}

FIG. 5 -– Change of critical brittleness temperature of the VVER-1000 vessel base metal (, steel 15X2HMFA) as a result of irradiation at the KORPUS facility (circles) and of the surveillance specimens from the Ukraine NPPs [7] (triangles). Lines: upper line – guaranteed values according to normative relation (model 1); lower line is plotted based on the relation describing T_c change at n-fold repetition or concurrence of damaging phenomena (model 3).

Fig. 4a presents experimental values T_c (small points) versus neutron fluence $F_{3,0}$ with energy E>3MEv, of which values are multiplied by spectral index g(0,5/3,0) for the 1st layer specimens of the facility first row capsule. As for this layer, calculated neutron fluence value $F_p=F_{3,0}(x)\cdot g(0,5/3,0)$ and experimental one $F_{0,5}$ (energy - E>0,5MEv) coincide and are equal to $10,7\cdot10^{19}$cm^{-2} . With distance from the core, neutron fluence with energy E>3,0MEv decreases more rapid than that with energy E>0,5MEv. Experimental values T_c displace along axis F to the area of lower values by value $\Delta F_{0,5}$~$F_{0,5}(x)\cdot(1\text{-EXP}(0,005\cdot x))$. The dependence form tends to the normative dependence one at fluence values from $1\cdot10^{19}$cm^{-2} up to $6\cdot10^{19}$cm^{-2}. The normative dependence is plotted based on the results of specimen irradiation, mainly at values g(0,5/3,0)~9 that are typical for the research reactor experimental channels. The better conformity of the experimental dependence form (the form of the curve) $T_к=T_{к0}+f(F_{3,0})$ with the normative dependence one can mean that the radiation embrittlement of pure base metal is determined by radiation damages created by high energy neutrons (model 6). In the area of low neutron fluence values $F_p<1\cdot10^{19}$cm^{-2} ($F_{3,0}<10^{18}$cm^{-2}) (Fig. 5a) this model as well as model 2 predicts low values of the critical brittleness temperature up to $T_к(F=0)$~ -(80-90)^{0}C at a fluence tending to zero.

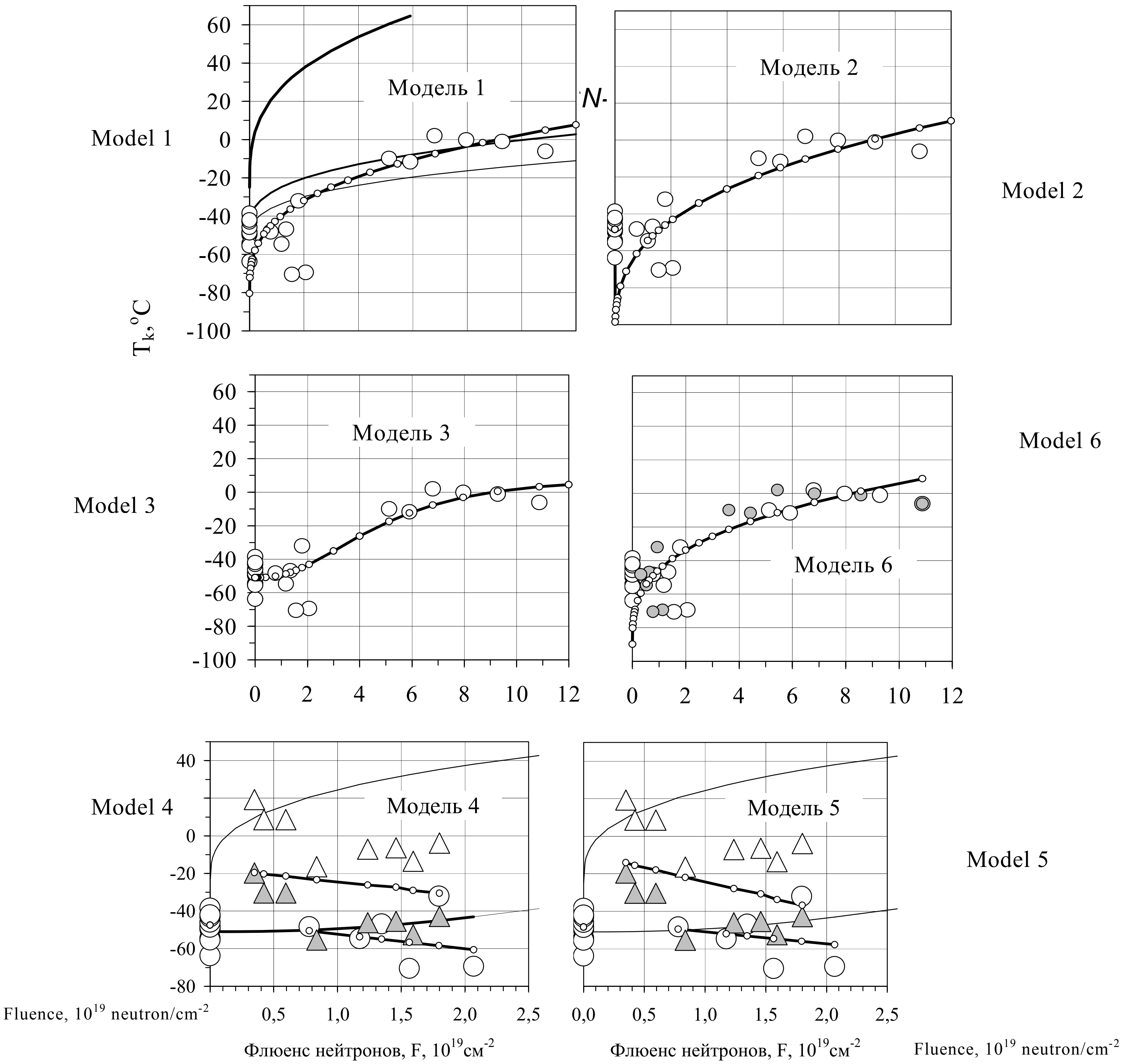


Рис. 5а. Сравнение моделей, интерполирующих изменение критической температуры хрупкости в результате облучения образцов основного металла корпуса ВВЭР-1000 в стенде КОРПУС и образцов-свидетелей в Украинских АЭС.

FIG. 5a. The comparison of results for base metal of VVER-1000 vessel derivable in the KORPUS facility with results of surveillance program of the Ukraine NPP

4.2. Radiation embrittlement of base meta layersl of the VVER-1000 standard shell at the nominal density of the reactor emission flux.

The first five metal layers were irradiated in the facility 3^d^-row capsule within 3.5 years. These layers are adjacent to the shell inner surface (1/3 of the reactor vessel thickness) and correspond to five metal layers irradiated at the increased flux density in the facility 1^st^-row capsule. The irradiation temperature is $286{,}5\pm4{,}5^0$C; the neutron flux density is $(5{,}3\text{-}2{,}5)\cdot10^{10}$cm^{-2}s^{-1} (energy - E>0,5MEv) and $(2{,}9\text{-}1{,}0)\cdot10^{17}$cm^{-2} s^{-1}

(energy - E>3MэВ); the neutron flux density for the 4^th^ layer is $3{,}1\cdot10^{18}$cm^{-2}s^{-1} (energy - E>0,5MэВ) (Table 1); radiation energy release is $(0{,}35\text{-}0{,}08)\cdot10^{-2}$W/year.

The preliminary results (Fig. 5; value range is indicate by a vertical line) show that the critical brittleness temperature of the standard shell base metal, which was used to certify the standard process of the VVER-100 reactor shell fabrication, changes insignificantly. According to normative relation ($A_F=23^0$C) we might expect that the critical brittleness temperature change would be ~33^0C. The relation of radiation energy release to neutron flux density for the operating capacity of the 3^d^-row capsules is more than that for the operating capacity of the 1^st^ and 2^nd^-row capsules. The insignificant change of the critical brittleness temperature, as compared with the allowed normative change, can be a feature confirming model 4 (radiation gamma-annealing model) or model 3 (model of T_c change at the n-fold damage of the metal local area) or model 6 (model of T_c change on accumulating the radiation damages created by high energy neutrons).

The model ranking and increment evaluation of the irradiation condition components will be performed according to the test results of the specimens irradiated within 3.5 years under conditions close to operation conditions of the VVER-1000 vessel metal.

4.3. Radiation embrittlement of base metal, weld metal and heat-affected zone metal of the VVER-1000 reactor vessel - Test of normative relations

Examinations were carried out under the contract with CAE. The specimens were made from base metal and weld metal (steel 15X2HMФAA) from the 4^th^ block of the Kalinin NPP. Weld metal was taken from weld No4; base metal – from the shell under weld No4. Diagrams of base metal and weld metal sampling through the vessel thickness are given in Fig. 6, 7. Charpy specimens (10×10×55mm) and tensile test specimens are fabricated according to requirements RCC-M MC 1000, GOST 6996-66 and GOST 9454-78. In addition, heat-affected zone specimens are fabricated. The chemical composition of metal is determined. Seven specimen arrays are irradiated and tested. The specimens were irradiated in the 1^st^-row capsule. They were displaced into cells 13 and 14 and the capsule was rotated to equalize neutron fluence through the specimen block.

Contract specifications on the irradiation temperature and neutron fluence (E>0,5 MEv) were satisfied:

- Nominal temperature *289,7^{0}C* within the whole irradiation period comply with value 290±15^{0}C.
- Average temperatures of each specimen within the whole irradiation period are in the range 275<281,7<290<294,0<305^{0}C.
- The average values of irradiation temperature are in the interval <100C (at irradiation substages).
- Root-mean-square deviation at each control point does not exceed 2,7^{0}C<5^{0}C within the whole irradiation period.
- Average neutron fluence value is $6{,}6\times10^{19}$ cm^{-2}.
- - Deviation of particular fluence values from the average ones does not exceed <10%.

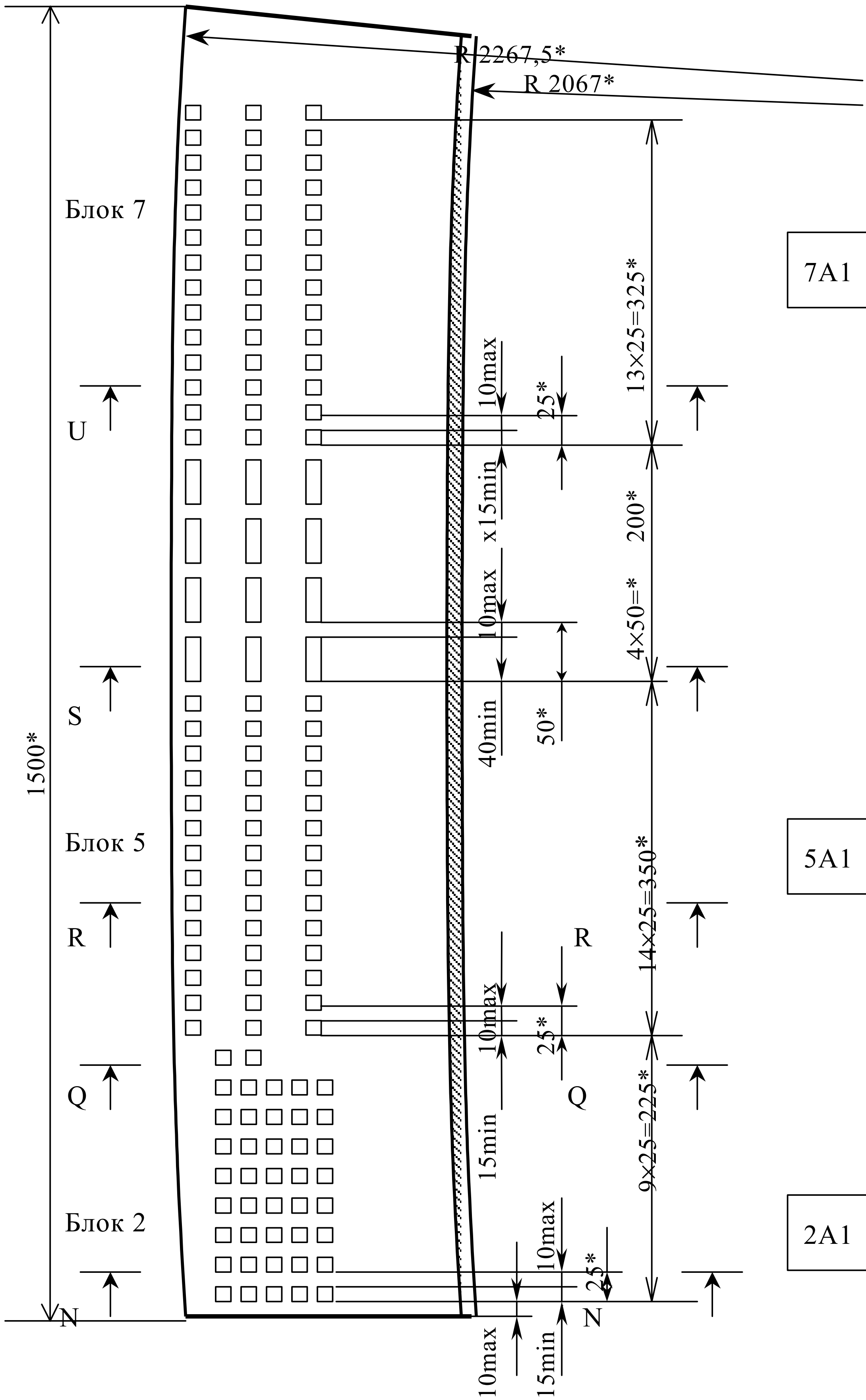


FIG. 6 – Weld horizontal cross-section V–V. Places of specimen cut off according to RCC-M (2A1, ...) and GOST 6996-66 (5A1,...; 7A1,...).

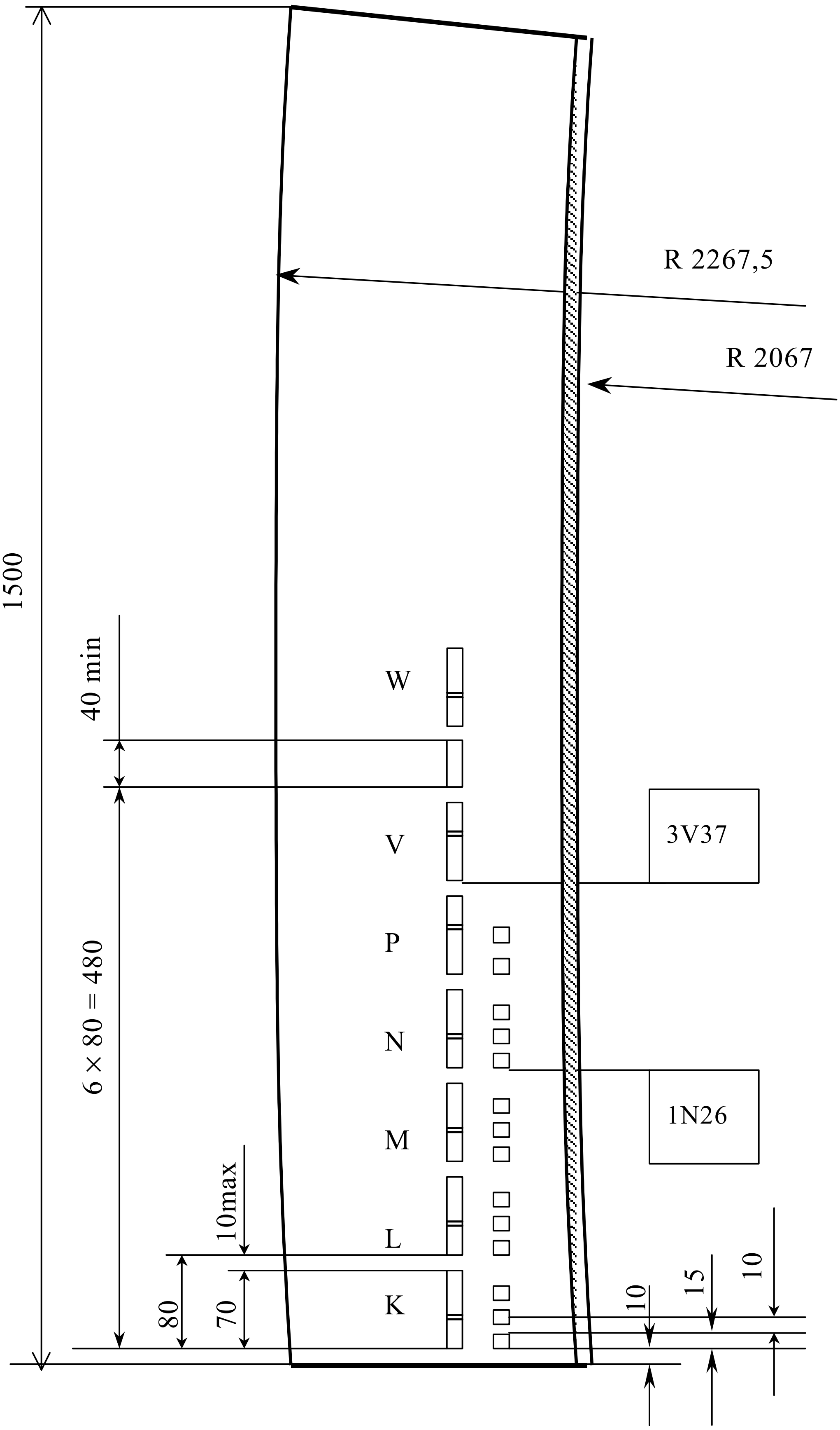


FIG. 7. Horizontal cross-section (under the weld cross-section). Places of specimen cut off from base metal according to RCC-M (1N26,..) and GOST 9454-78 (3V37,..). Areas of specimen cut off K, L, M, N, P, V.

It follows from the test results of irradiated and non-irradiated specimens that the critical brittleness temperature of the irradiated specimens does not exceed the guaranteed values. These values are calculated on the basis of the guaranteed values of the critical brittleness temperature in the initial state and of the radiation embrittlement coefficient for base metal, weld metal and heat-affected zone metal.

4.4. Radiation embrittlement of the VVER-1000 vessel weld metal at increased reactor emission flux

Specimens subjected to irradiation were fabricated from weld metal layers (Fig.6) adjacent to the inner surface of the Kalinin NPP 4th block reactor vessel. The specimen layer location in a capsule corresponds to their location in the reactor.

As for the material science purpose (service life determination of all vessel layers and of the vessel through the whole weld thickness), the experiment corresponds to the earlier one carried out for base metal in the facility 1st-row capsule (4.1.)

The capsule is irradiated in cell 14 without any displacements and rotations. Nominal temperature ***288,5 ^{0}C*** within the whole irradiation period corresponds to value 290 ± 15 ^{0}C. Average temperature values of each specimen are in the range **282,5** < 290 < **298,8** ^{0}C within the whole irradiation period. The root-mean-square deviation at each control point does not exceed **2,3 ^{0}C** < 5 ^{0}C within the whole irradiation period.

Average neutron flux density (E > 0,5 MEv) in the central layer is $(5,06 \pm 0,51)\cdot 10^{17}$ cm^{-2}day^{-1}. The neutron flux density values (energy - E > 0,5 MEv) can be estimated as $6,52\cdot 10^{19}$ cm^{-2} according to NAS measurements in the central channel of capsule M8 at the core central plane.

Neutron fluence values (energy - E>0,5 MEv) in the specimen cross-section centers are calculated according to Quality Assurance Program and on the basis of the neutron fluence value in the measuring channel center. Correlation coefficient r(F,T) between fluence values and irradiation temperatures is 0.144. Measurements of NAS located in the specimen blocks are being completed.

The calculation determination of gamma-emission field and radiation energy release distribution was performed. The radiation energy release distribution through the specimen cross-sections as well as distribution of the irradiation temperature and neutron fluence is given in Fig.8. Radiation energy release distribution correlates with distribution of neutron flux (and fluence) density r(q, F) = 0,989 and does not correlate with specimen irradiation temperature distribution r=(q,T)=0,12. When analyzing the result concerning the damage energy of the irradiated specimen, the attempt will be taken to mark out the components characterizing the increment of fluence, radiation energy release and irradiation temperature.

4.5. Radiation embrittlement of weld metal of the VVER-reactor vessel at increased reactor emission flux

Specimens subjected to irradiation were fabricated from weld metal layers (Fig.6) adjacent to the inner surface of the Kalinin NPP 4th block reactor vessel. The specimen layer location in a capsule corresponds to their location in the reactor.

The specimens for the experiment irradiation at the nominal and increased density of reactor emission flux were taken from the corresponding layers (either even or odd). They represent "the unique" weld metal capacity. As for material science purpose (service life determination of all vessel layers and of the vessel through the whole weld thickness), the experiment corresponds to the earlier one carried out for base metal in the facility 3d-row capsule (4.2.)

The capsule is being irradiated in cell 34 without any displacements and rotations. The average irradiation temperature of the specimen block is ~285-290^{0}C.

		1,47	1,47	1,46	1,46	1,46	1,45	1,44	1,43	1,42	1,41	A
		1,27	1,27	1,27	1,27	1,27	1,26	1,25	1,25	1,24	1,23	B
		1,11	1,11	1,11	1,10	1,10	1,10	1,09	1,08	1,08	1,07	C
	4 этаж	0,96	0,96	0,96	0,96	0,96	0,95	0,95	0,94	0,94	0,93	E
Level 4		0,84	0,84	0,84	0,83	0,83	0,83	0,82	0,82	0,81	0,81	H
		0,73	0,73	0,73	0,73	0,72	0,72	0,72	0,71	0,71	0,70	P
		0,63	0,63	0,63	0,63	0,63	0,63	0,62	0,62	0,61	0,61	Z

а

a

		1,54	1,54	1,54	1,53	1,53	1,52	1,51	1,51	1,49	1,48	A
		1,34	1,34	1,34	1,33	1,33	1,32	1,32	1,31	1,30	1,29	B
		1,16	1,16	1,16	1,16	1,16	1,15	1,14	1,14	1,13	1,12	C
	3 этаж	1,01	1,01	1,01	1,01	1,00	1,00	0,99	0,99	0,98	0,97	E
Level 3		0,88	0,88	0,88	0,88	0,87	0,87	0,86	0,86	0,85	0,85	H
		0,76	0,76	0,76	0,76	0,76	0,76	0,75	0,75	0,74	0,74	P
		0,66	0,66	0,66	0,66	0,66	0,66	0,65	0,65	0,65	0,64	Z

		0,100	0,100	0,100	0,099	0,098	0,097	0,096	0,095	0,093	0,092	A
		0,082	0,081	0,081	0,081	0,080	0,079	0,078	0,077	0,076	0,075	B
		0,068	0,067	0,067	0,067	0,066	0,065	0,065	0,064	0,063	0,062	C
Level 4	4 этаж	0,057	0,057	0,056	0,056	0,056	0,055	0,054	0,054	0,053	0,052	E
		0,049	0,049	0,048	0,048	0,048	0,047	0,047	0,046	0,045	0,045	H
		0,043	0,042	0,042	0,042	0,042	0,041	0,041	0,040	0,040	0,039	P
		0,038	0,038	0,037	0,037	0,037	0,037	0,036	0,036	0,035	0,034	Z

б

b

		0,100	0,100	0,100	0,099	0,098	0,097	0,096	0,095	0,093	0,092	A
		0,082	0,081	0,081	0,081	0,080	0,079	0,078	0,077	0,076	0,075	B
		0,068	0,067	0,067	0,067	0,066	0,065	0,065	0,064	0,063	0,062	C
	3 этаж	0,057	0,057	0,056	0,056	0,056	0,055	0,054	0,054	0,053	0,052	E
Level 3		0,049	0,049	0,048	0,048	0,048	0,047	0,047	0,046	0,045	0,045	H
		0,043	0,042	0,042	0,042	0,042	0,041	0,041	0,040	0,040	0,039	P
		0,038	0,038	0,037	0,037	0,037	0,037	0,036	0,036	0,035	0,034	Z

		284,9	287,9	290,2	291,7	292,3	292,2	291,2	289,5	286,9	283,6	A
		284,6	287,7	289,9	291,4	292,0	291,9	290,9	289,1	286,6	283,2	B
		284,5	287,5	289,7	291,2	291,8	291,6	290,7	288,9	286,3	282,9	C
	4 этаж	284,4	287,4	289,6	291,0	291,6	291,5	290,5	288,7	286,1	282,7	E
Level 4		284,3	287,3	289,5	290,9	291,5	291,3	290,3	288,5	285,9	282,5	H
		284,3	287,3	289,5	290,9	291,5	291,3	290,3	288,4	285,8	282,4	P
		284,4	287,4	289,5	290,9	291,5	291,3	290,2	288,4	285,8	282,3	Z

в

		287,4	290,5	292,8	294,2	294,9	294,7	293,8	292,0	289,5	286,1	A
		287,2	290,3	292,5	294,0	294,6	294,4	293,5	291,7	289,1	285,8	B
		287,0	290,1	292,3	293,7	294,4	294,2	293,2	291,4	288,8	285,5	C
	3 этаж	286,9	289,9	292,2	293,6	294,2	294,0	293,0	291,2	288,6	285,2	E
Level 3		286,9	289,9	292,1	293,5	294,1	293,9	292,9	291,1	288,5	285,0	H
		286,9	289,9	292,1	293,5	294,0	293,8	292,8	291,0	288,4	284,9	P
		286,9	289,9	292,1	293,5	294,1	293,8	292,8	291,0	288,3	284,9	Z

FIG. 8 – Change of neutron fluence, 10^{19}cm^{-2}, E>0,5 MEv, (a), radiation energy release, W/m, (b), irradiation temperature,^{O}C, (c) in the specimen cross-section centers.

CONCLUSION

General regulations of the Nuclear Energy Development Strategy (up to 2010) provide the safety assurance and profitability for reactors of the first and second generation as well as their service life prolongation on the basis of the real life determination (first of all the vessel and unchangeable elements).

Precision of the vessel life of the VVER-1000 reactors under operation is possible due to precision of the time dependence of metal radiation embrittlement in the critical crack tip, account of radiation embrittlement rate change through the reactor vessel as well as validation of the prediction conservatism degree as the reliability of experimental results increases. Precision of the change regularity of material properties at the increased and nominal flux density of reactor emissions, generalization of prediction experience on VVER-1000 vessel life and comparison of the predictions with changes of metal properties under operation conditions will assure more representative determination of the vessels' residual life and validation of their service life prolongation.

Results of radiation embrittlement of base metal and weld metal through the whole wall thickness of theVVER-1000 vessel at the adequate distribution of the irradiation conditions at the increased and nominal flux density of reactor emissions can be considered as the basis for re-estimation of the VVER-1000 vessel life taking into account the previous and newly obtained results.

The facility can be used for irradiation of the VVER witness-specimens as well as for specimen irradiation during validation of new projects, new level of requirements to irradiation conditions and new redaction of the Standards. Examination results will increase the reliability of conclusions on life and service life prolongation of reactors of the first and second generation. Also they will contribute to implementation of the Nuclear Energy Development Strategy.

REFERENCES

[1] Tsikanov V.A., Vaneev Yu.E., Golovanov V.N., Grushanin A.A., Zaritsky S.M., raetsky V.M., Seriodkin S.V. KORPUS Irradiation Facility for Simulation of Operation Conditions of the VVER Reactor Vessel Materials. The 3d Inter-.Branch Conference on Reactor Material Science. Dimitrovgrad, 1992.
[2] Golovanov V.N., Markina N.V., Raetsky V.M., Seriodkin S.V. and Tsykanov V.A. Irradiation of Vessel Materials Using KORPUS Facility of RBT-6 Reactor. The 18th Symposium on Effects of Radiation on Materials. Hyannis, Massachusetts, USA, 1996.
[3] Standards for Strength Calculation of Equipment and Pipelines of Nuclea Power Facilities. (PIAE G-7-002-86). GOSATOMNADZOR USSR, - Moscow, Energoatomizdat, 1989, p. 525. (Standards and norms of nuclear engineering).
[4] Brodkin E.B., Egorov A.L., Golovanov V.N., Markina N.V., Raetsky V.M., Sulaberidze V.Sh., Tsikanov V.A., Zaritsky S.M.. The Irradiation Facility KORPUS for Irradiation of The Reactor Structure Materials. Reactor Dosimetry, ASTM STP 1228, American Society for Testing and Materials, Philadelphia, 1994.
[5] Raetsky V.M., Golovanov V.N., Gremyachkin V.A., at al. Irradiation Conditions for VVER vessel Irradiation in the KORPUS Facility // Proceedings of the Fifth Inter-Branch Conference on Reactor Material Science. Dimitrovgrad, SSC RF RIAR Volume.2.Part.1.P.69-88.

[6] Tsikanov V.A., Raetsky V.M., Golovanov V.N., Koslov D.V., Lichadeev V.V., Prohorov V.I., Krasnoselov V.A., Pechiorin A.M., Pimenov V.V., Sulaberidze V.Sh., Tihonchev M.Yu., Shamardin V.K., Shimancky G.A., Shushakova T.V., Yuskaev R.A., Yashina N.V., Kushnir Yu.A., Kashkirov A.A., Chernobrovkin Yu.V. Simulation of Radiation Embrittlement of the VVER-1000 Vessel Materials at the KORPUS Facility of the RBT-6 Reactor. // Paper on International Scientific Conference «Research Reactors and High Technologies" June, 25-29, 2001, Dimitrovgrad.

[7] Vishnevsky I.N., Grinik E.U., Chirko L.I., Droganov O.V., Revka V.N. Foct G., Trolla K. Radiation Embrittlement of Vessel Steels form the Ukraine NPPs. Proseedings on XIV International Conference on Radiation Phenomena Physics and Reactor material Science. June 12-17, 2002, Alushta, Crimea. RC XFTI, Xarkov, 2000, p. 94-95.

TECHNICAL CO-OPERATION WITH CENTRAL AND EASTERN EUROPEAN COUNTRIES WITH SPECIAL FOCUS ON ENGINEERING ASPECTS OF LIFETIME OPTIMIZATION

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Abstract

A regional Technical Co-operation project for Central and Eastern European countries was established and funded by the IAEA to improve long term integrity of primary circuit components as a fundamental technical prerequisite to ensure plant operation until or beyond the design lifetime. In a series of workshops and technical meetings, issues relevant to primary circuit component structural integrity with special regard to operational life optimisation and meet the region's needs, have been being addressed. Among the key issues are: flaw detection and sizing methods, requirements and their application in lifetime evaluation; improvement of in-service inspection effectiveness; component condition monitoring; evaluation of material degradation; application of leak-before-break concept; integrity assessment technologies. The project is implemented in co-operation with the European Commission's integrity related PHARE and TACIS projects.

1. INTRODUCTION

Long-term structural integrity of reactor pressurized components is essential for safe and reliable operation of NPPs. The assurance and assessment of structural integrity needs to take into account many factors, such as component design, quality assurance during manufacturing and commissioning, plant operation and maintenance strategy, loading condition data, material properties and their degradation, and the location and size of existing flaws. Structural integrity assessment is also a tool for evaluating the 'remaining' life of the main components with regard to safety margins, and is therefore essential for creating the technical basis for service life optimization of any nuclear unit.

The reactors designed in the Former Soviet Union (WWERs and all RBMKs), which form the majority of NPPs in operation in Central and Eastern Europe, were not designed to facilitate in-service inspection (ISI), and the reactor pressure vessel (RPV) surveillance programmes can be improved for most WWERs as well. Some of the countries operating WWERs are now establishing/strengthening their infrastructures for the assessment of structural integrity of primary circuit components. Moreover, extension of the operational life of the WWER NPPs became part of the generation strategy in most of these countries.

The IAEA has accumulated wide experience and knowledge in the field of structural integrity in general, and of primary circuit components integrity assessment of WWER NPPs in particular. Previous and ongoing regional Technical Co-operation (TC) projects have created the necessary environment for technology transfer and exchange of information on ISI related issues, specific problems of neutron exposure and monitoring related to RPV structural material embrittlement as well as managerial and economic aspects of life cycle management.

2. MAIN FEATURES OF THE TC PROJECT ON IMPROVEMENT OF PRIMARY CIRCUIT COMPONENT INTEGRITY

According to its mandate, the IAEA is authorised to make provisions for materials, services, equipment and facilities to meet the needs of research on, and development and practical application of, atomic energy for peaceful purposes, including the production of electric power, with due consideration for the needs of developing areas of the world. The assistance is provided as a priority to the developing Member States of the IAEA through the TC programme. Projects, which are implemented on the basis of requests from Member States, have to meet strict criteria in their formulation and implementation to be accepted by the IAEA. IAEA staff will assist to the extent desired by Member States at all stages of the project cycle. TC projects are financed entirely from the voluntary contributions of Member States of the IAEA to the TC Fund, but additionally donors may make extra contributions to projects where they may have a special interest.

The regional TC project entitled *'Improvement of primary circuit component integrity'* was established in 2000 as the logical continuation of a previous regional project dealing with application of advanced non-destructive examination (NDE) methods during ISI. The project is planned to run until 2004, with an amount of around US$ 700,000 foreseen as project core budget. The participating countries are Armenia, Bulgaria, Croatia, Czech Republic, Hungary, Lithuania, Romania, Russian Federation, Slovak Republic, Slovenia and Ukraine. The Government of Croatia has offered the country's capability (equipment of its facilities and experience on WWER-440/1000 ISI activities) to support the project, and the Government of Czech Republic has provided some US$ 40,000 extrabudgetary contribution. The major end-users of the project are representatives of utilities/NPPs and regulatory authorities responsible for component integrity assessment, safety evaluation, ageing and life management, ISI and surveillance, but representatives of technical support institutes have also benefited.

The overall objective of the project is to improve long-term structural integrity of primary circuit components at NPPs in operation in the region as a fundamental technical prerequisite to ensure plant operation until and beyond the design lifetime. It is expected that increased understanding and use of state-of-the-art methods and tools will contribute to reduction of the risk of loss of structural integrity of primary circuit components. The better informed decision making on primary circuit component integrity will lead to improvements in the safe and reliable operation of NPPs in the region, and improved technical information for decisions about optimization of NPP performance and service life will be available.

In a series of workshops and technical meetings issues relevant to primary circuit component structural integrity with special regard to operational life optimisation have been being addressed. The key issues are: flaw detection and sizing methods, requirements and their application in lifetime evaluation; improvement of ISI effectiveness; component condition monitoring; evaluation of material degradation; application of leak-before-break (LBB) concept; integrity assessment technologies. The workshops/meetings create a forum for technology transfer and exchange of information and experience. The works are co-ordinated with the European Commission's (EC's) integrity related assistance projects (PHARE and TACIS), and some of the activities are jointly organized with the WANO Moscow Centre.

3. INTERIM RESULTS OF THE PROJECT

Since we are approximately in the middle of the project cycle, some representative examples have been selected to demonstrate the rationale and the main outcomes of the major topical workshops/meetings.

3.1. Qualification of ISI systems

In 1998, the IAEA published the document entitled 'Methodology for Qualification of In-Service Inspection Systems for WWER Nuclear Power Plants' as a suitable framework for developing credible evidence that ISI systems such as equipment, procedure and personnel are capable to reliably detect and size flaws and so can meet the ISI objective [1]. This framework includes Technical Justification (TJ) and practical trials where required to develop sufficient confidence in the capability of ISI systems to achieve safety goals.

Based on the recommendations of the Member States in the region, a Pilot Study to assist Member States with practical advice on how to apply the Agency's qualification guidelines to a real NPP component was organised. The target application in the Pilot Study is TJ for ultrasonic testing (UT) of a shell weld of a WWER-1000 RPV (unit 5 of Kozloduy NPP, Bulgaria). The motivation for this work lies in the lack of systematic approach to demonstrating ISI capabilities in WWER NPPs. The Pilot Study is intended to contribute to general application to other components and other plants as well. It is important to state the purpose of the Pilot Study is to evaluate the methodology contained in the IAEA guidelines with *simulation* of the qualification process and not to endorse or qualify any specific organization or ISI programme.

The working team of the Pilot Study is composed from expert of those institutes which play role in the qualification process. These are the Bulgarian Nuclear Safety Authority, the Kozloduy NPP, the main designer of the nuclear steam supply system of WWERs (OKB Gidropress, Russian Federation), the ISI company (INETEC - Institute for Nuclear Technology, Croatia), and EPRI, USA. EPRI having large international experience on inspection qualification as well as being Qualification Administrator in the USA, has facilitated the work and played the role of the Qualification Body until the Bulgarian Independent Qualification Centre, as a result of the Pilot Study, was not established. To ensure consistency with both a bilateral assistance project of the UK to Bulgaria, and the European qualification methodology, representatives of the UK and EC's ENIQ (European Network for Inspection Qualification) were also integrated into the working team.

Simulation of the qualification process has been achieved through real technical contributions of the aforementioned organizations. The products of the Pilot Study include a technical specification, an NDE procedure, and a TJ, all of which were specifically developed by the appropriate members of a Pilot Study team [2]. It has to be mentioned that, as one of the most important input to the technical specification, the main designer provided a plant-specific calculation for the target depth of sub-surface flaws. The fundamental conclusion of the TJ is that based on presented evidences of the inspection system performance as well as taking into account considerations of TJ and ISI requirements, the inspection system is capable of reliably detecting and sizing flaws connected to clad/base metal interface equal or larger than the critical size requested. The system capabilities for ligament detection and sizing, however, need to be proven. A high level of confidence is expected by carrying out practical trials.

The results of the Pilot Study were discussed and disseminated in several workshops. The main conclusions are that, as a consequence of the Pilot Study, understanding on qualification of ISI systems is very much improved, and it contributed that countries started to establish systematically their qualification infrastructure and to launch pilot national qualification programme. It also helped that all interested partners of the qualification process like regulatory authority, utility, inspection company and qualification body, where available, working together which is essential for success.

Benefits deriving from ISI qualification having strong connection to NPP life management have the following features:

1) Inspection qualification increases confidence in the ability of the inspection to meet inspection objectives. This can be decisive in the process of obtaining license for the period beyond the design lifetime.
2) Effectiveness of inspection is being increased where safety, reliability and economics are clearly beneficial. This will result in less repetition of inspections, and plant availability will be increased. It is important that repair actions if necessary can be made at an appropriate time which also contributes to better maintenance planning and potential cost reductions.
3) ISI activities are better harmonized which leads to better competition, i.e. control of ISI costs. It has been better understood both by the NPP operator, who has the ultimate responsibility for carrying out the ISI and its qualification, as well as inspection companies.
4) Inspections are conducted in a systematic and well prepared manner due to the preparation required by the qualification. This will lead to an improvement in efficiency.

Central issue of the discussions was the need for reducing qualification costs. On one hand, it has been clear for everybody that ISI qualification has an added value in terms of decreasing component failure risk, reducing inspection time, extending inspection interval, changing plugging criteria, etc. On the other hand, qualification is costly exercise and its costs can and should be reduced by sharing available and applicable resources (test pieces, qualification information). Also recognition by one country of qualification result available in another country may lead to cost effectiveness of qualification. In this context, a joint effort with ENIQ to establish qualification test piece database, which will be accessible not only for ENIQ members but for others in the region, was initiated.

3.2. ISI of WWER reactor pressure vessels

In the Western PWR type NPPs, the periodic ISI of the RPV is traditionally performed from the inside surface. The WWER-440 model 213 reactors as well as the WWER-1000 units are equipped with permanent installations (USK-213 and SK-187 respectively) in order to facilitate mechanized ISI of the RPV from outside. Besides, in case of the WWERs, after removal of the internals, there are excellent access conditions for complete internal inspection of the RPV. Both ways of inspection have their specific merits in terms of impact on plant outage duration, flaw detection sensitivity, performance aspects and related radiation exposure, etc. This unique feature of WWER reactors has justified the need for thorough discussion on the topic.

In view of many of these aspects, ISI from inside and from outside can complement each other, so that they can be combined into a safety-conscious and cost-effective ISI strategy. Since the technical capability and reliability of the originally installed outside inspection equipment in the Russian designed plants have proven inadequate, upgrading of these systems have been systematically carried out in Bulgaria, Czech Republic, Hungary, Russian Federation and Ukraine. The upgrading usually meant replacement of manipulator controller (sometimes also the manipulator itself) and the UT module. The latter one usually resulted the change of originally used UT technology (tandem technique on the basis of mode conversion) to different one (share wave pulse-echo technique or phased array technique).

It was concluded that, by establishing complementary ISI systems for inside and outside inspections of the RPVs in WWER plants with appropriate performance level and maximum coverage of areas to be inspected, the ISI strategy could be optimized. Proof of the required performance level of the ISI system should be established on the basis of qualification of the ISI system. Also, extension of the inspection interval for the inspection from the inside e.g. from 4 to 8 years can be considered. This inspection is directly affecting the duration of the plant outage. Execution of the inspection from the outside should be kept every 4 years because this has practically no effect on the outage duration. Later a reduction of the scope or an extension of the intervals can also be considered, if reliable ISI system has not revealed any additional flaws or any growth of existing flaws.

This approach in turn allows to follow an ISI strategy for WWER RPVs which will be of equal benefit for the regulatory authorities by providing unambiguous results, and the NPP operators as it may help to reduce the critical path of outages through the impact of potentially longer inspection intervals and shorter inspections with increased efficiency. This may lead, in an economic analyzis concerning life extension, to practical benefits.

3.3. Steam generator performance and inspection

The major problematic areas of the WWER steam generators are:

1) Primary collector integrity (ligament cracking) of WWER-1000 steam generators;
2) Heat exchanger tube integrity (intergranular/transgranular stress corrosion cracking) of both WWER-440 and WWER-1000 steam generators;
3) Primary collector flange area (thread holes cracking) in case of WWER-440 steam generators;
4) Feed water distribution header (erosion damage at T-joint and at distribution nozzles) of WWER-440 steam generators;
5) Primary to secondary collector weld integrity (Weld No. 111) in case of WWER-1000 steam generators.

WWER steam generator tube degradation varies from NPP to NPP and from country to country. It can also be stated that more severe degradation is observed in WWER-1000 NPPs than in case of WWER-440s. The reasons for these differences could be explained by the following:

a. The inadequate structural material selection for components in the secondary circuit (feed water and main condensate systems);

b. The differences in the heat exchanger tube material properties (e.g. chemical composition, microstructure, residual stresses);
c. The differences in secondary circuit water chemistry regime;
d. The operational conditions/culture;
e. The modernisation/safety improvements measures completed.

Assessing operational conditions as fundamental influencing parameters of WWER steam generator performance and analyzing damage mechanisms occurring, it is obvious that control of secondary water chemistry in harmony with structural materials used in feed water and condensate systems, has primary influence on corrosion behaviors being responsible to deliver corrosive agents to the secondary side of steam generators. Replacing copper-based alloys as condenser tubes by stainless steel or titanium will help to eliminate copper transport in steam generators as well as may reduce oxygen intake in condensate system. Using adequate structural materials for, at least, elbows or T-connections suffering erosion/corrosion by high resistant material can help to reduce iron transport in steam generator. On the other hand, continuous working on enhancement of corrective treatment of feed water by chemical reagents is vital. Also, enhancement of water conditioning technology for improving makeup water quality, improvements on equipment conservation as well as installation of modern automatic systems for monitoring water chemistry are among the emerging issues.

Although the ISI of heat exchanger tubes became a daily routine work in almost every NPP at the region, there are differences in inspection strategy (scope), plugging criteria and level of inspection qualification if the latter is considered at all. Need for guidance on WWER steam generator heat exchanger tube inspection regarding inspection strategy, requirements for eddy current testing equipment, technology, personnel training, qualification and determination of alternate plugging criteria has very strongly been justified.

Assessment of structural integrity of steam generator heat exchanger tubes is a multidisciplinary issue. It needs expertise on reactor technology (system engineering), water chemistry, material science and technology (corrosion mechanisms), strengths and fracture mechanics analysis, NDE, repair technologies, etc. A well co-ordinated and harmonised activity of representatives of these various technical and scientific areas is essential for proper handling of the problems.

The lack of methodology of reasonable plugging criteria still seems to be a major problem. The new CRP on Verification of WWER steam generator tube integrity [3] has had a good acceptance from the countries in the region in the hope that it will certainly improve this situation. Results of non-destructive and destructive tests on tubes pulled out from steam generators, as part of the CRP, will make remarkable contribution to establishment of methodology for plugging criteria.

3.4. Structural integrity assessment

Structural integrity assessment of primary circuit components is a permanent task of operating organisations, regulatory authorities as well as design and technical support institutes. It has clear safety significance and very severe plant availability aspects. Nowadays, the emphasis of structural integrity assessment lays on life management (lifetime assurance, operational life extension). In some of the countries of the region new methodologies have been approved and introduced to facilitate lifetime evaluation with special focus on life extension. A dialogue between various players of this important task, and a dialogue between representatives of the different technical areas of structural integrity assessment (loading calculations – fracture mechanics, ageing - material degradation characterisation, and detection and sizing flaws by means of NDE) is vital.

The common approach of structural integrity assessment is to follow the relevant codes and use code criteria. In case of non-compliance with the acceptance criteria, the codes usually offer an individual evaluation method using actual material data, flaw size and loading conditions. It was recognized that differences on boundary conditions of the codes applied could lead to different evaluation results (e.g. different transient selected, different postulated defect size, different safety margins). Necessity of a benchmarking on structural integrity assessment procedures is desirable.

As one of the challenging integrity evaluation concept, a special workshop was dedicated to application of the LBB concept, which allows eliminating from the design basis the double-ended guillotine break of the main circulation line (an event with very low probability) as well as its dynamic effects. A fracture mechanics analysis should, however, demonstrate that a through-wall flaw, of a size giving rise to a leakage still will be detectable by the plant leak detection systems, remains stable even under accident conditions including the safe shutdown earthquake (SSE). The LBB concept is now widely used in the nuclear industry. It received a systematic attention during the IAEA's Extrabudgetary Programme on the Safety of WWER and RBMK Nuclear Power Plants as well [4].

There is a common understanding that the application of the LBB concept and the arrangements of pipe whip restraints are comparable measures with respect to general plant safety. Analysing the LBB applications it was concluded that LBB concept is applicable for the main circulation lines of WWER NPPs if the load case SSE will be upgraded by the installation of snubbers. In case of LBB application qualified leak detection systems should be installed. Also qualified ISI should be applied in the case of extensive LBB application. Up to now the LBB concept has been successfully applied for the following NPPs: Kozloduy NPP units 1-4, Temelin NPP unit 1, Bohunice NPP units 1-4, Mochovce NPP units 1-2. LBB concept is under implementation at the following plants: Ignalina NPP units 1-2, Temelin NPP unit 2, Novovoronezh NPP units 3-4, Kola NPP units 1-2, Balakovo NPP units 1-2.

3.5. Condition monitoring methods and techniques for assessing lifetime

Condition monitoring is an element of high importance in plant life management. Its integration into the entire life management concept, however, needs more comprehensive approach. The following prerequisites have to be considered: the monitored parameter has to be an appropriate indicator for the condition of the component and its acceptance criteria and limits should be available; all potential modes of degradation are addressed; traceability and predictability of the potential failure behavior should be available, and proper monitoring technique has to be installed.

4. ACHIEVEMENTS AND FUTURE WORKS

The regional project has contributed to better understanding of structural integrity related issues and their role in NPP lifetime optimisation. In particular, awareness of merits and limitations of state-of-art NDE and monitoring techniques, assessment methods, and correlation between operational regime and component degradation process have been focussed on. Proven international practices used in the Member States and summarized in IAEA technical documents have been thoroughly discussed and transferred.

The project has also contributed to strengthening national infrastructure in various areas of component integrity assessment, e.g. in the field of ISI qualification. Result of the Pilot Study on how to apply IAEA guidelines helped Member States to launch their national programme. The very practical outcome of the Pilot Study is the creation of a network of specialists working on the same topic in the region's countries. Development of the qualification test piece database is an example of sharing resources to increase programme effectiveness.

The project has contributed to initiation of new activities in the region. National TC project requests have been submitted to IAEA by some of the region's countries. Also the need for co-ordinated research on steam generator heat exchanger tube integrity criteria was unanimously expressed, and IAEA established a Co-ordinated Research Project on *'Verification of WWER steam generator tube integrity'* in 2001. Representatives of WWER operating countries as well as industrialized countries being involved in ISI of steam generators in WWER operating countries and in development of inspection capability for WWER steam generators came together and formed the necessary research network to collaborate.

Recognizing the need for co-ordination between EC and IAEA, and exchange of information about activities under PHARE and TACIS projects and the IAEA regional TC project, it was recommended to organize a joint EC/IAEA technical meeting every year. The main objectives of these meetings, through exchanging programme information and experience concerning ongoing and planned, national and international activities, to make sure that duplications are avoided, assistance programmes are harmonized, and critical research and development issues to support programmes are identified.

The forthcoming part of the project will basically continue the technical co-operation and technology transfer in the field of life cycle optimisation. High priority areas will be: ageing mechanisms and mitigation methods (RPV embrittlement, steam generator tube corrosion); on-line and periodic monitoring of component condition; maintenance and ISI programme optimisation to support lifetime optimisation; and lifetime evaluation methods.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Methodology for Qualification of In-Service Inspection Systems for WWER Nuclear Power Plants, IAEA-EBP-WWER-11, Vienna (1998).

[2] TRAMPUS, P., REED, J.M., Technical Cooperation with WWER operating countries to establish ISI qualification infrastructure, Proc. of 2nd International Conference on NDE in Relation to Structural Integrity for Nuclear and Pressurized Components, May 2000, New Orleans, USA, Volume 1, A-17 to A-25.

[3] GUEORGUIEV, B. et al., IAEA activities on steam generator life management, 4th International Steam Generator Conference, May 2002, Toronto, Ontario, Canada.

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Final Report of the Programme on the Safety of WWER and RBMK Nuclear Power Plants, WWER-SC-207, IAEA, Vienna (1998).

RM-ODP TO EXPRESS NUCLEAR LICENSING

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Abstract

Although some specific requirements in any licensing procedures of nuclear installations aren't discussed by the Brazilian Government, it is clear the basic requirements and recommendations that must be met to ensure design and acquisition control activities, to provide and to improve a good understanding of licensing procedures of those installations. Mean Qualified Suppliers, System Responsible, Licensing Requirers and others Member States and Suppliers may agree to establish, following the RM-ODP standards – enterprises modelling concepts, one federation that should support the licensing procedures, supplying technical and regulatory information's directly in a consistent approach of RM-ODP standards. This work outlines RM-ODP concepts that are applied for modelling activities in progress at different organizations of the nature of interactions between organizations and of the consistent approach of ODP. In conclusion, it will be showed that the level of flexibility and the confidence of any licensing procedures will improved for the population, in general, with the application of the mentioned standards in those regulatory procedures.

1. INTRODUCTION

The Brazilian Government allow the start up of Angra 2 nuclear power station and the construction of ten new thermo-electric power stations, and provides the start up of the major part of forty nine power station which are included into the Brazilian Thermoelectricity Program, up to year 2002 and 2003. From the point of view for building schedule, the projects are began and several are being into configuration revision phases and evaluation of their technologies into the specific projects phase, such as: project, building, operation, generation and the distribution, considering the recommendations from Regulatory Programs.

The CNEN programs and others from the institution which are responsible for environment licensing establish requirements to guarantee and keep the society safe from hazard operational aspects and to minimize the environment impact due to the implantation and operation of those installations include nuclear power stations and nuclear facilities.

The program introduced from ANEEL (Agencia Nacional de Energia Elétrica – Energy Agency) is named of ECS (Esquemas de Controle e Segurança – Safety Information) of the Brazilian Integrated Energy System. For being included into ECS Program, the Power Station needs to establish one information infra-structure able to support action, operational and safety informations for ONS (Operador Nacional do Sistema Elétrico – Senior Operator of

National Electricity Distribution), CNOS (Centro Nacional de Operações do Sistema Elétrico – Operational Centre of Electricity Distribution), power station's owner and university's institutions. Such infra-structure needs one improvement in communication's systems and cooperations actions between different organizations involved in construction, operation of power generation stations, and needs improvements for fixing limits to the professional actuations from licensing sectors of regulatory organizations. Due to that, automation and information applications have to be capable to support, each organization involved, clear informations and functionalities for sending and receiving requisitions, as the same way, that implements additional informations and functionalities to offer and to use specifics services.

The reference model ODP (RM-ODP) provides the behaviour, context, methods for abstraction and composition, and share concepts to realized an architecture specification of the information technology, means to establish one system of distributed process applied to ECS as licensing procedures of nuclear installations. This distributed system can to contribute to demonstrate for de society the safe uses of those installations.

2. ODP REFERENCE MODEL

The nuclear behaviour is traditionally conservative and although has in the world several nuclear installations in operation, at first the Brazilians projects are even in implantation phases. To guarantee the conformity during implantation phases of those Brazilians nuclear systems, the computational systems have to work with quite a lot of informations related with all stages of nuclear licensing procedures and to attend all aspects of regulatory interest, validation and adequability of those nuclear projects.

2.1 The specification language of ODP Reference Model

The ODP reference model is a specification language [1]. O RM-ODP is a standard used to model objects from architecture of distributed process. The reference model applies, in distinct approach, the fundamentals of heterogeneous and open system with distributed process, with the objective to simplify the implementation of those fundamentals. The model allows that modelling of the system been constructed with uses of one language oriented to distributed process, in the way to avoid ambiguities of intension and in order to obey the final specification, resulting from architecture process, with semantics and syntaxes well structured.

The reference model established concepts of tasks organization of the distributed system development, further than to define the terminologies and uses' rules. The ODP model supplies conditions to obtain an complete interoperability between organizations and process, offer also conditions for work environment definitions, and shows the necessary fundamentals to map the functionalities and non-functionalities of the systems in development.
Those fundamentals and concepts are placed as rules, behaviour, and architectures and projects standards, which will be used for support the orientation and the definition of implementation mechanisms.

The ODP has a lot of concepts to established common terminologies and for use as a guide to specify a software architecture of the system, which is object oriented. The ODP basic concepts are:

(a) Open system of distributed process;
(b) Accuracy specification of operational environment;

(c) Composition, abstraction, improvement;
(d) Links between several abstraction's levels;
(e) Consistency between specification and implementation; and
(f) Tests conformity.

The object behaviour is defined by one group of actions and by restrictions for those actions. Safety actions are included in this group, which are enabling when any restriction is achieve. The RM-ODP models the behaviour where the objects and their actions operate. This fact is a very important point from modelling activities, due to the behaviours show additional restrictions for objects operations. In the case of nuclear licensing, there are restrictions from the nature of licensing process due to norm requirements. Therefore, there are additional restrictions belongs to design scope of each organization, individually, as well as, for realization of the scope of the federation, group of organizations or the communities. One example of those restrictions is the in company processing resources existing in each organization.

A contract is used in RM-ODP to characterize the interaction between objects. In the same way at business area, a contract is an agreement between objects and it governs what has or has not to do. A contract, in software context, defines what has to do by the software entities and not how things will be done. The "how" is determinated by the architecture specifications.

The policy concepts is also used in RM-ODP to characterize and to control the behaviour and for to define the purpose of some groups of objects. The policy is a group of sentences and requirements, which support permitions to execute an activity. Policies are established to all abstractions levels and one can complement another.

The distributed process system, which implements requirements of nuclear licensing, is complex and show several aspects heterogeneous in its architecture, projects, and in selected products which have to be grouped. Therefore, the successful of the implementation and operation of these systems is function of the successful of the management program of development activities and of exactly business process modelling. This management program is defined to capture all the possibles solutions, which cover the heterogeneous behaviours and the system distribution. The program has to permit information changing between uses placed geographically distributed and with distinct functionality needs.

The management of nuclear licensing is very complex, due to a lot of alternatives and requirements, which have to be followed. At the same manner, the management of nuclear installations often support financially means to minimize and solve these complexities. At specific situations, the manager, verifier and executers of the development have an approach very conservative improving complexity level of the licensing process, and as consequence, the complexities of the support systems.

The nuclear standards, which are established by CNEN and by others regulatory organizations, are open for public knowledge. These standards established conditions for to capture and included into the definition of architecture elements of the system the following aspects:

(a) Analyse of the information structure within the standards;
(b) The communication, between different scopes, of a problem to be solved;
(c) The objectives of the solution;

(d) The rules of the licensing's interaction activities, which should restrict the possible technical solutions;
(e) Obligations;
(f) Restrictions;
(g) Integrations and expected results.

All these ODP architecture elements motive and generate all necessities for system definition [2]. The system exists only to be a support for the business, which in your case is an obtainment of formal licensing act.

3. META MODEL

The CNEN scope, which is defined in normative fundamentals, establish the context where the activities for the nuclear licensing of an installations are realized by the professionals, machines, systems and other entities of real word. The CNEN objectives, which attend licensing aspects, were specified and define how the system has to be architect and build, and what important aspects were considered into their constitution. The behaviour, where the software system will operate, is likely defined through the characteristics of the business process, its means, for licensing scope.

The essentials objectives have the finality to capture, as well as be possible, the highest level of the business abstraction, which details are implemented up to successively refinement during the development phases. Therefore, the licensing specification, general vision of the project, has to be global even in their earlier stages, and these factors shouldn't generate problems for the realization of the architecture specification, in the consistent way and despite the fact that it is not possible to know all the aspects of the business (nuclear licensing) at the earliest stages of the development.

The figure 1 shows the top-down context of the domain specification process that results an architecture specification, where the common stages by the architect for each application are showed in grey ton [2]. The method of architecture showed in figure 1 put at the first place the business domain which will be modelling in terms of applicable terminologies. In the licensing context, these terminologies constitute one collection of terms of nuclear licensing, such as: Mean Contractors, Requirers, Agent of Reactor Coordination and others. These terminologies have to be well defined and be public knowledge, by all professional involved in nuclear licensing procedures.

The architecture specification defines and uses the knowledge architecture concepts, which are: abstraction and refinement level, point of view, validations and objectives. The specification also maps the terms specific of application domain used terminologies of the system of distributed process. Your specific domain is the nuclear licensing, which has the following: one agent licensing requirer which assumes one or more defined roles, one main contractor which is defined as a community, one PSAR (Preliminary Safe Analysis Report) which is defined as a software component, and others. As the terminologies, concepts and rules are defined and the mapping of these terms into the specification of distributed process is concluded, using the domain languages, the specification of the architecture turns conceptually well structured.

In auxiliary the works to specify an architecture, the UML (Unified Modelling Language) [3] was used to brings the visual representation of the architecture. Thus, the concepts and the specific domain of the distributed process are translated into those UML concepts. The UML can represent the agent of licensing requirer or the CNEN politics to nuclear licensing, which are defined as some objects classes or, yet, the following way: the require agent is defined as one actor into a process of the Uses-Cases (UML concepts) and politics as Classes.

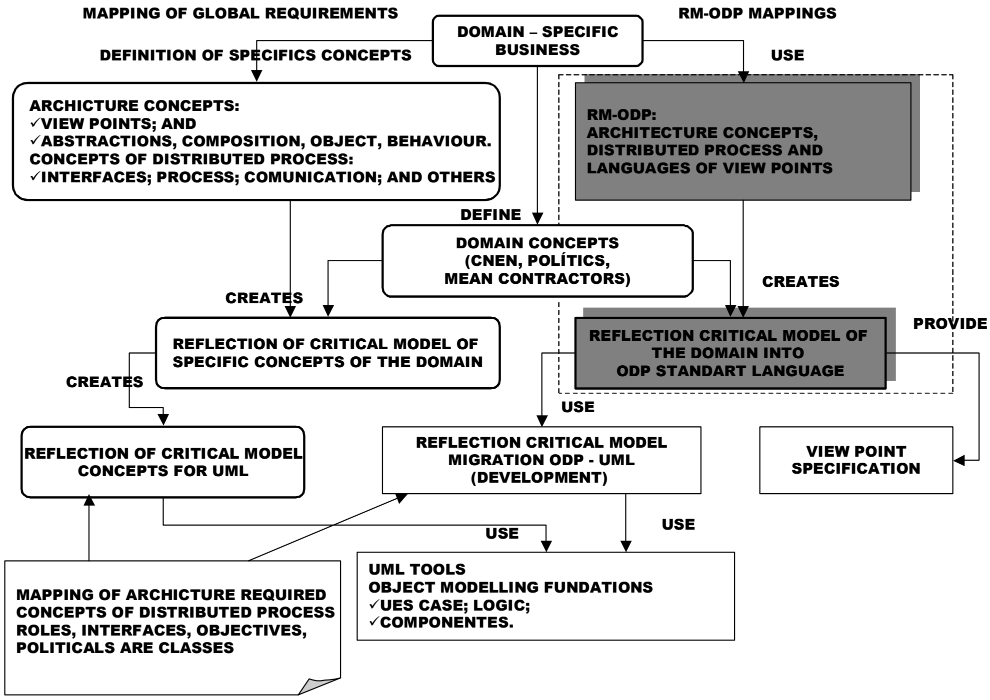

FIG. 1 - Methods of architecture definition

The specification process begins by the scope definition of the system through enterprise and information point of view. The CNEN scope is defined with the conclusion of the activities which are executed by persons, machines, entities of the real world and by the proposing system. The CNEN objectives are the licensing of nuclear installations and they will define what the system has to do and their importance. All factors are define, which related with the business environment where the software will operate. The are, in no doubt, one or several software systems belongs to CNEN with different kinds of the specifications considering the business view point which have to be grouped in only one system. This combined system has to be specified in terms of their roles, objectives and relationships (interaction and process) with external system or entities of the real world.

3.1 The Business Process

One of the criticals factors at the information specification of the licensing process is an integration of the application showed above, it means, the integration of all components into RM-ODP languages. The integration of those applications has to be conduced to support one information environment which is possible be shared with all components.

The figure 2 shows one general model to define the architecture, through a configuration of distincts applications. Therefore using the mentioned model and refining the critical reflection model, the IDEF0 modelling is inserted to map and integrate the actions outside of the CNEN communities.

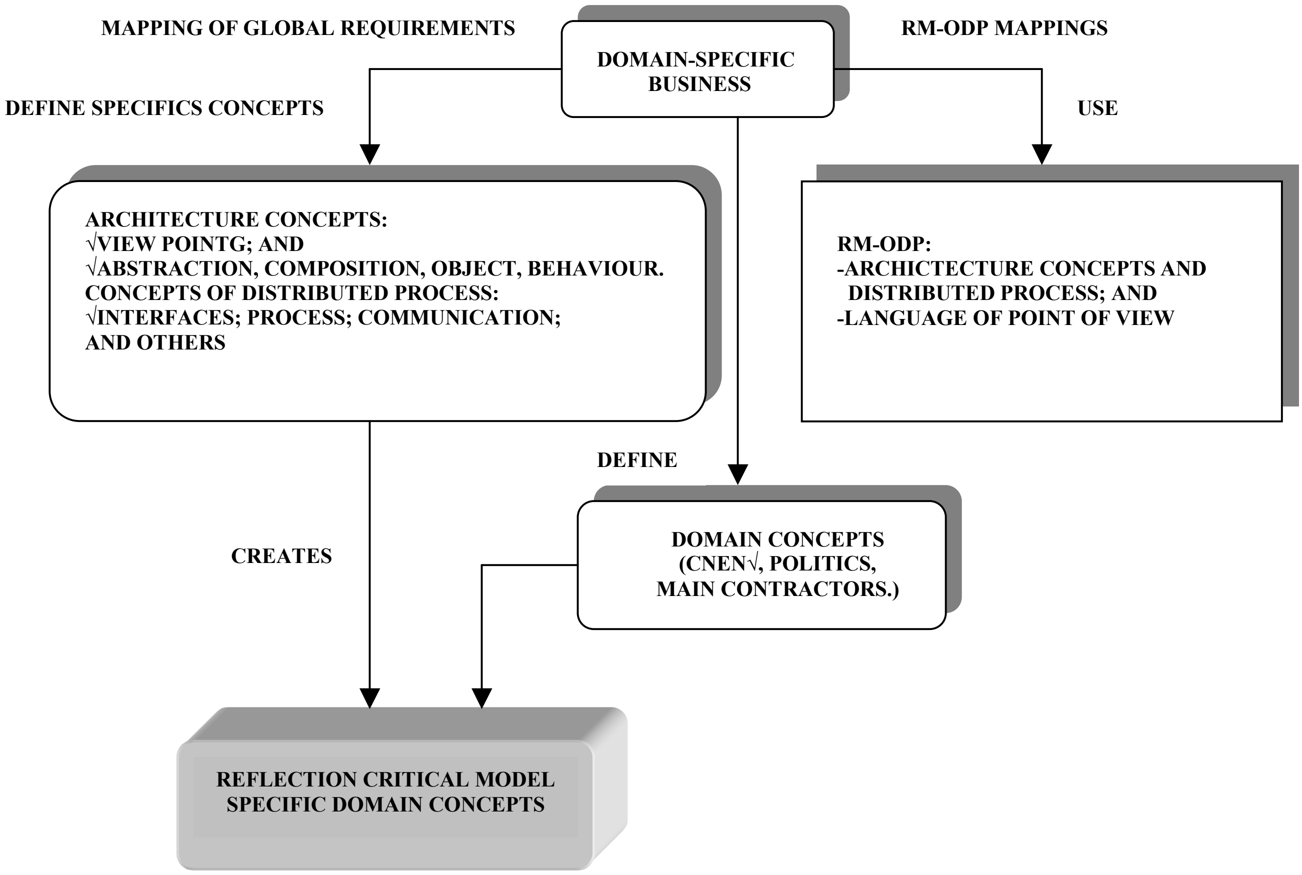

FIG. 2 – Critical Reflection Model

The critical reflection model for specific domain concepts has the finality of to integrate all applications defined into the maps of information specification. This integration supports the definition for sharing the information environment and creates additional data to define each applications and interactions.

The shared environment definition of the information and the actions for integrating the applications and their interactions are input objects of the IDEF0 modelling. The modelling control is established by the applications of the rules of the architecture specifications, such as: composition, abstraction, refinements and precision. The result of the modelling established concepts of the business domain, it means, concepts of nuclear licensing which are proposed in terms of RM-ODP concepts. In this way, the definition of all the information is established through the process mapping between the communities as result of the business process specification which was modelled by the IDEF0 diagrams.

The specification of business process defines the information and the information processing through the communities, independent of their interfaces, dada basis or any others implementations mechanisms.

The modelling of the process licensing is composed by the following means blocs [5]: site approval, construction licensing, and authorizations for utilization of nuclear materials and authorization for operation. Each of these blocs are controlled by following rules: global, site approval, construction licensing, nuclear material utilization and safe operation. All blocs, except the bloc of site approval, are objects of inspections and auditoria as is defined by licensing rules.

The process model input, yet in the early level of abstraction, are the requirements, modifications, tests, experiences and technical modifications. With the exception of the requirements which are the input at all defined blocs, the modifications, test, experiences, and technicals changing are input to the bloc of operation authorization. The infra-structure necessary for modelling the all system were not mapped completely at IDEF0 diagrams.

The output of model blocs are the following acts: site approval, construction licensing and the authorization for nuclear material utilization and for installation operation. Besides these acts, is specified the cancel acts for operation of the installation. The CNEN, attending the public interest, might revise the licensing gave and, also, retrieve the licensing due to risk of nuclear damage or due to law modifications, include the modification of CNEN specifications.

The site approval bloc showed at figure 3 is refined into the following blocs: Global Aspects, Populations Distribution, Physical aspects of the site, and Pre-operational environment monitoring. The table 1 shows the rules, input and output referring to this bloc.

Table 1 - Site Approval – Descriptions of the Input and Output Rules.

Global Rules
• CNEN standards statements; • Completely descriptions of the intendment modifications.
Site Approval Rules
• Standards statements of the CNEN to place the installations.
Requirements <site approval>
• Intendment uses; • Nominal capacity; • Nature and radioactive material inventory which have to be confined; • Probability and consequences of an accidental liberation; • Definition of Plant Reference; • Population distribution and traffic access; • Physical aspects of the plant site (seismology, meteorology, geology, and hidrology); • Potential influency at environment due to installation construction and operation phases; and • Pre-operational environment monitoring.
Output <site approval>
• Approval of Site Report.

The requirements (input) are located into specific unit and are organized by their informations. This selection was be made to emphasize where the information is relevant for each process phase.

The Licensing Construction Unit has as input the requirements, which context will be refined as preceding described.

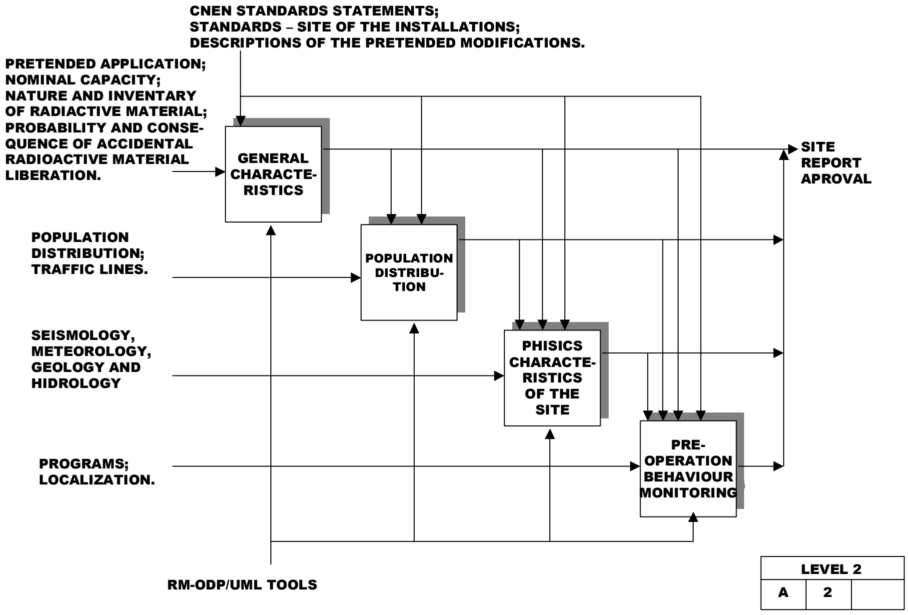

FIG. 3 – Modelling of Site Approval Report

The next, it will show, in the figure 4, the relevant topics of the input objects of licensing construction modelling as PSAR elements, for example:

(a) Technicals qualification of requirer;
(b) Description and safety analysis of the site defined to place the installation;
(c) Description and installation analysis;
(d) Preliminary project of the installation;
(e) Preliminary analysis and project validation, and performances of installations items;
(f) Description and justification of variable choice, conditions and others characteristics;
(g) Preliminary plan of personal training;
(h) Quality assurance program;
(i) Characterization of installations items;
(j) Identification of potential risks;
(k) Preliminary plan for proceedings of emergency situations;
(l) Description of the system for control of effluents and radioactive discard liberation; and
(m) Description of preliminary fire protection plan.

Besides the informations showed into table 1, there are others informations related from SIPRON (Nuclear Protection System) Program which are included into licensing construction modelling, as know:

(a) Organization, to face the emergency situations;
(b) Agreement and contracts;
(c) Measurements in case of accidents;
(d) Resources;
(e) Provisions for treatments;
(f) Access; and
(g) Training.

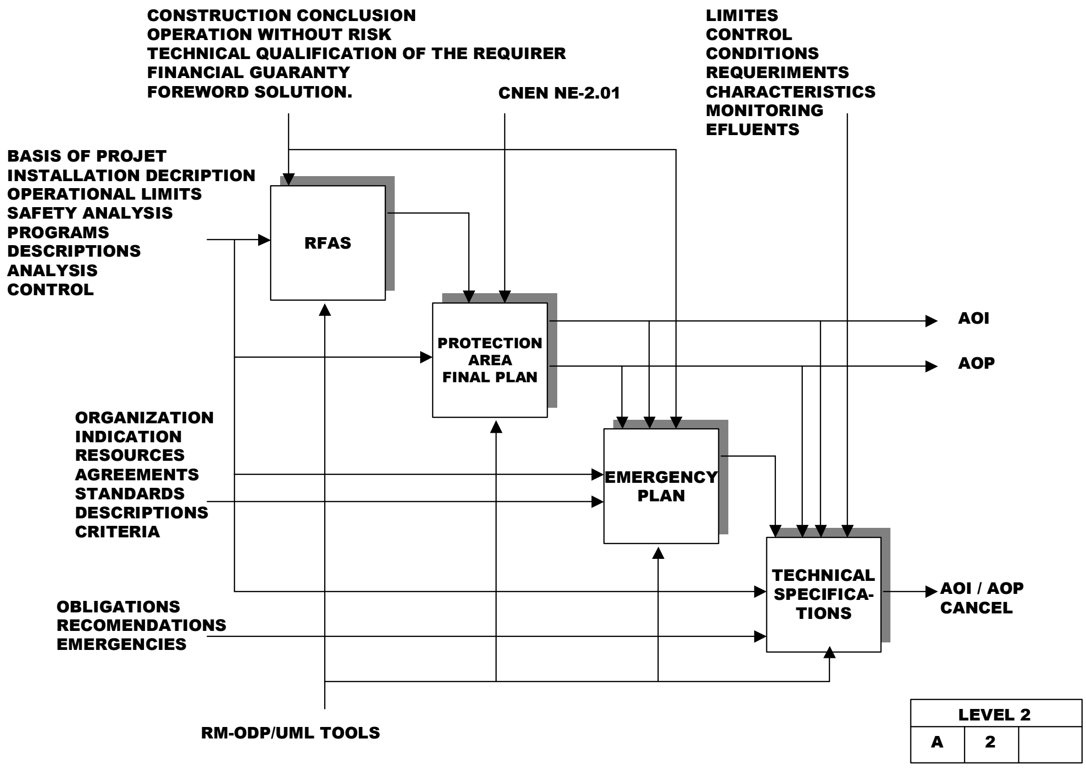

FIG. 4 – Modelling – Construction Licensing

All of these informations are directly related with renew, suspension, get back, modification or improvements of construction licensing gave.

The rules of construction licensing and global, as well as, the obligations referring to inspection and auditoria were grouped into rules, obligations, code and standards for Licensing Construction Unit and refers to subjects showed in the table 1, as know:

(a) Reports, proceeding and informations;
(b) Notifications to CNEN;
(c) Permitions to open access; and
(d) Implementations of functional interfaces.

The last modelling unit of nuclear licensing process is the unit of Operation Authorization. This unit is showed at figure 5.

The requirements for this unit cover programs, descriptions, analysis and controls which structure the informations need to make the FSAR evaluations. The evaluations for emergency plan need the informations of type: organization, indications of personal allocation, and conditions of verification of radioactive liberation materials, agreements firmed for public evacuations, standards, descriptions and criteria for new start-up of the installation. The evaluations of the technicals specifications consider the definition of recommendations, emergencies and obligations, which are integrated with the limits and security adjusts, operational limits conditions, requirements of inspection and periodic tests, features, control and effluents. These criterions are schematic showed at the figure 5 and their scope makes the correlation between the proposed specifications and important topics for the CNEN evaluations [6]. The rules and obligations refer the result of the construction, operation without risk, technical qualification of the requirer, financial guarantee and the foreword solution.

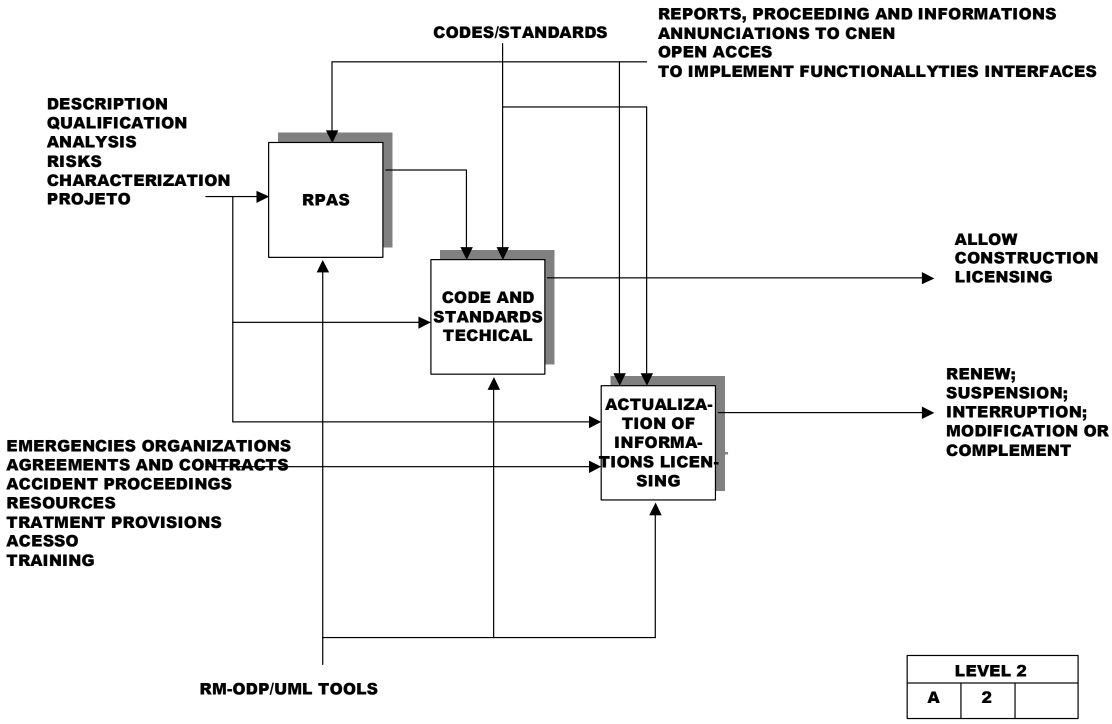

FIG. 5 – Modelling – Operational Authorization

4. FINAL CONSIDERATION

The concepts and definitions showed in this paper defined the specific business domain, which is used by context ODP. There were defined the process functionality requirements of nuclear licensing, the relationships scope, the interaction rules which contributed to specify the nuclear licensing. Thus, the domain definitions following the architecture concepts and allows to implement the critical reflection model, where, with the auxiliary of IDEF0 diagram, the interaction between externals domain were mapped.

The figures 6 and 7 showed two model examples of the specification for enterprise view point into ODP context. The model of figure 6 is one generic model for the associations and relationship of dependence for CNEN community, which were refined. Therefore, the requirer community, which requires the licensing of construction or operational authorization, has one interaction, type dependence, with the community of Diretoria de Radioproteção e Segurança (DRS) of CNEN, it means, the requirer community shows dependence relationships between the following entities: [agente_requerente_cnen] and [agente_diretor_cnen]. The community [cnen.drs] has dependencies with their communities defined as: [cnen.drs.cglc] and [cnen.drs.cglc.codre], which are respectively the communities of general coordinator of licensing and control, and the reactor coordinator.

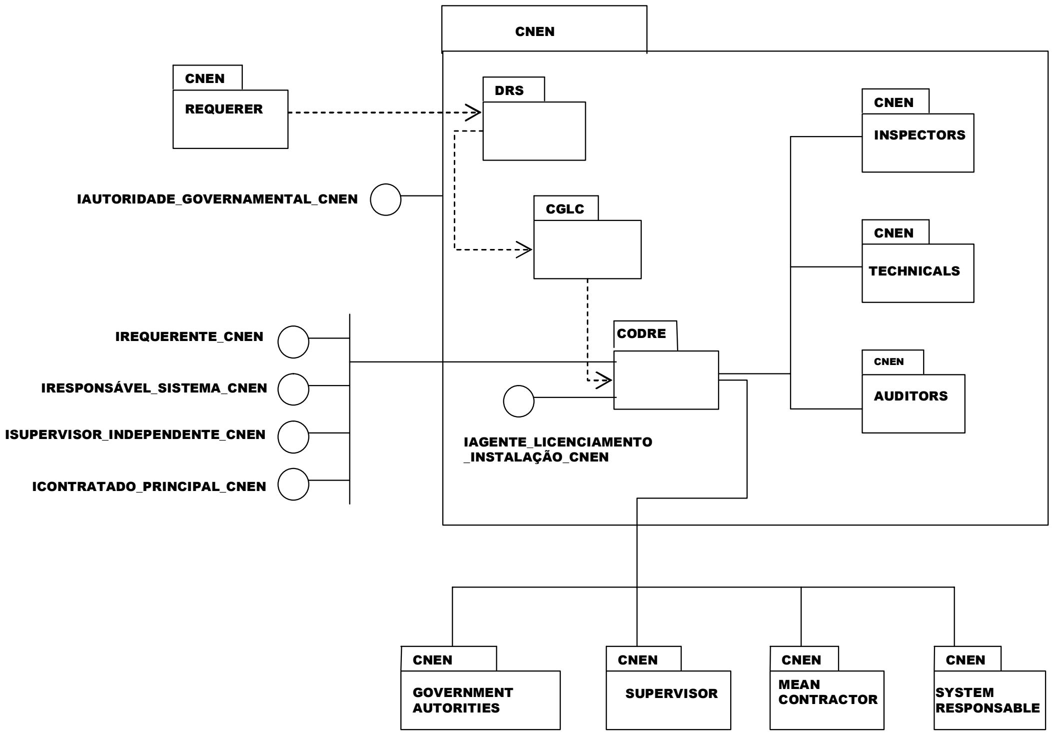

FIG. 6 – Associations of the CNEN community

The community [cglc.codre] is part of the community [cnen.drs] and established relationship of type association between the technicals community - [agente_tecnico_cnen], inspectors – [agente_inspetor_cnen], and auditors – [agente_auditor_cnen], which are constituted from the cnen community [cnen].

The community [cglc.codre] realizes one internal interface of the CNEN community identified as agent of installation licensing – [IAgente_licenciamento_instalação_cnen]. The external interfaces are established with the following agents [7]:

(a) Requirer of construction licensing or authorization for operation: interface [IRequerente_cnen];
(b) System responsible: interface [IResponsável_sistema_cnen];
(c) Mean contractor: interface [IContratado_principal_cnen]; and
(d) Supervisor: interface [ISupervisor_independente_cnen].

The figure 7 refines little more the defined interactions. O RM-ODP makes easy the communication with the final user and with of all involved into the system development. The necessities of one specific domain are reported to the ODP elements (domain defined as a community) into the enterprise view point. The architect uses this concept, community, and there relationships with others concepts for creating open specifications, which use languages well defined in terms of distributed process for communicating exactly what it intends.

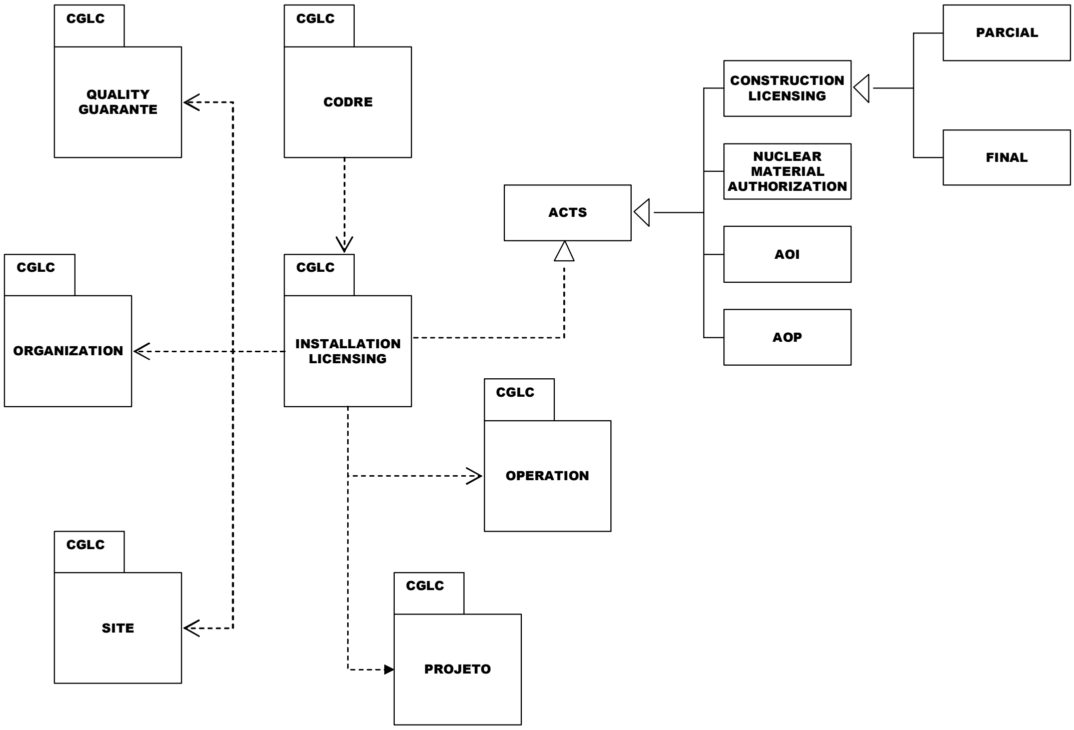

FIG. 7 Associations of Community [cnen.drs.cglc.licenciamento.instalação]

REFERENCES

[1]. AEGEDAL, J.O., "Modelling Virtual Enterprises and Character of their Interactions", IEEE (1999).
[2]. PUTMANN, J.R., "Architecting with RM-ODP", Prentice Hall, Inc. (2000).
[3]. BOOCH, G., "The Unified Modelling Language User Guide", Addison Wesley (1998).
[4]. IDEF0, "The Unified Modelling Language User Guide", Federal Information Processing Standard – FIPS (1993).
[5]. COMISSÃO NACIONAL DE ENERGIA NUCLEAR (CNEN), "CNEN-NE-1.04 Licenciamento de Instalações Nucleares", Brazil (1984).
[6]. COMISSÃO NACIONAL DE ENERGIA NUCLEAR (CNEN), "CNEN-NE-1.28 Qualificação e Atuação de Órgãos de Supervisão Técnica Independente em Usinas Nucleoelétricas e Outras Instalações", Brazil (1999).
[7]. COMISSÃO NACIONAL DE ENERGIA NUCLEAR (CNEN), "CNEN-NE-1.16 Garantia da Qualidade para a Segurança de Usinas Nucleoelétricas e outras instalações", Brazil (1999).

INFORMATION SUPPORT TO THE NPP OPERATING PERSONNEL WITH THE USE OF SPDS

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("Armatom" Institute)
ARMENIA

Abstract

This document contains a description and basic functions of the informational safety parameters display system (SPDS) at the ANPP Unit 2, including its structure, description of a program-technical complex, its special functions, including the subdivisions, as well as a software (SW) with its basic parts.

1. INTRODUCTION

SPDS of the ANPP Unit 2 is a real-time informational system designed by DS&S Company together with «Armatom» Science-Research Institute. Basically, SPDS is designed to provide the MCR (main control room) operating personnel with the information support during incidents and accidents. Moreover, SPDS monitors parameters characterizing critical safety functions (CSF) During all the operational modes.

The support to the nuclear power plant operating personnel is provided through:

- presentation, within the SPDS structure, of the information on CSF status with the purpose of detecting timely the symptoms of violation of the safety barrier integrity conditions, accepted according to the « safety-in-depth» concept, and determination of a safety-priority direction for the personnel actions;
- presentation to the NPP operating personnel and managing staff of the generalized information on parameters, which indicate the NPP safety;
- control over the correct performance of the emergency actions.

2. DESCRIPTION OF SPDS GENERAL FUNCTIONAL STRUCTURE

Establishment of SPDS aims at [1-3] increasing the power unit operational safety level and reducing the probability of emergency occurrences at the unit for the account of:

- early detection of deviations in the operation of a power unit;
- presentation to the operating personnel of the integrated information about the status of safety barriers, safety systems, major equipment and parameters of a power unit;
- reduction of the amount of human errors caused by a wrong evaluation of a situation and/or by a wrong decision-making.

SPDS has the following functions:

(a) data acquisition on process line parameters and status of valves and mechanisms of a power unit, and their transfer.

(b) data processing and managing:
- implementation of algorithms of input data processing;
- implementation of algorithms for calculation of estimation variables;
- maintenance of the SPDS variables database.

(c) data recording:
- analog and discrete signals, in on-line archives;
- analog and discrete signals, in long-term archives;
- events, in archives of alarms on deficiencies;
- discrete signals, in special archives;
- analog- and discrete signals, in archives of disturbances.

(d) data display:
- monitoring the CSF status;
- monitoring the CSF status trees;
- monitoring the key process line parameters in a normal- and emergency operation modes of a power unit;
- monitoring the safety systems status;
- monitoring the current- and archive values of process line parameters;
- data display upon the formats (system mimic displays, trends, status diagrams, bar graphs);
- output of the help information;
- documenting the information.

Using the SPDS, ANPP Unit 2 is monitoring the following critical safety functions:

- Subcriticality;
- Core cooling;
- Heat removal in the secondary circuit;
- Integrity of the primary circuit;
- Leak-tightness.

3. STRUCTURE OF THE SPDS HARDWARE

Figure 1 shows the block diagram of the SPDS hardware.

The hardware consists of the following components:

(a) Data acquisition system:
- subsystem of input-output;
- G2 data server (processor of input-output).

(b) Data transfer line:
- redundant optical network of data acquisition;
- redundant optical network of the top-level acquisition.

(c) Alpha-server:
- workstations for display of the information;
- uninterrupted power sources and printers.

4. STRUCTURE OF THE SPDS SOFTWARE

The SPDS software (SW) consists of the following parts:

- system integrated software including the operational systems and covers, standard service programs, standard programming systems and database management systems;
- special software including various software tools that resolve all complex of functional tasks at a power unit level. These software tools include: programs of processing and analyzing the information coming from the unit; databases; means of recording and documenting the information; programs of display and graphic presentation of the information to the NPP personnel.

On all the SPDS workstations, the operational system Windows NT4.0 has been installed.

For SPDS of the ANPP Unit 2, the special software SAIPMS and SDS/DV of DS&S Company is used.

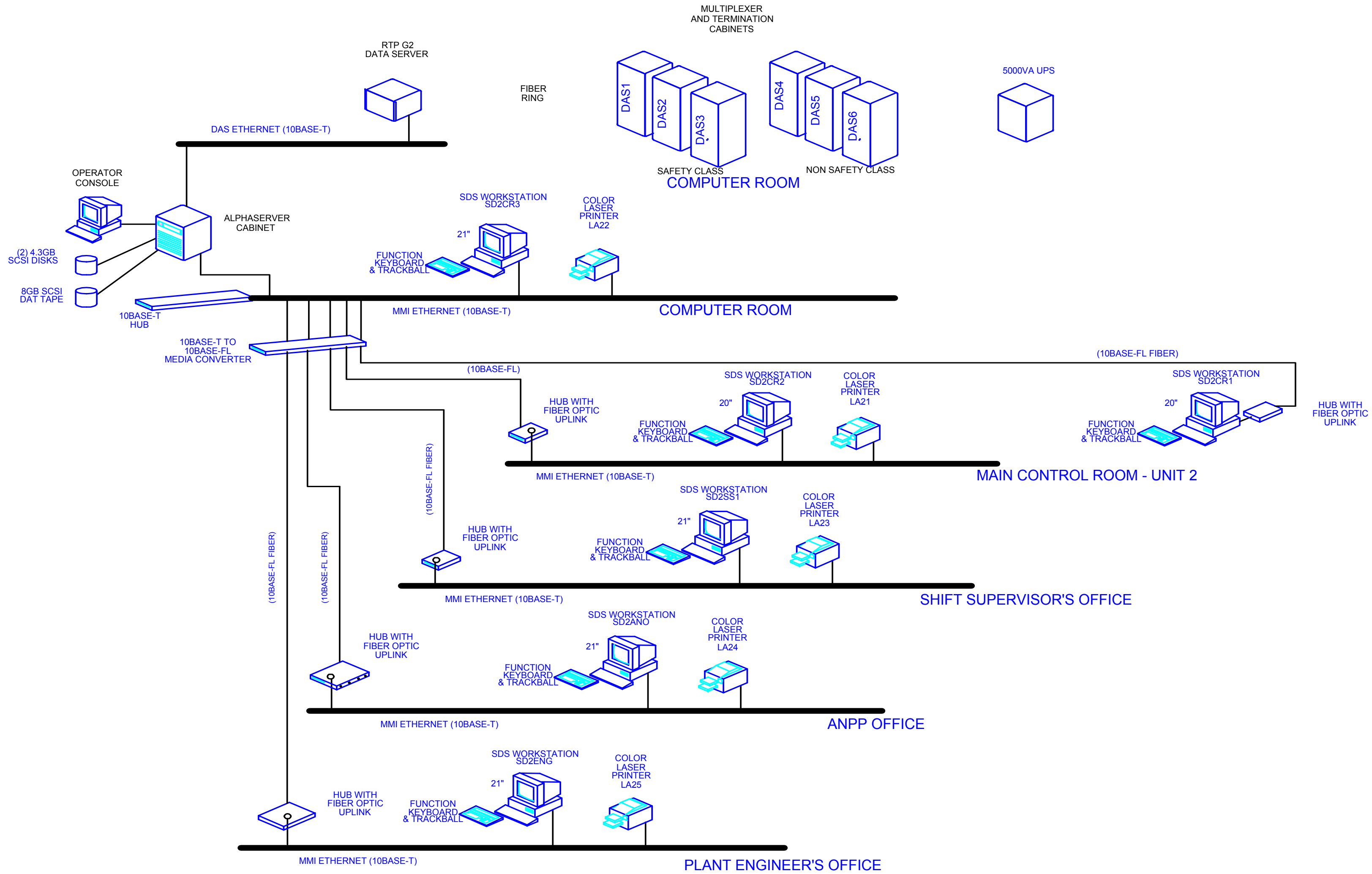

FIG. 1. The Block Diagram of the SPDS Hardware

SAIPMS is a flexible software for managing the real-time database connected closely with SDS/DV. It works under the control of operational system Open VMS and provides implementation of the following functions:

- management of interface with the data acquisition system;
- support and management of the real-time database;
- verification of input data validity;
- performance of various types of arithmetic, mathematical and logic operations;
- estimation of thermodynamic conditions, filing the database (in archives), etc.

SDS/DV is a software for organization of the man-machine interface in a real-time mode intended for creation of formats and tools of the user interface. Using the SDS/DV, the user can create various types of formats placing upon them the objects, establishing their dynamic attributes, adhering the dynamic objects to the sources of information.

It can be used, as such the objects, the following:

- diagrams;
- single- and multi-parameter diagrams;
- status trees;
- X-Y diagrams;
- simulating systems;
- tabular data output;
- horizontal and vertical variable bar graphs;
- animation symbols;
- dynamic textual messages.

SDS/DV has an access to, displays and manipulates with the data received practically from all the following SAIPMS components: database, on-line archives, long-term archives, subsystem of recording and analyzing the transients. Besides, SDS/DV provides the interface to an alarm subsystem and all server appendices. Figure 2 shows the overview of SAIPMS and SDS/DV, appendices and supporting functions, which SAIPMS consists of.

5. DESIGNATION AND DESCRIPTION OF FORMATS

The SPDS basic formats are designed for monitoring the power unit parameters. The SPDS formats are the basic means of presentation to the operator of the operational information about the unit parameters. They are organized in such a manner that the information, concerning various systems, is grouped upon the different formats according to the functional attributes.

For the presentation of the information, the following types of formats are used in SPDS:

- system mimic displays;
- status diagrams;
- tables;
- time dependencies of the variables;
- historgams;
- alarms.
-

On each of parameters, there can be obtained the additional help information including:

- information on a variable;
- list of inputs;
- trends on the selected parameters;
- general information on the database.

In the SPDS, the special place is occupied by the top-level formats designed to display the CSF. These formats are designed for conducting the symptom-oriented diagnostics, which role is to:

- calculate and display on the screen the CSF status;
- specify for the operator the necessary symptom-oriented instruction on recovery of the CSF.

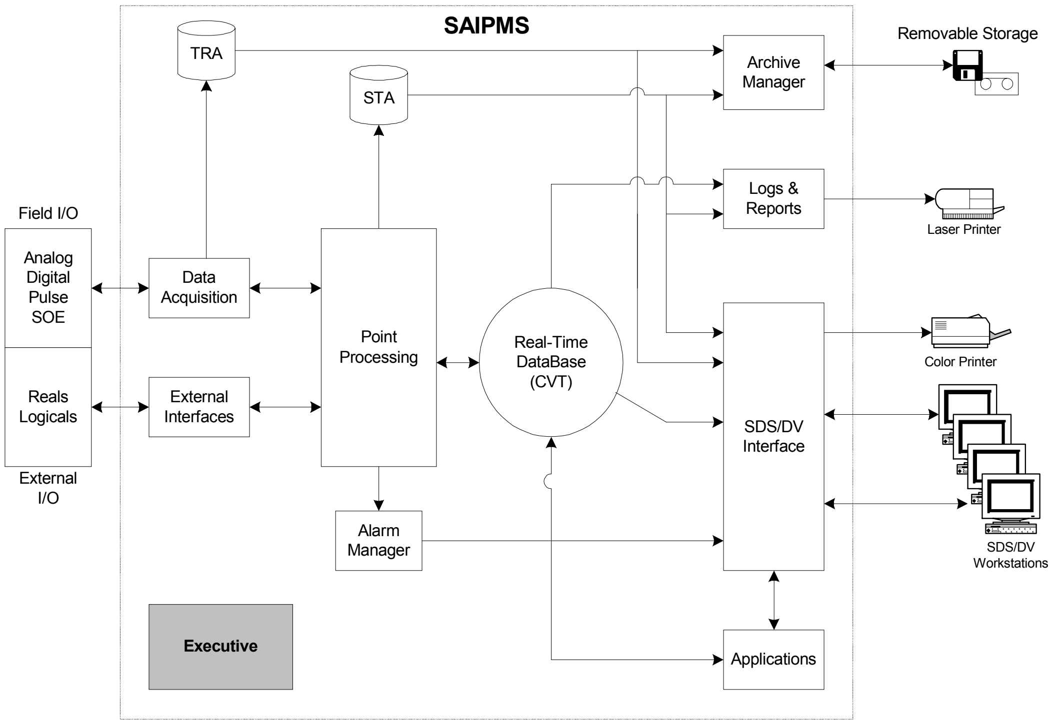

FIG. 2 The diagram of data transfer in SAIPMS

In conclusion, it should be mentioned that, since June 2000, the SPDS, representing 641 analog- and 370 discrete parameters, has been implemented at the ANPP Unit 2. The whole program-technical complex is operating normally, and it is a powerful information system supporting the operating personnel during the power unit operation.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety-Related Instrumentation and Control Systems for Nuclear Power Plants. 50-SG-D8, 1984, 50 pp.

[2] Functional design criteria for safety parameter display system for nuclear power stations. IEC 960,1988.

[3] W.R.Corcoran, D.J.Finnicum, F.R.Hubburd, et.all. Nuclear Power-Plant Safety Functions,-Nuclear Safety,Vol. 22.2.1981., p.p 179-191.

Panel Session

INTEGRATED PLANT LIFE MANAGEMENT— WHY AND HOW

E D F
Electricité de France

IAEA 2002

FRENCH NUCLEAR PLANTS LIFE MANAGEMENT PROGRAMME

REACTOR PRESSURE VESSEL INTEGRITY ASSESSMENT

G. BEZDIKIAN

Y. ROUILLON
J. BOURGOIN

CONTENTS

- Reactor Pressure Vessels assessment approach And Methodology considered by EDF
- Fluence level
- RT_{NDT} values and materials Parameters - Irradiation Surveillance Programme - Vessel by Vessel Programme
- In-Service Inspection Programme
- Mechanical analysis assessment of safety margin
- Conclusion
- Maintenance strategy

REACTOR PRESSURE VESSEL LIFETIME METHODOLOGY APPLIED BY EDF

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- The objective of the approach considered by EDF

  Demonstration of the integrity of the vessels in all conditions of loading, using last generation and evolution of mechanical studies and considering :

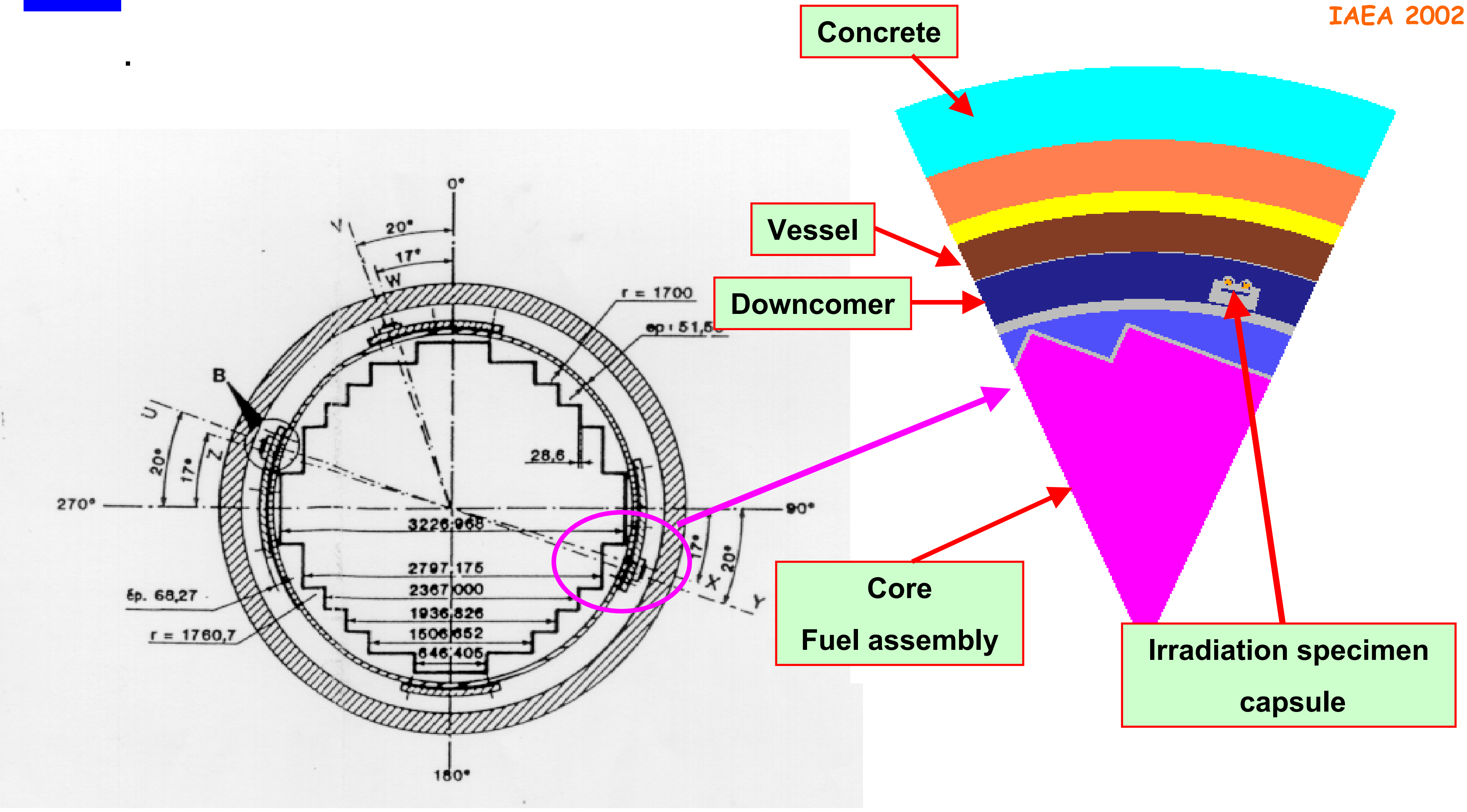
  - the RT_{NDT} at the end of life,
  - all parameters.

  The most severe conditions considered is the pressurized thermal shock and taking account for hypothesis shallow cracks beneath the cladding (subcladding area) or in the first layer of cladding.

- Justification of the Vessel integrity

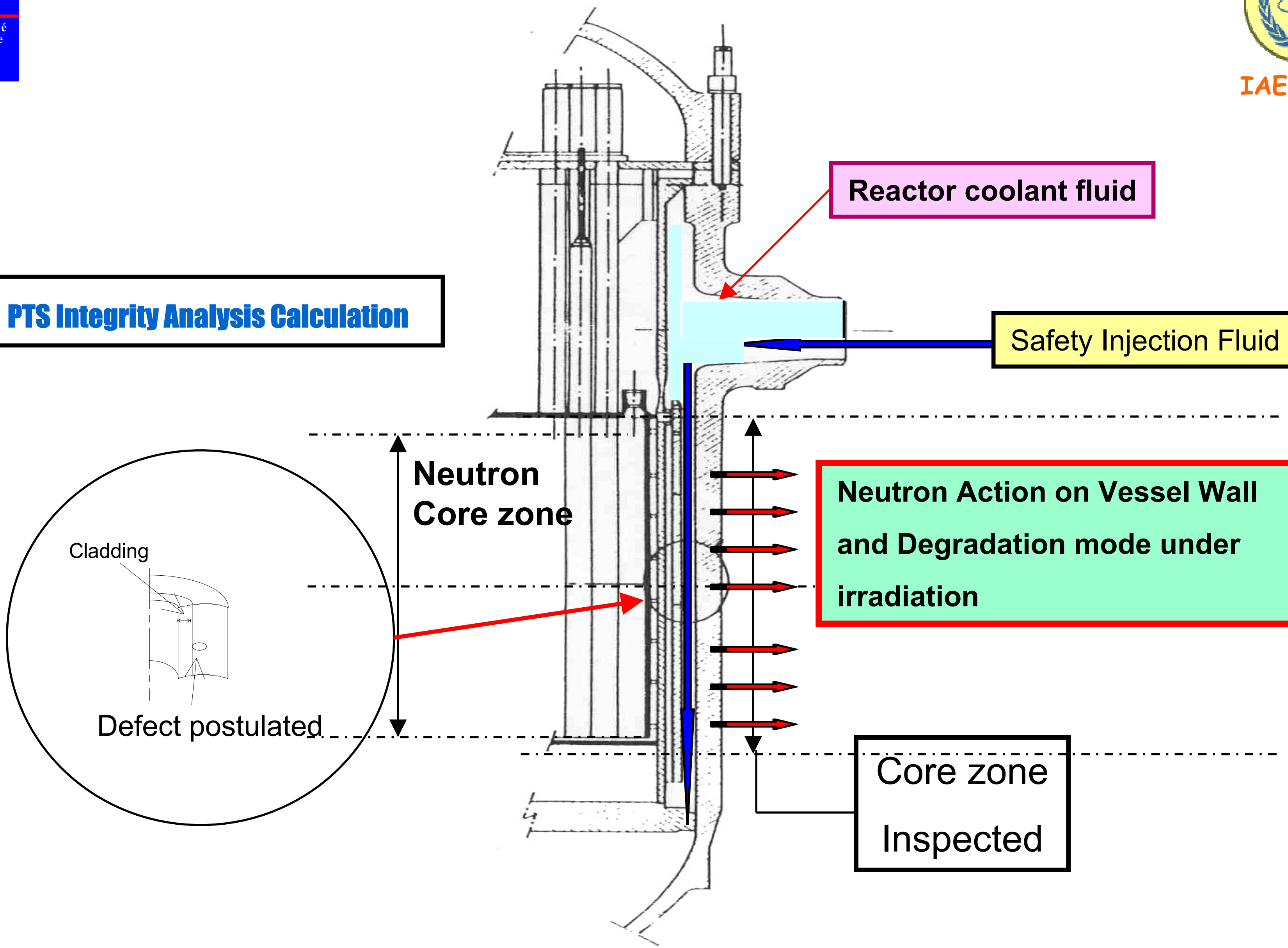
  Demonstration of the margin on

  - brittle fracture
  - ductile fracture (ductile tearing).

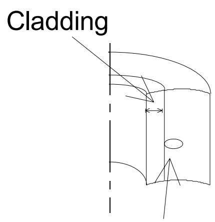


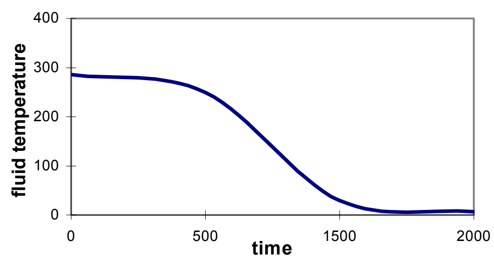
Concrete
Vessel
Downcomer
Core
Fuel assembly
Irradiation specimen capsule
0°
20°
17°
V
W
r = 1700
B
U
Z
28,6
270°
90°
3226,968
2797,175
2367,000
1936,826
1506,652
846,405
Ép. 68,27
r = 1760,7
X
Y
180°

PTS Integrity Analysis Calculation

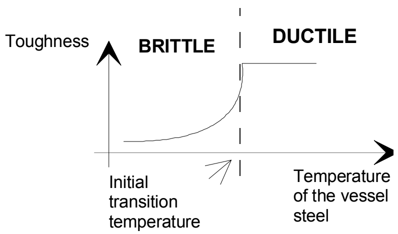


Assessment of RPV integrity : fluence calculations at EDF





Fluence   initial properties of steel



Defect   Transient

T(x,t) and σ (x,t) distribution

Stress intensity factor K_{CP} - computation at crack tips

Toughness K_{1C} - computation at crack tips

Margin factor K_{CP} / K_{1C}

- Demonstration, considering the risk of brille fracture , based on verification of the margins " **M** " (or Safety Coefficient **Cs**) in all cases of loading and each point of the structure in the wall

$$\text{"Margins" or "Cs"} = \frac{Kcp}{K_{1C}}$$

The demonstration is obtained by the verification of the fundamental relation

$$A_{(T)} \cap B_{(D)} = K_1 \leq \mathbf{M} \, . \, K_{1C}$$

K_{cp} = stress intensity factor "applied" on a flaw (D) postulated in subcladding area, with loading corresponding with the most severe transient (K_1 plastic correction)

K_{1C} = Stress intensity factor "resistant" depending on the temperature, the initial characteristics of the material and the effect of the damage under irradiation

TWO EVALUATION APPROACHES WERE CARRIED OUT
1 - GENERIC EVALUATION

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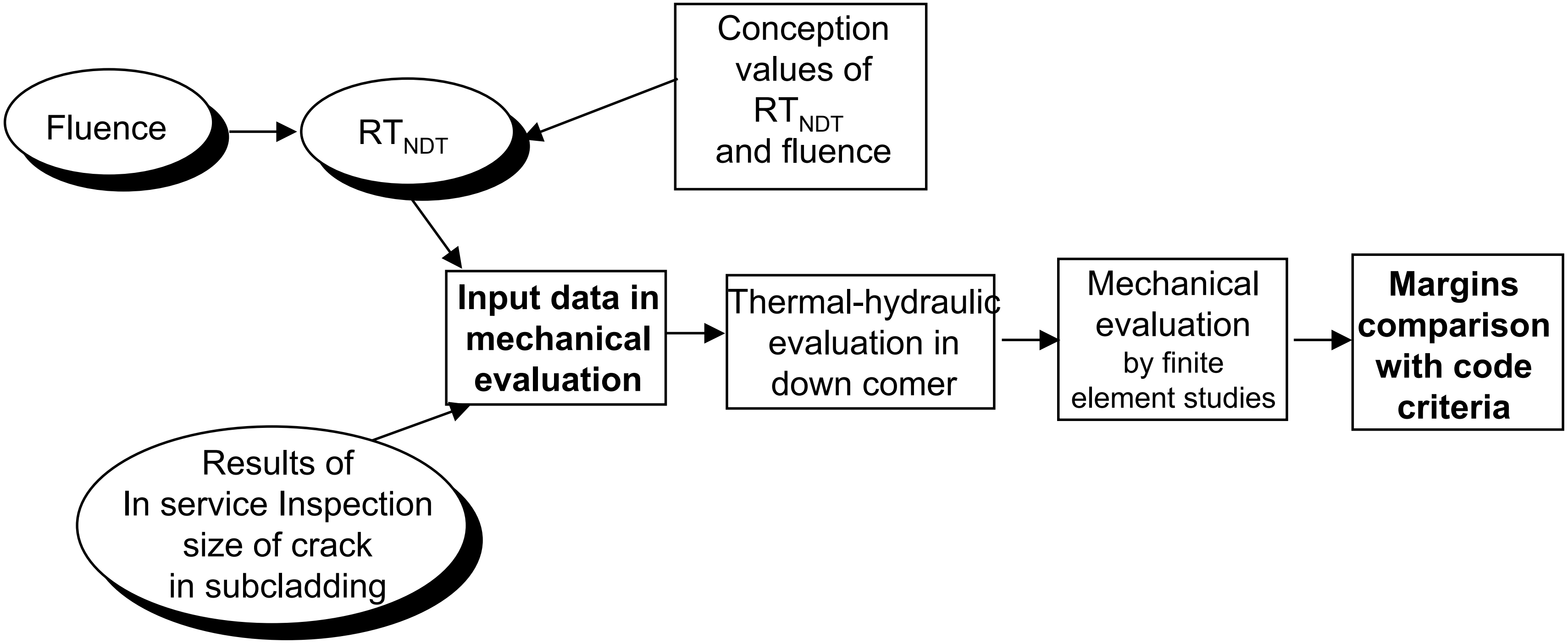
1 - GENERIC EVALUATION



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2 - SPECIFIC VESSEL by VESSEL EVALUATION

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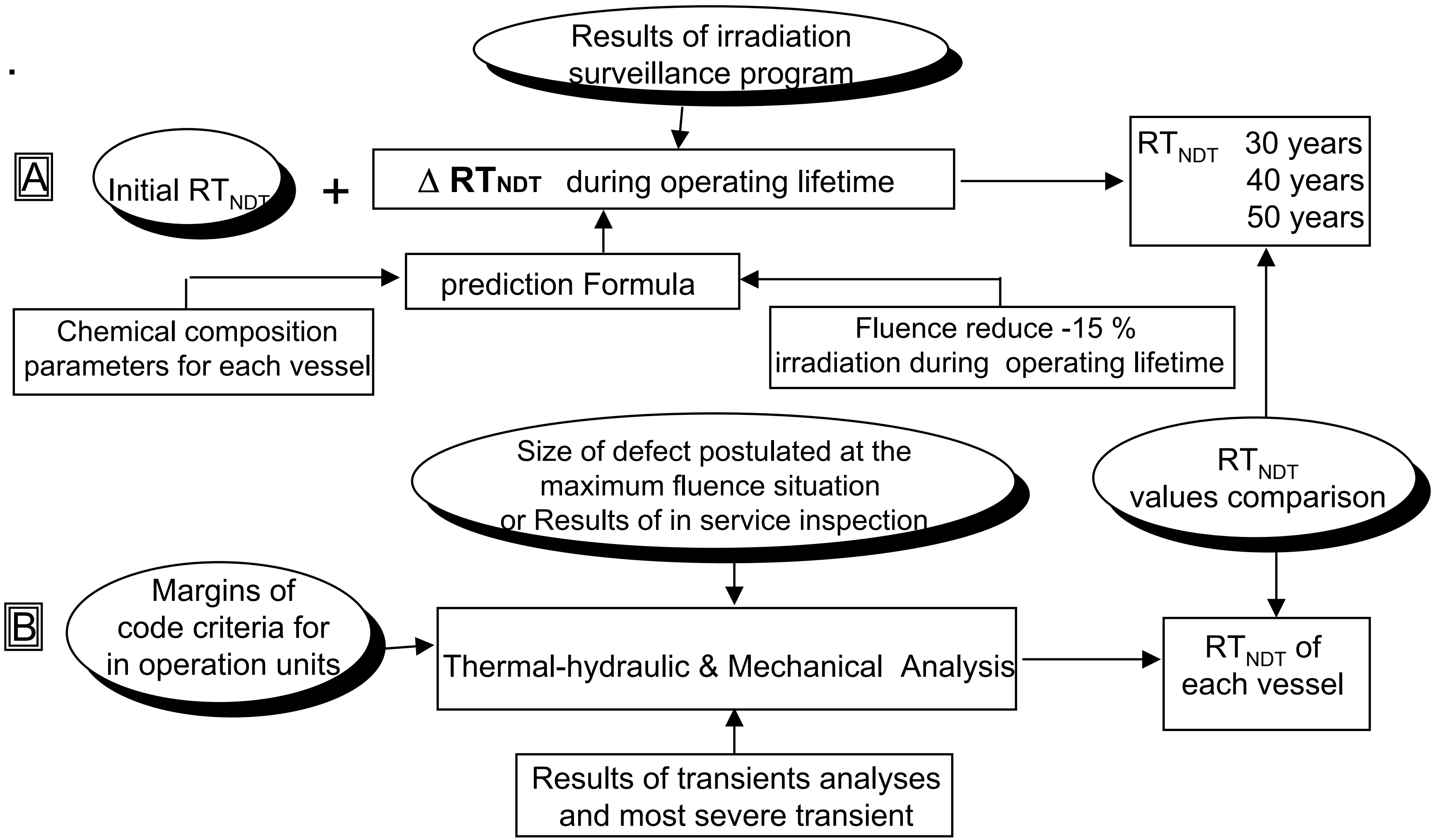

"M" or "Cs" is margin factor in codes criterias

- RCC-M (basic design conception)
- RSEM (for units in operating)

Level	Margin factor basic design - conception	Margin factor in operating
A	2.5	2
C	2	1.6
D	1.25	1.2

FLUENCE

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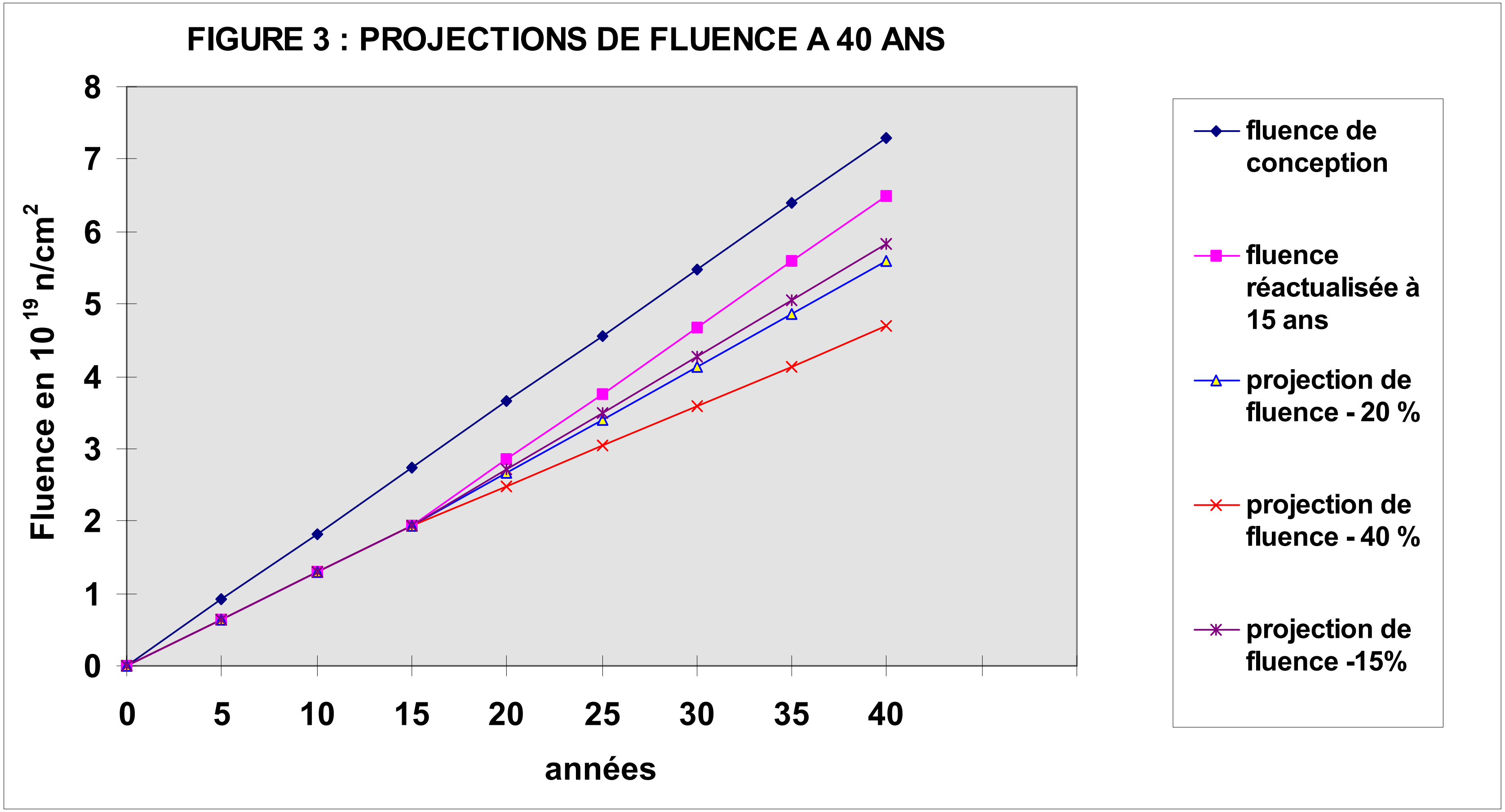
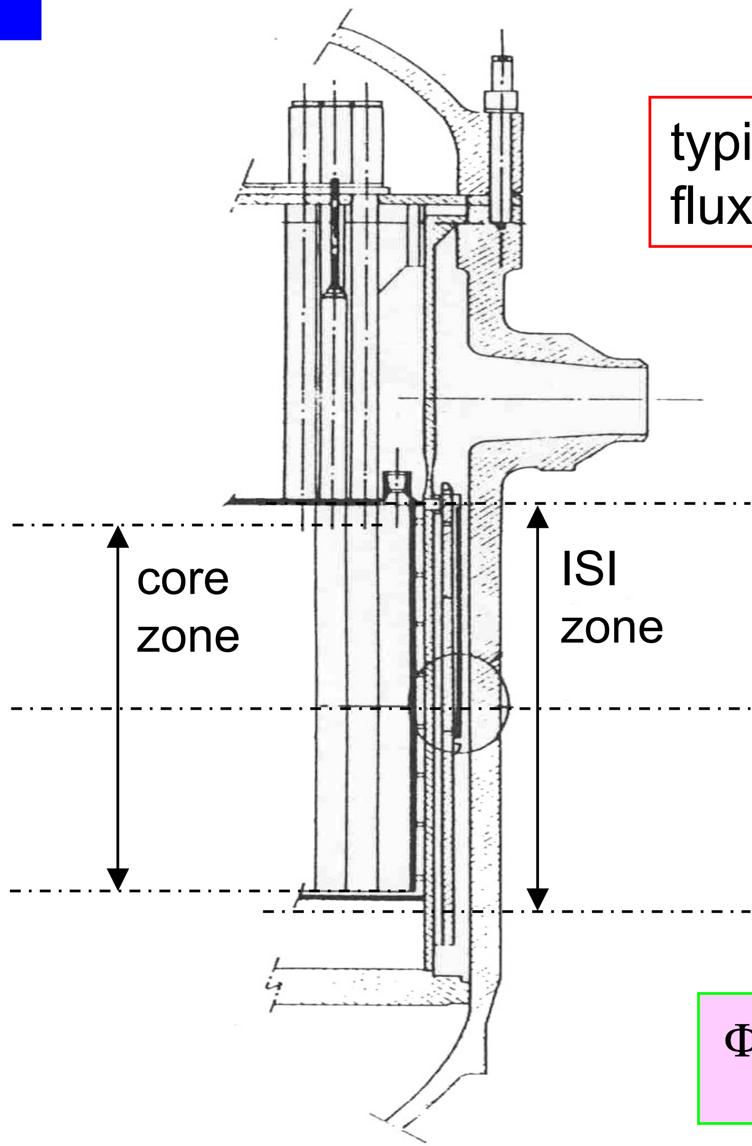
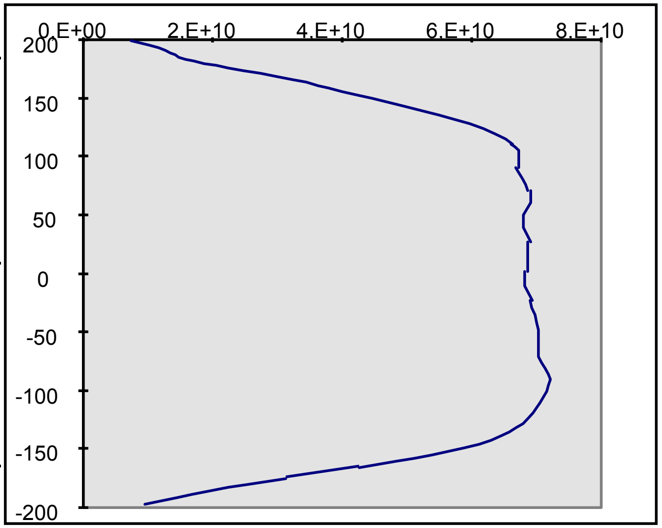


fluence calculations and Irradiation expertises Program " ESTEREL "



typical axial variation of flux in a 900 MWe unit

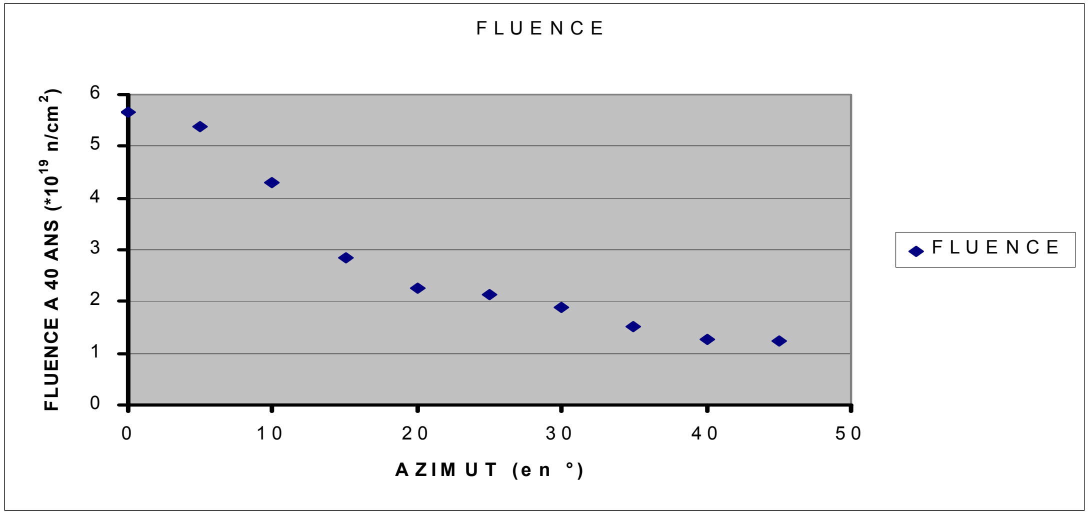
flux > 1 MeV ($n/cm^2/s$)



Φ > 1Mev **justified with a special irradiation program named "ESTEREL"**

FLUENCE

AZIMUTAL DISTRIBUTION OF THE FLUENCE



- **The basic fluence value for 40 years lifetime is : 7.3 10^{19} n/cm^2**

  Considering
  - **operating conditions for each NPP units,**
  - **all of fuel management plan program including fuel MOX.**

- **The actual fluence value for 40 years lifetime (considering the reduction of fluence) is : 6.5 x 10^{19} n/cm^2**

RTNDT VALUES - RPV SURVEILLANCE PROGRAMME - CORE ZONE VIEW

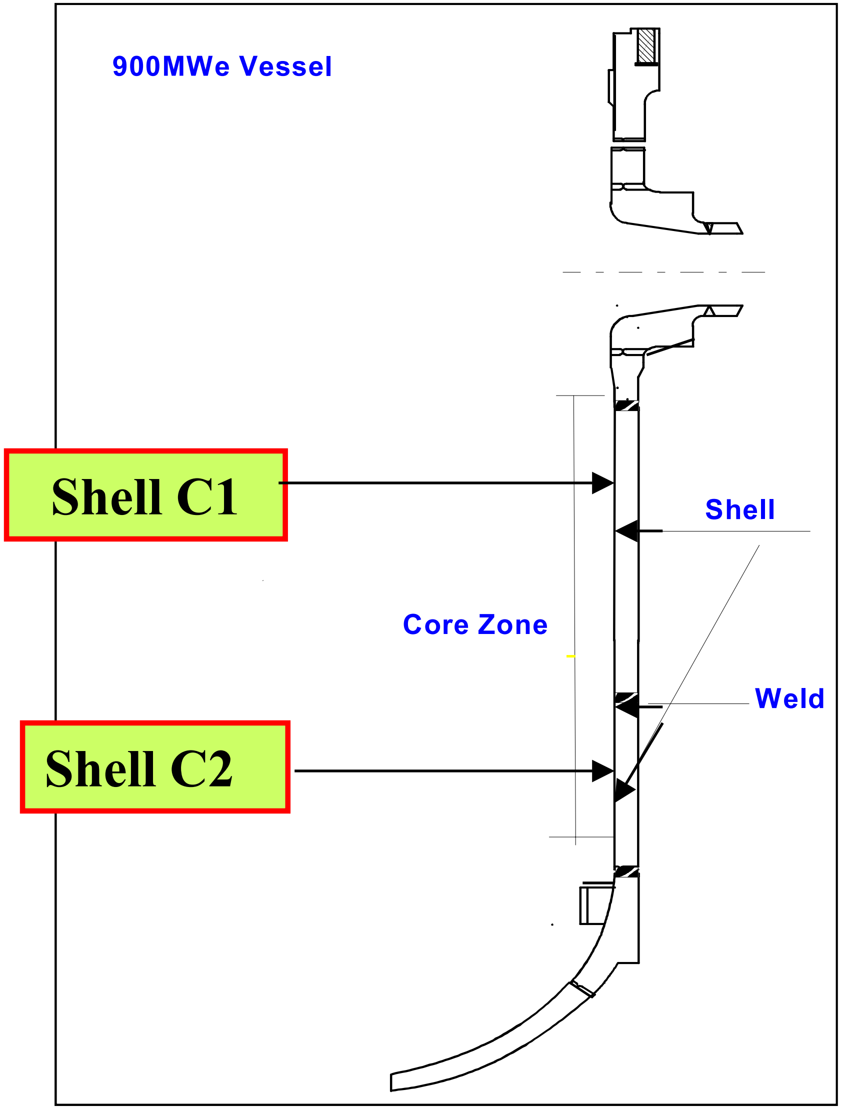
Reactor vessel is composed of several parts in low alloy ferritic steels (16 MND5) core shell zones subject to irradiation by neutron

- **vessel shell rings C1 and C2**
- **welds (B/C1 - C1/C2)**

<u>Embrittlement under irradiation and methodology for RTNDT assessment</u>

Transition temperature shift has for origin

initial RT_{NDT} + Δ RT_{NDT} under irradiation



E D F
Electricité de France

IRRADIATION SURVEILLANCE PROGRAM
EVALUATION OF RTNDT

IAEA 2002

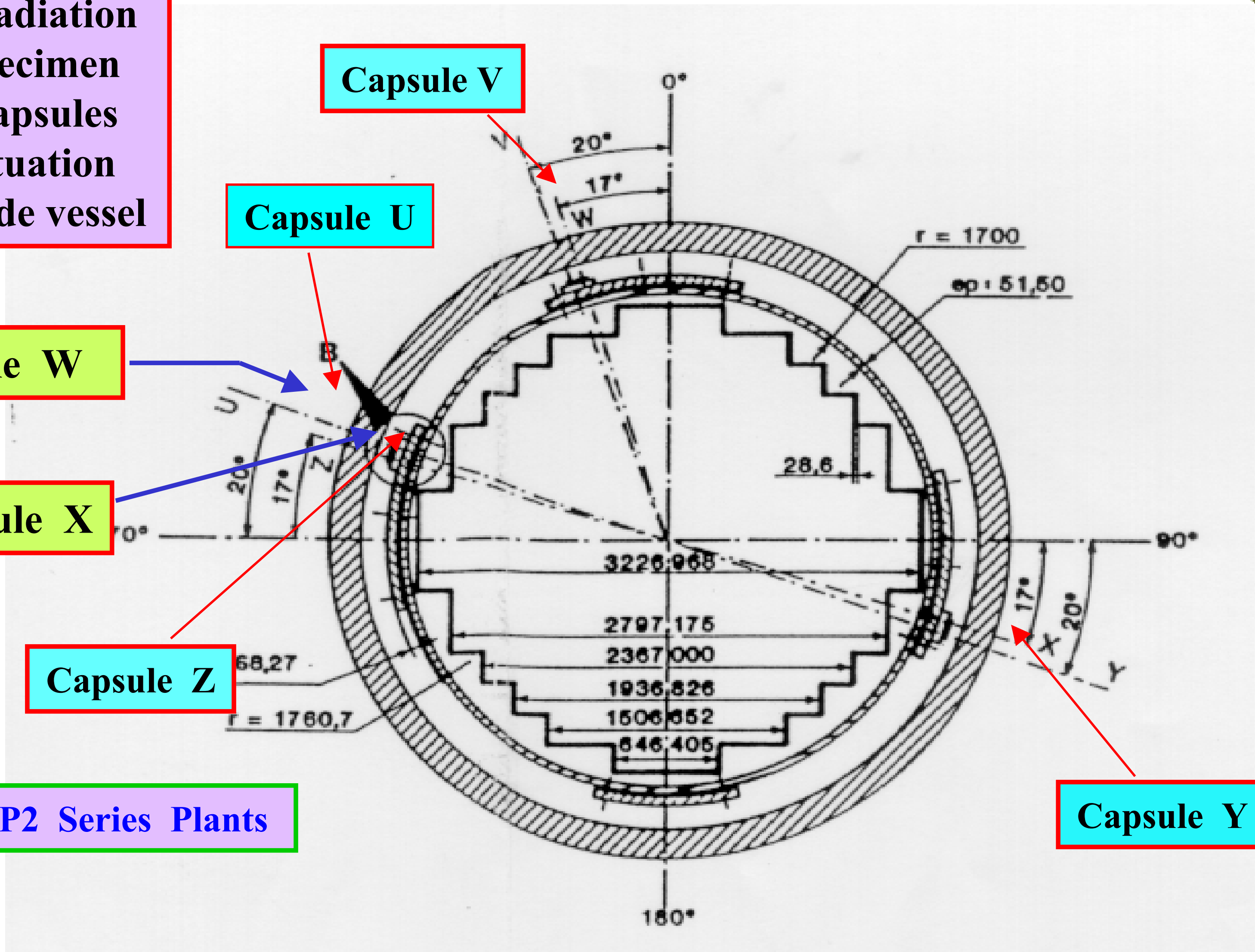
	Capsule 1 *(u)*	Capsule 2 *(v)*	Capsule 3 *(z)*	Capsule 4 *(Y)*
Time of duration in vessel (years)	***4***	***7***	***9***	***14***
Equivalent time of irradiation of the *Vessel (Years)*	***11.2***	***19.5***	***28.0***	***39.1***

Extension of the RPV irradiation surveillance programme based on the introduction of reserve irradiation capsules is engaged on the French plants since 1999 for all reactors.

Two reserve capsules W and X in place of capsules U and Z after removing from reactor these capsules U and Z.

E D F
Electricité de France

Irradiation specimen Capsules situation inside vessel



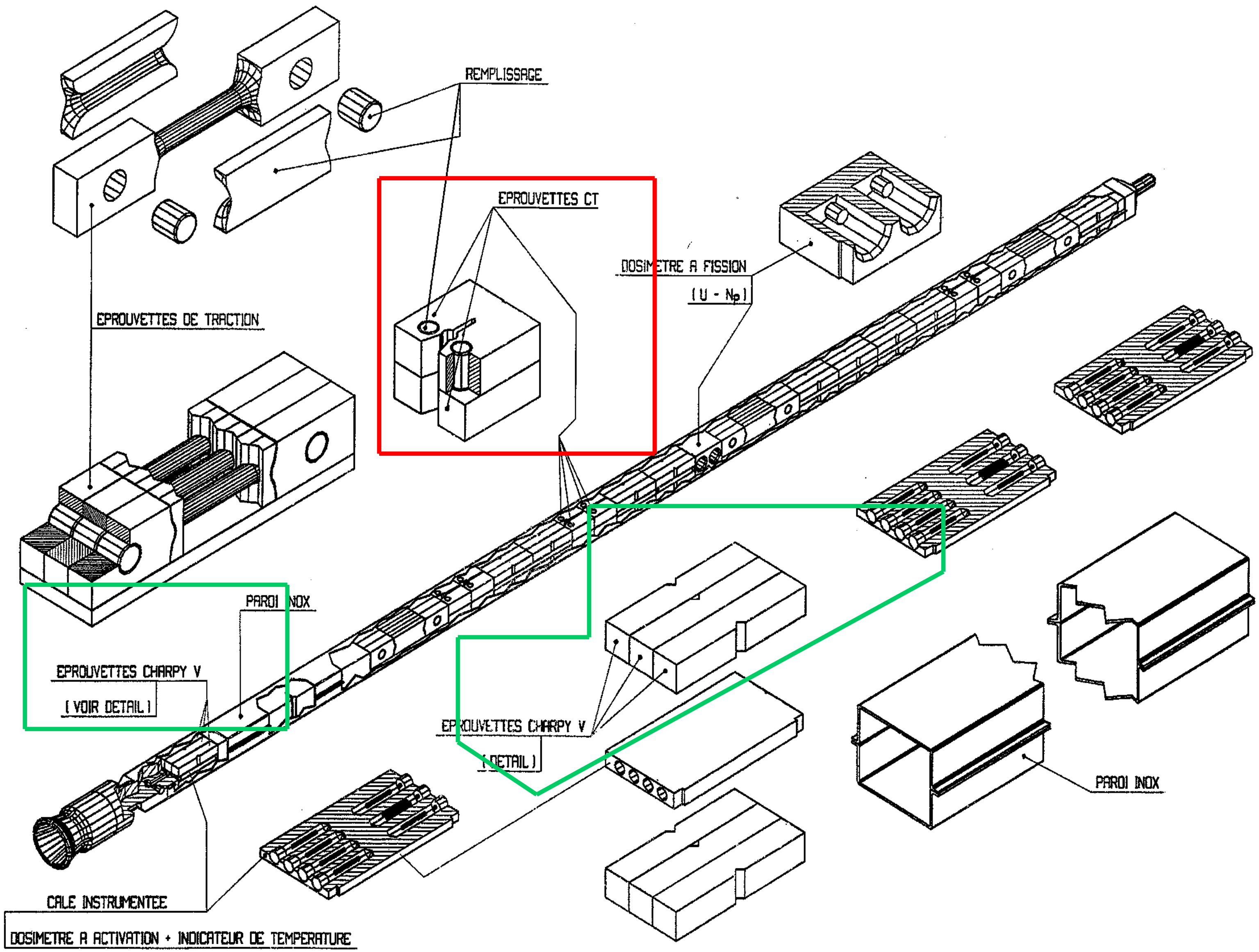
CP1 – CP2 Series Plants

CHEMICAL COMPOSITION - IRRADIATION CAPSULE
Results average values for 34 vessels Reactors 900 MWe

	Cupper	Phosphore	Nickel
Base Metal	0.04 - 0.08	0.005 - 0.013	0.64 - 0.84
Weld CP 0 Serie	0.09 - 0.13	0.012 - 0.019	0.07 - 0.51
Weld CP 1 - CP 2 Series	0.03 - 0.04	0.003 - 0.015	0.52 - 0.78

weight in %

IRRADIATION SURVEILLANCE CAPSULE



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- Vessel material embrittlement measured following the evolution and increase of transition temperature ΔT_{CV} .

=> T_{CV} is determined in function of resilience **CHARPY V** curve

$$\Delta RT_{NDT} = \Delta T_{CV}$$

RT_{NDT} at 40 years

- 3-loop CP0 series **RT_{NDT} = 82 °C**
- 3-loop CP1&CP 2 series **RT_{NDT} = 73 °C**

MECHANICAL ANALYSIS ASSESSMENT OF SAFETY MARGIN

IAEA 2002

- Evaluation of the brittle fracture margins in each cases (normal, upset, emergency, faulted)

– Two dimensional elastic analysis with input on dominant transient in level A - C - D,

– Three dimensional elastic plastic calculations for the most severe conditions combined with the last studies thermal-hydraulic finite element computations in the downcomer,

the most severe SB LOCA transients.

- Using RT_{NDT} values for 40 years, fluence level, all cases of loading, the results of mechanical computations justify code criteria margins:
  - » defect 6 mm deep postulated in most severe position in subcladding zone, for all vessels,
  - » defect 12 mm deep real position in subcladding for exceptional case

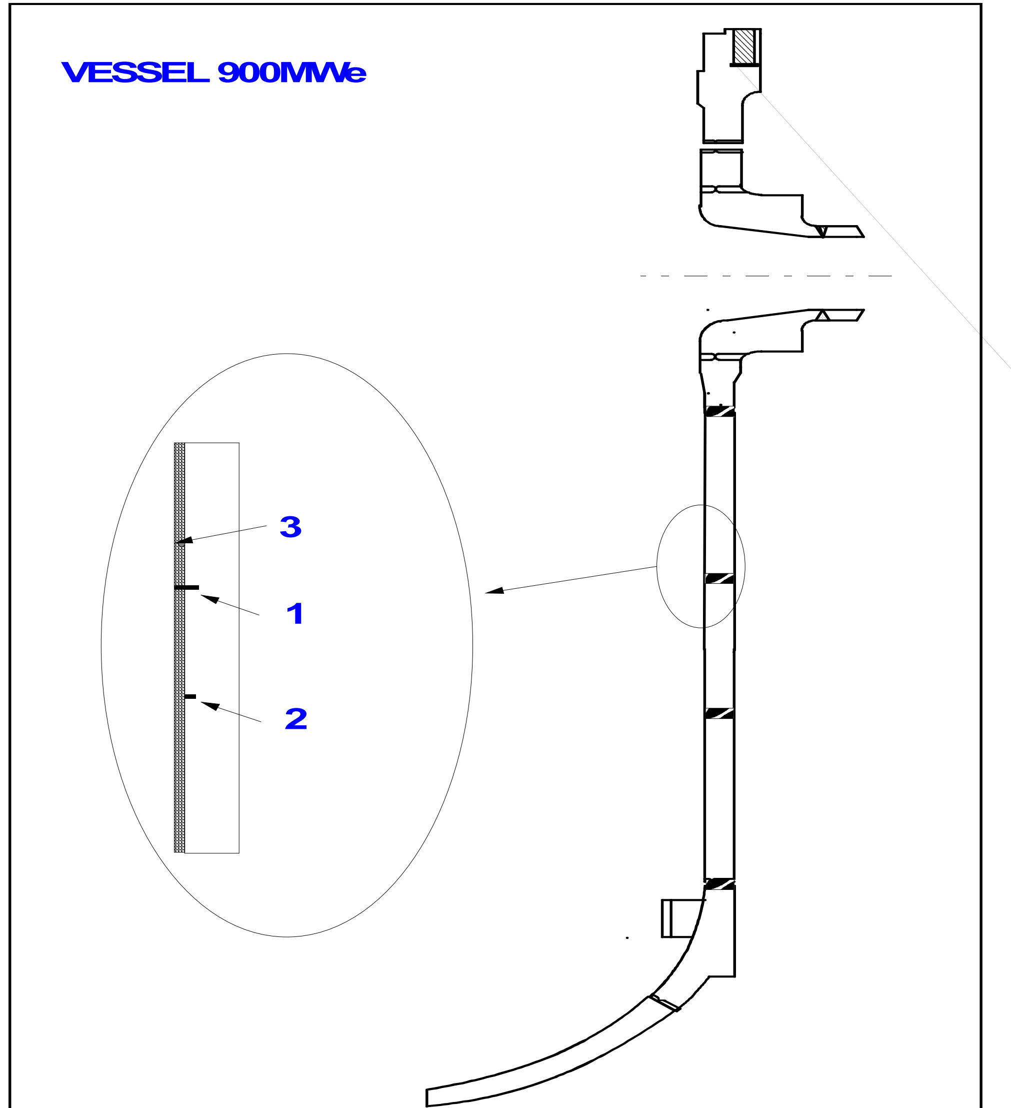
OBJECTIVES To verify safety margins regarding brittle fracture with margin in the code criteria for units in operating

INTEGRITY ASSESSMENT - MECHANICAL ANALYSES

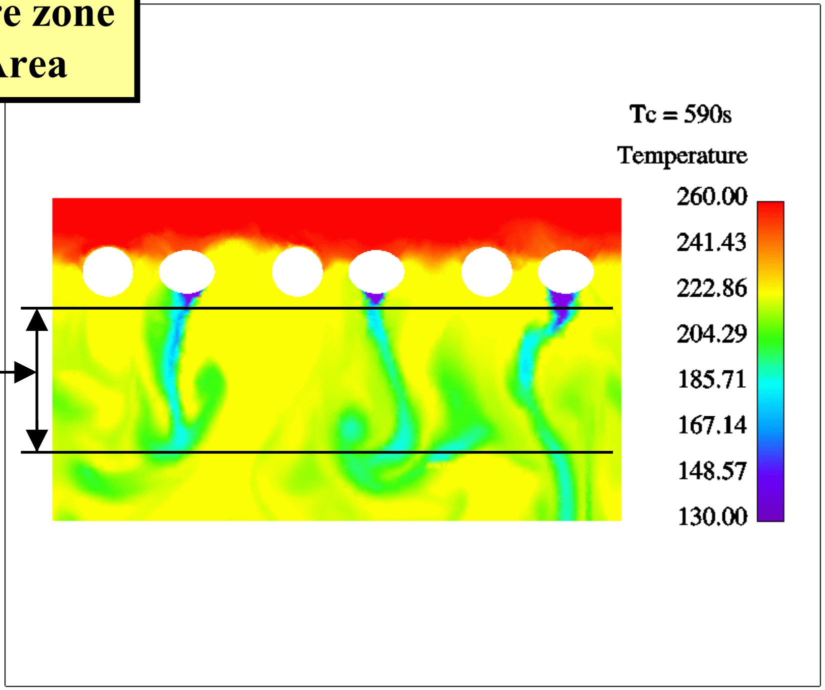
1st case studied : **1/4 thickness Throughwall defect**

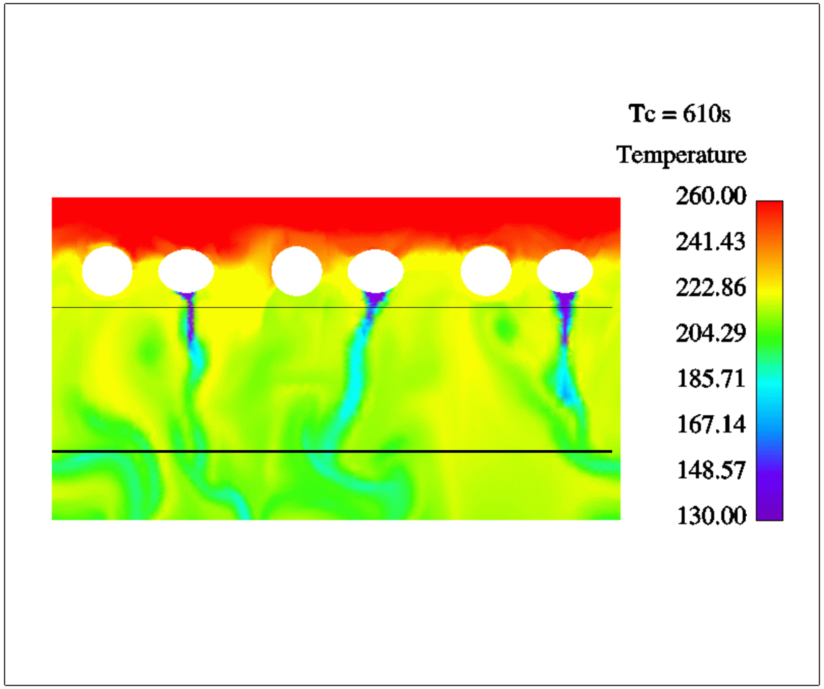
2nd case studied : **underclad flaw (6mm x 60mm) in level A - C - D**

3rd case studied : **defect postulated in the cladding in level A - C - D**



Core zone Area





Plume fluctuation on a developed downcomer section

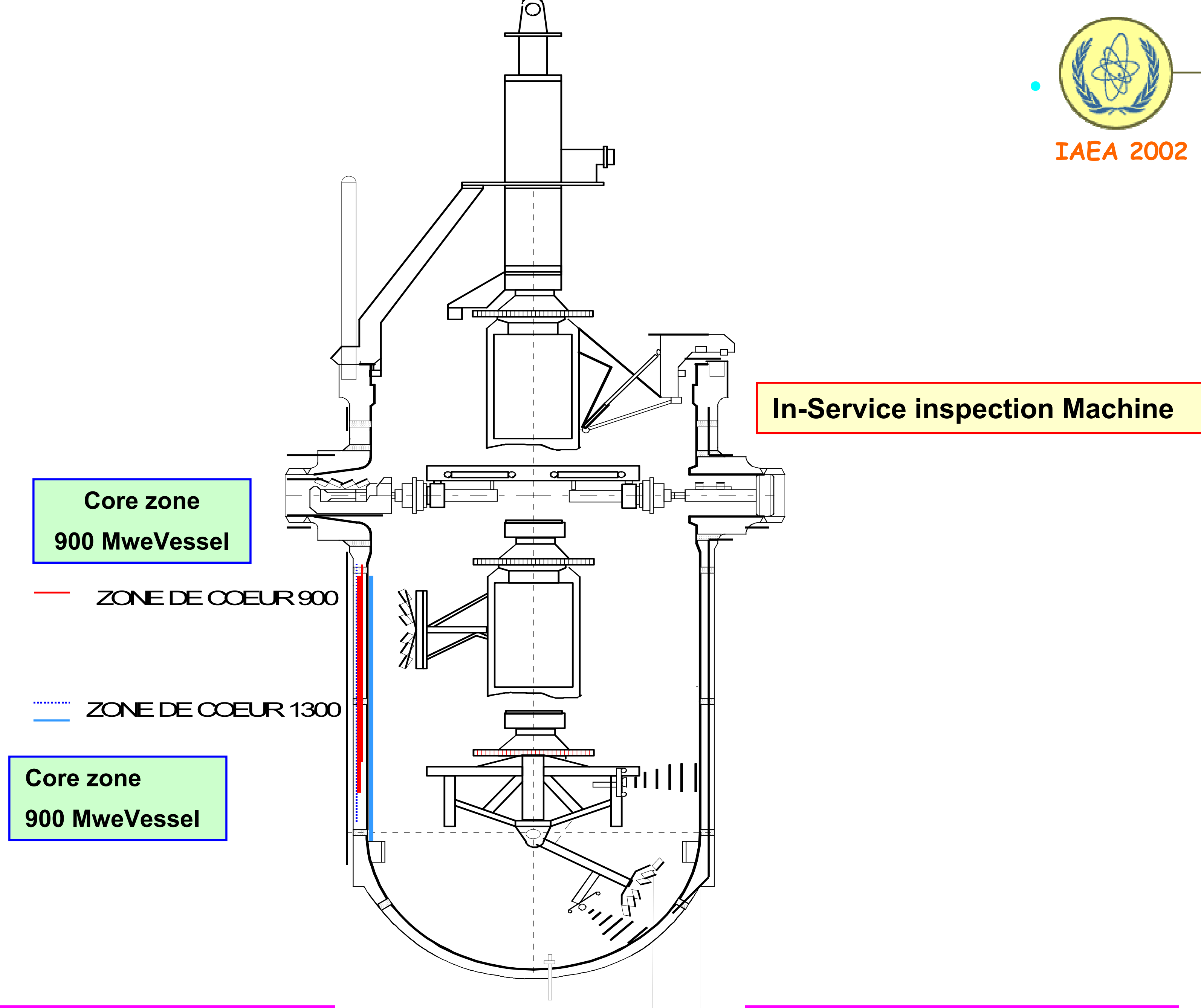
IN-SERVICE INSPECTION PROGRAM

- NDE in service inspection was engaged by EDF and focused on the core zone of each vessel :
  - with "VPM" NDE ultrasonic special tools, during the 2nd unit's 10-year outages - the 1st "VPM" inspection was begun at TRICASTIN 1

The in-service inspection results confirm the hypothesis on the input data for mechanical assessment, in term of postulated defect sizes (height of defect in subcladding area)

EDF
Electricité de France

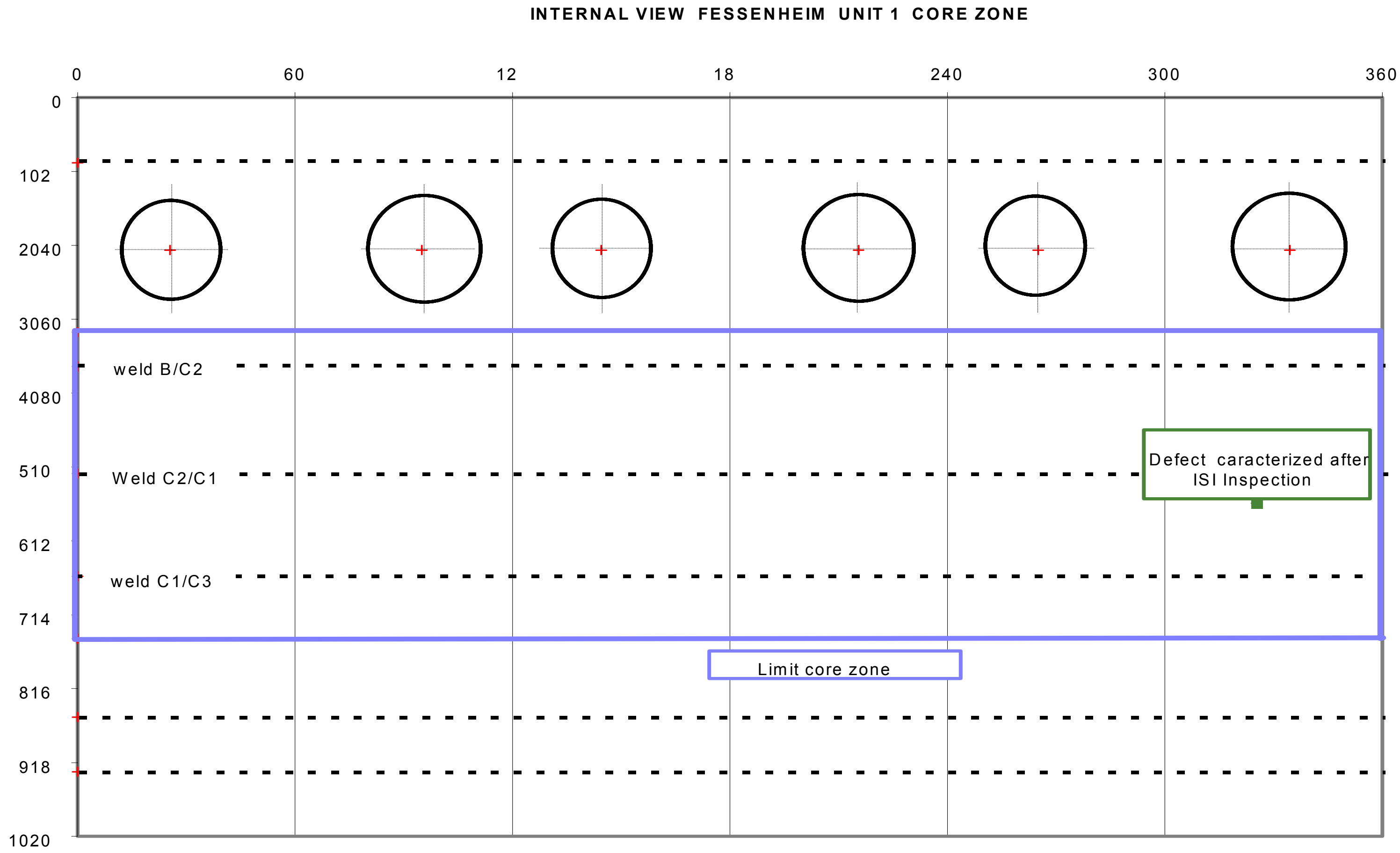
IAEA 2002



Inspection de la cuve avec la MIS
(machine d'inspection en service)

inspected by "VPM " UT tools core zone area

IAEA 2002



CONCLUSION MAINTENANCE PROGRAMME TO SUPPORT RPV SAFETY ASSESSMENT

- Specific fuel management plan monitoring vessel by vessel fluence level management ; objective → reduction of the fluence : with special code "EFLUVE "
- Continuation of "VPM" in-service inspection programme during the second 10- years outages for all of vessels and same inspection during the third 10-years outages.
- Extension of the pressure vessel irradiation surveillance program reintroduction of reserve capsules in reactor during the second 10-years outages.

  The objective is to confirm the resistance to embrittlement and to predict what might happen after 40 years.

CONCLUSION - APTITUDE IN OPERATION OF THE VESSEL

Justification of the vessels 900 MWe units at least 40 years :

- ***Fluence , RT_{NDT} ,***
- ***Dimensions of postulated defect (or results of in-service inspection) and introduced in mechanical thermal-hydraulic analyses,***
- ***Transient consideration (most severe consideration of transient) , and temperature of safety injection ,***
- ***material properties,***

Results obtained:

- safety coefficients greater than code criterias , and

- RT_{NDT} at 40 years lower than basic design RT_{NDT} value.

The fuel plan management with reduction of the neutron flux (-15%)

Fluence projection at 40 years = 6,5. 10^{19} n/cm² (fluence reduced)

(7,3 . 10^{19} n/cm² fluence value of conception) .

Result obtained with economy of fluence ,

RT_{NDT} < 82 °C ->CP0 units and in order of 73°C -> CP1/CP 2 units.

The reactor vessels of 3 loop series are apt in operation

For at least 40 years.

Socio-political issues for PLIM

maintaining existing plants
versus
decommisioning/re-building
('clean' plants)

issues ⟷ stakeholders

Socio-political issues for PLIM

Stakeholders:

- **"Structurised" SOCIETY**
  - **CITIZEN, staff in the sector**
  - **LOCAL COMMUNITIES, REGIONs**
  - **Member Satets (Ministery, regulators)**
  - **EU**
- **ENVIRONMENT**
- **Society at large**
  - **"world" - global energy sustainability**
  - **medium-long term availability of energy**

existing plants versus decommisioning/re-building

Socio-political issues for PLIM
"Structurised" SOCIETY

- **Individual risk perception (risk-education)**
- **Awareness - Acceptance - Media**
  (somehow negative public opinion)
- **Re-licensing iter versus new NPP approval iter**
- **Unemployment**
  (~100 NPPs, supporting sectors, regulators, R&D)
- **Regional issues – local economy**
- **National Energy plans/strategies**
- **Energy sources supply diversification**
- **3S, safe & secure supply for EU, availability, reliability**

existing plants versus decommisioning/re-building

Socio-political issues for PLIM
ENVIRONMENT

- **ENVIRONMENTAL impact**
- **Decommissioning wastes (>100 NPPs!)**
  (volumes! transports, radiation protection, etc.)
  "PLIM as tool for waste management"
- **CO_2 emission issues; at least during construction time of new 'clean' plants**
- **Environmental impacts of new plants** (nuclear/non-nuclear)
  - **land, water, siting (seismic studies, etc.)**
  - **civil works, ground movements, etc.**

existing plants versus decommisioning/re-building

Socio-political issues for PLIM

Society at large

- medium-long term availability of energy
- “world” global energy sustainability
- Global issues on energy sustainability
  - limit spoiling development countries
  - “low-tech” energy primary sources
  - ethical issues (next generations)
  - security (terrorism, etc.)

existing plants versus decommisioning/re-building

Socio-political issues for PLIM - Summary

"Structurised" SOCIETY

- **Individual risk perception (risk-education)**
- **Awareness - Acceptance - Media ; (somehow negative public opinion)**
- **Re-licensing iter versus new NPP approval iter**
- **Unemployment** (~100 NPPs, supporting sectors, regulators, R&D)
- **Regional issues – local economy**
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- **medium-long term availability of energy**
- **"world" global energy sustainability**
- **Global issues on energy sustainability**
  - limit spoiling development countries / "low-tech" energy primary sources / ethical issues (next generations)
  - security (terrorism, etc.)

existing plants versus decommisioning/re-building

Integrated NPP Life Management Panel Discussion

Economic Issues

Issue: Proliferation of Business Literacy

- Knowledge of the fundamentals of business literacy among management and operators of nuclear power facilities is diverse. Traditional focus and training development has been on achievement in areas of safety and operational reliability. With the transition to competitive generation, nuclear plant management and analysts at all levels will need to embrace and implement financial and business concepts into the day to day operations of nuclear plants. A much greater sensitivity to and awareness of fundamental business and economic principles will be required. The development of sound business literacy will become an imperative for all nuclear plant decision makers.

Issue: Key Measures of Economic Value

- In developing strategic and operational goals, nuclear plant managers will be required to embrace and articulate clear and measurable business objectives and goals which not only assure the achievement of safety and reliability but in addition eliminate unnecessary costs and identify investment opportunities which balance operating and safety risk while optimizing plant revenues, earnings and ultimately insure the profitability of electric generating facilities. In doing so, it will be essential for nuclear plant managers to articulate integrated goal achievement through the application of effective, measurable economic performance indicators which are understood by all and institutionalized within the plant organization and the nuclear power industry as a whole.

Issue: Integration of Safety Reliability and Business Objectives

- Traditionally, at the company or corporate level, regulated electric utilities were chiefly concerned with business matters of electric supply, demand and price including strategic and financial planning, capacity planning and electricity price and rate regulation. Strategies, goals and decision making concerning these business matters were more often formulated and measured on a consolidated basis often combining the impacts, risks and rewards of individual electric generating facilities within a given utility enterprise. With deregulation and open market pricing of electricity, the business and financial success of operating nuclear plants must be considered to a much greater extent along with the successful achievement of safety and reliability objectives.

Issue: Application of Efficient Business Analysis/Decision Tools

♦

Issue	National	International
Regulation Basis	Risk-based C.L.B.	Required vs. Recommended
Tech. Criteria	Detailed by Plant	Detailed by Type
Human & Culture	Detailed Guide	Models & Mgt School
Long-range Plan	Decommission	Spent Fuel Transmutation

Role of the Regional and International Cooperation

Dr. Katona Tamás János
NPP Paks
November 2002

IMPORTANCE OF THE INTERNATIONAL ENVIRONMENT IN THE NATIONAL DECISION MAKING

- **PRECONDITIONS OF THE PLEX DECISION at PAKS NPP**

international environment

- **new trends, changes in national energy policies (USA, Finland)**
- **international initiatives (IAEA, OECD NEA)**
- **feed-back to public acceptance**

- **CONDITIONS OF THE SUCCESS of Paks NPP Project**

international acceptance

IMPORTANCE OF THE INTERNATIONAL ENVIRONMENT

- transefer of best practices and experiences
- international acceptance of national projects and initiatives
- levelised requirements
- better communication of the progress of the industry

IMPORTANCE OF THE REGIONAL CO-OPERATION

- a „not to be alone" effect
- in the period of time of renewed licence (2012-2037) of Paks NPP several nuclear units (8 WWER/213 units in neighbour countries + 4 units of Paks NPP) will be operated in Central-Europe

CONDITIONS FOR AN EFFICIENT INTERNATIONAL CO-OPERATION

- concerning safety equal rules of game, although the national requirements might be different:
  - preconditions for PLEX
  - safety requirements
- transparency of national programmes
- well organised activity (e.g. use of experience of the IAEA)

DIFFERENT APPROACHES, THE SAME MINIMUM REQUIRED SAFETY LEVEL

- different regulatory framework and different approaches how to extend the plant operation life, but the safety level to be ensured should be the same:
  - national regulations should be consequent, easy to understand and easy to follow
  - adequate resources (human and financial)
- the international cooperation should focus on the essential problems and on the content of approaches (LR, PSR)

IAEA TC PROJECT RELATED TO PAKS NPP PLEX PROJECT

- POSITIVE EXPERIENCE (e.g. seismic safety project, MTC)
- EXPECTATIONS
  - know-how transfer due to the expert missions: **review-recommendation - improvement loop**
  - scientific visits: to learn the internationally accepted best practice, utility and regulatory experiences
  - adopt the advanced and internationally accepted methods, international experience
  - transparency of national programme
  - the TC project not equal to a cheap engineering support

ORGANIZATION OF THE IAEA TC PROJECT

- ACTIVITIES OF THE TC PROJECT LINKED TO THE NATIONAL PROJECT ACTIVITIES
- SCIENTIFIC VISITS AT THE BEGINNING OF NATIONAL PROJECT ACTIVITIES
- EXPERT MISSIONS LINKED TO THE “DRAFT”, PRELIMINARY OR INTERMEDIATE RESULTS TO BE ABLE TO USE THE RECOMMENDATIONS FOR THE CORRECTIVE ACTIONS

CO-OPERATION FOR THE PROSPERITY OF THE NUCLEAR ENERGY INDUSTRY

- The use of best practice and the international acceptance are very important success conditions.
- International co-operation is necessary for ensuring the minimum acceptable safety level.
- The intensive international co-operation may create very good climate for the national PLEX programmes.

Panel Session

Topic: Technical Issues

Š. ČEPČEK

SLOVAKIA

The title of a panel discussion is “Integrated NPP Life Management – Why and How?”

Why yes? This is a safety concern but at the same time it can be also an economical issue.

Why no? This is probably neither safety nor economical issue.

And How?

During the Symposium there were presented approaches of National Regulatory Authorities, the licence approaches and the concrete fully technical papers as well, dealing with specific ageing problems.

If we suppose that for the selection of the SCCs into Ageing Management Programme there exist justified criteria, the next issue is an identification, monitoring and evaluation of ageing mechanisms affecting concrete system, structure and component.

Generally three fundamental groups of ageing mechanisms can be identified:

1. Existing and known
2. Hypothetical (potential occurrence)
3. Unknown (possible existence)

How to manage them?

Group 1)

- implementation of appropriate monitoring tools
- measurement of parameters, treatment of them, trending and evaluation (calculation, surveillance, modelling)
- implementation of corrective action when acceptance criteria reached

Group 2)

- implementation of monitoring
- analyse the ageing effect if occur

Group 3)

- failure analysis
- corrective and preventive measures

As we could learn from the presentations, a big and continuous effort to the research and development is paid.

The main objectives of these works are:

- to study in more details the ageing mechanisms in order to increase the knowledge to better understand them

- to decrease the uncertainty (and conservativeness) of the evaluating methods and procedures, including the precision of safety criteria
- to utilize new monitoring and evaluation methods and their perspective further development
- to cover all safety related issues and ageing phenomena

In this field a lot of national and international projects are going on and/or are planned, to be introduced.

For the conclusion there can be stated:

There is need to continue in the further research and development project and in the inter-corporation of theirs results into Plant Life Management Programme.

At the same time following question can be raised in connection with Plant Life Extension. What is problem?

In many cases, the plant life extension I supposed to reach 10, 20 and may be more years beyond Design Lifetime.

During this time, it is necessary to assume development in all areas involving in the process, e.g. calculating method, monitoring technique, knowledge, etc. (hardware, software).

From these assumption following question:

IS THE DATABASE ON AM. DEVELOPED ACCORDING PRESENT KNOWLEDGE SUFFICIENT ENOUGH FOR THE FUTURE NEEDS?

HOW TO MANAGE THIS ISSUE?

The answers to these questions should give an Integrated Ageing Management Programme.

Panel Session

Topic: Existing experience and future plans

L. FRANCIA
SPAIN

- The world electricity demand is increasingly growing during the last years. The nuclear sharing in electricity production plays a very important role in achieving diversification in energy, supply sources and, particularly, it is and will be playing a vital role in limiting CO_2 emissions.

  The challenge facing the electricity utilities/companies and the governments is how the energy sector will expand to meet this increasing demand both without putting the environment at further risk and at the lowest possible cost.

  We believe that the role of the existing NPPs and, specifically, the long-term operation of their lives will be vital to meet this challenge.

  Also, under the economics point of view, the issue is very straightforward. Based on our own experiences during the last 10 years in the refurbishment of our older NPPs, and also on the assessment to ascertain which equipment and systems will be necessary to refurbish or replace in order to operate beyond the design life (up to 50 y.), while keeping the same level of safety and reliability, the cost of the kWh produced in the extended period is estimated to be very competitive comparing with the total generation cost of a new combined-gas cycled power plant or the present electricity price in the Spanish electricity production market (pool).

  In summary, the long-term operation of the NPPs (beyond 40 years) will be very economically attractive in a deregulated market and will be further due to the consequences of the Kyoto commitments to restraint the CO_2 emission.

- As a result of the attractiveness of the Long-term Operation (LTO) of the Spanish NPPs, the Spanish Electricity Sector established the objective of operating their NPPs beyond the initial design life of 40 years, maintaining the safety and efficiency levels. As a first step, to achieve this strategic objective, the present goal is to control and monitor the ageing of the important NPP components in order to guarantee an operating life of 40 years while technically keeping open the option of operating the plants beyond 40 years.

  Thus, a PLIM methodology was developed by the Spanish utilities, through UNESA, to manage the ageing and life of the SCCs of their NPPs. This methodology, evaluated and accepted by the Spanish Nuclear Regulatory Authority (CSN), has been described this morning.

  All the Spanish NPPs have implemented, or are in course of implementing, individual Life Management Programmes based on the above-mentioned common methodology. The CSN requires to be informed annually about the progress or updating of each programme.

  In addition, the CSN requires the performance of a Periodic Safety Review (PSR) every 10 years to provide a global evaluation of the plant safety. The PSR is the main tool used by CSN to grant and established the additional requirements for a new Operating License/Authorization (typically for a 10 years period). In this way, the NPP Operating License can be periodically renewed, given that the license life is open in Spain and no legal time limit exists for NPP operation.

- With respect to the existing Spanish experience in PLIM, I have addressed it in my this morning presentation. So I am going to focus in our future plans.

  Just to say that implementing PLIM programmes contributes:

  – To improve the NPP safety level, as the plant components are studied and evaluated to control their ageing, and
  – To help reducing the cost of kWh generated. Implementing PLIM Programmes result in decreasing the NPP unplanned unavailability factors. In this way, you can increase the kWh production at the same O&M cost.

- Which are our future plans?

  The utility of one of the Spanish NPP closer to reach its design life (2001) has announced its intention to run the plant beyond this life.

  So, now in Spain we are in the phase of developing the regulatory framework to apply or to be required to the NPP long-term operation. That is, we are going to begin a transition time from PLIM to LTO.

  The CSN has issued and presented to Spanish utilities draft documents anticipating the conditions that will rule the safe operation of the NPPs beyond 40 years design life (LTO).

  These documents, supposed to be in the future safety guides, are now under discussion by CSN and Spanish utilities.

  The basic criteria and conclusions under consideration for the NPP LTO are mainly:

  – To consider PSR as the basic tool to get the operating license beyond 40 years.
  – It will be necessary to add to the analyses in the PSR those that prove the safe operation in the extended period:
    * Safety report updating.
    * Review of Tech. Espes.
    * Review of PLIM methodology (scope of 10CFR54 / NEI 95-10).
    * Environmental Radiation Impact Study.

  From the Utilities side, our position would go to support the maintenance of the License Basis and/or Design Basis, in general terms, aware being that the exemptions will have to be justified. Also, those cases in which the compliance with safety requirements is based on a time-limited life design (TLAAs) must be identified and re-analysed (mainly passive components: vessel, internals, piping, …).

  Also, it would be good to give full credit to the actual programs in force, mainly those related to maintenance associated to active components.

- In this all-above context, any official document from IAEA, that I am aware is going to be produced, will be welcomed. There are many valuable IAEA documents, in certain sense developed to cover and clarify many items of interest.

  But a qualified/official methodology or methodologies with criteria confortable to both Utilities and Regulatory Bodys on NPP LTO or NPP operation beyond the design life will be well received from IAEA, this allowing Member States to support or develop their own domestic regulations.

Panel Session

Topic: Regulatory Approaches for Assessing Life Management Activities in NPPs: Key Issues

PH. TIPPING
SWITZERLAND

From a regulatory standpoint: Why PLIM?

• **PLIM should contribute significantly to increasing public safety by enhancing the reliability of all aspects of NPP operation (e.g. integrity of pressure boundaries, avoidance of failure in service of safety-related components, avoidance of scenarios that could lead to accidents etc.). There are basically two aspects of PLIM as below:**

Structures, systems and components (SSCs)

1) NPPs must avoid uncontrolled releases of harmful substances to the environment or that such substances arising from the operation of a NPP will endanger the public in any way. There are many ways this can be done. One way is to ensure that NPPs possess SSCs which always satisfy technical specifications (even when ageing is present). Designed-in levels of conservatism are an *a priori* acknowledgement as to the potential effects of ageing in reducing safety margins. SSCs must accordingly be maintained and inspected and replaced, modified or repaired to guarantee both integrity and sufficient safety margins. Changes in operational conditions may also contribute to maintaining safety levels or reducing the rate at which these may degrade. Examples are neutron shielding (RPV embrittlement) and hydrogen water chemistry in BWR (corrosion issues).

Human aspects

2) Not only are SSCs subject to ageing, but operational procedures and personnel may also become "outdated" and require progressive adjustment or training respectively. As operational experience is gained within a given NPP then the need arises to assure that the transfer of knowledge and retention of data is also guaranteed. Herein lies a necessity for the management of change (procedures) according to the NPPs new situation. Record keeping is essential to maintain know-how, and would appear to be an integral part of PLIM strategy.

SOME PERCEIVED KEY ISSUES CONCERNING PLIM:

(Not in any order of importance)

• Ageing of systems, structures and components (SSCs) takes place in all technological devices or installations and NPPs are no exception. Comprehensive knowledge about and control of known relevant ageing mechanisms are essential if ageing is to be managed to the advantage of both nuclear safety and the NPP's economics. As NPPs age further it is important to increase vigilance for the occurrence of any new ageing mechanisms that could manifest themselves (e.g. very high neutron fluences, fatigue life change due to environmental effects, long time at temperature and consequential phase transformation of alloys previously thought to be stable etc.) The possible effects of any changes in operational methods (temperatures, pressures, water chemistry, etc) on ageing rates must be monitored and where necessary, new methods must be developed toward this end.

- With liberalised electricity markets, it may be that, for example, NPPs will apply for changes in inspection intervals or operational procedures to accomodate for increased economic pressures on the utility. Inspection intervals, for example, can only be modified when enough data and experience has been accrued to establish benchmarks and also regulatory comfort. These and any other measures taken require a) increased regulatory supervision and b) closer collaboration between all persons operating NPPs. Awareness of possible far-reaching effects on safety due to any economically-based changes will be essential to continued safe operation. Much work seems to be needed here.

- How to measure the actual "level" of safety margins available remains a problematic issue. It is mostly based on an "experience value." How safe is safe? How much conservatism is "reasonable"? In this context, until more data and experience are available, we are confronted with subjective arguments, and have to still ensure safety through conservative approaches. There seems a necessity to identify ways by which possible excessive conservatism may be reduced while not lowering safety by any significant level. Both probabilistic and deterministic approaches could be examined.

From a regulatory standpoint: How to achieve integrated PLIM

NPP life management is a complex interaction of many things (SSCs, operating conditions, personnel etc.) It is therefore essential to integrate these aspects in such a manner as to create a global approach towards recognising all relevant ageing mechanisms, identifying crucial SSCs, optimising operational conditions (avoidance of transients etc.) ensuring personnel are well-trained etc. and thereby measurably enhancing safety levels.

- The IAEA TECDOC (TD) series of publications, dealing with the the "Assessment and management of ageing of major nuclear power plant components important to safety" provide a wealth of essential information on items such as cables, instrumentation, pressure vessels, steam generators etc. There are many of these documents available. What appears to be needed now is a <u>single</u> condensed document embracing all the very best practices in PLIM known to date, for each type of NPP, plus a guide for strategies needed from the design to decommissioning phases of a NPP. Both regulators and operators will be aided by a "Recommended and Best Practices Method" , since even if there are PLIM programmes already established for most NPPs, these can be benchmarked and where necessary, brought up to the state of the art.

International Symposium on Nuclear Power Plant Life Management

Budapest, Hungary
4-8 November 2002

SERVICE LIFE MANAGEMENT ISSUES OF NUCLEAR POWER UNITS IN OPERATION AND UNDER DEVELOPMENT - TECHNOLOGICAL ASPECTS

Dr. Boris A. Gabaraev
General Manager
Research and Development Institute of Power Engineering
Russian Federation

Most owners intend to run their plants for as long as their operation is safe and economically efficient:

- tough competition among power producers;
- mounting capital costs of new construction;
- national economy development;
- prevention of CO_2 emissions (Kyoto Protocol)

- PLIM is an integrated optimisation process aimed at safe and cost-effective operation of a plant for as long as this is feasible (regardless of the lifespan specified in design documentation)
- Such a scientific and technical approach to reactor and plant life management, which treats continued operation and decommissioning as one unbroken process, is basic to long-term strategic planning.

Efficient improvement of the performance and safety characteristics

- diagnosis of the current state of the power unit

plus

- allowance for the conditions throughout the life cycle ("pre-operation" "operation" "post-operation")

"Pre-operation" stage:

- design,
- construction,
- commissioning

"Operation" stage:

- operation during the design service period;
- preparations for continued service;
- continued operation (operation license renewal or conversion activities);
- preparations for decommissioning

"Post-operation" stage:

- mothballing,
- entombment, or
- dismantling and disposal

PLIM objectives:

- maintaining/improving the plant safety level;
- keeping the components and the plant as a whole serviceable and durable

PLIM provisions, made throughout the ***life cycle,*** have the following aspects:

- regulatory;
- methodological;
- information;
- technological

A key to PLIM success is robust consistency between these aspects, which are supported by databases established and built up starting with the very first stages of the life cycle.

The feasibility of continued operation depends primarily on the condition of the so-called “critical” components which vary with different reactor facility types.

PWRs and VVERs:

- reactor vessel;
- primary components (including steam generators and pressurisers)

AGRs and RBMKs:

- metal structures of the reactor;
- graphite stack;
- large components

Generally, such components are nonrepairable and nonrecoverable, and their replacement is impossible or inexpedient for technical or economic reasons. ***It is these items that restrict the residual service life of a nuclear power unit.***

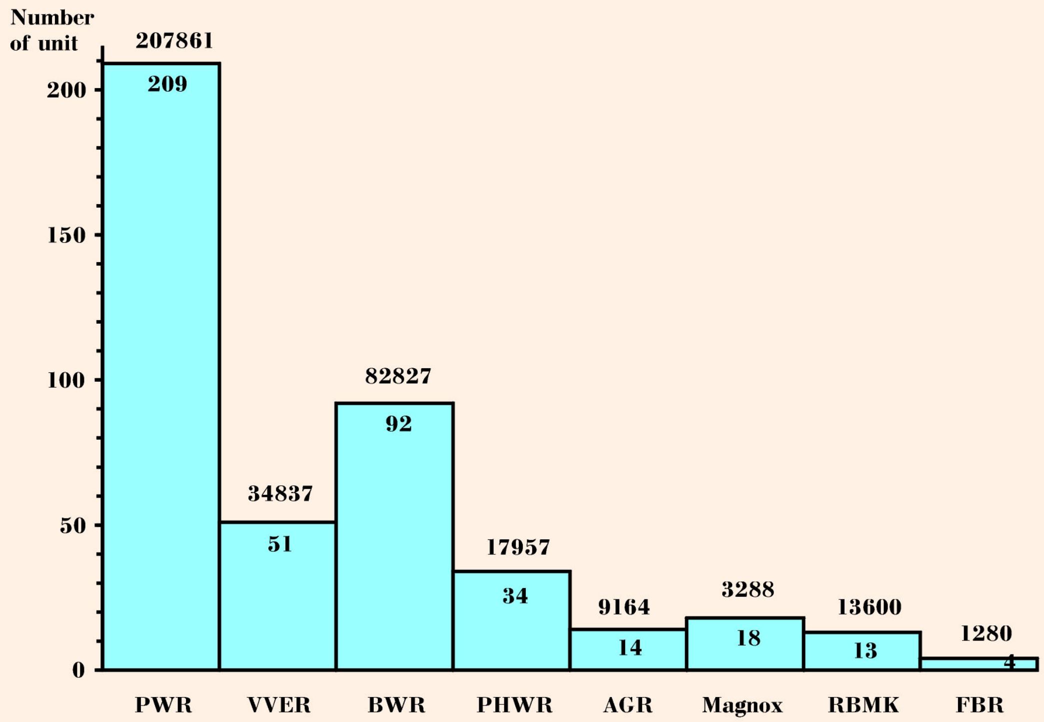
The stage of “operation” proper includes the following procedures:

- in-service inspection of welds and overlays in pipelines and components by nondestructive techniques (UT, TV examination, measurements, etc.);
- repairs by automatic welding;
- treatment of components to prevent possible damage during operation;
- in-service testing and monitoring of mechanical properties and crack resistance of equipment and pipeline metals;
- water chemistry monitoring;
- modern repair and maintenance methods;
- management of operational radwaste;
- SNF management.

The following unified processes and tools belong to the "post-operation" stage:

- decontamination and prolonged monitored storage;
- RW and SNF management methods;
- dismantling of contaminated structures and components;
- disposition of contaminated structures and components.

Fig.1. Nuclear power units with various reactors perating in the world Distribution in number and capacity



Figures inside and above the bars show the number of generating units and total capacity (MWe), respectively

Fig.2. Plant life management (power unit)

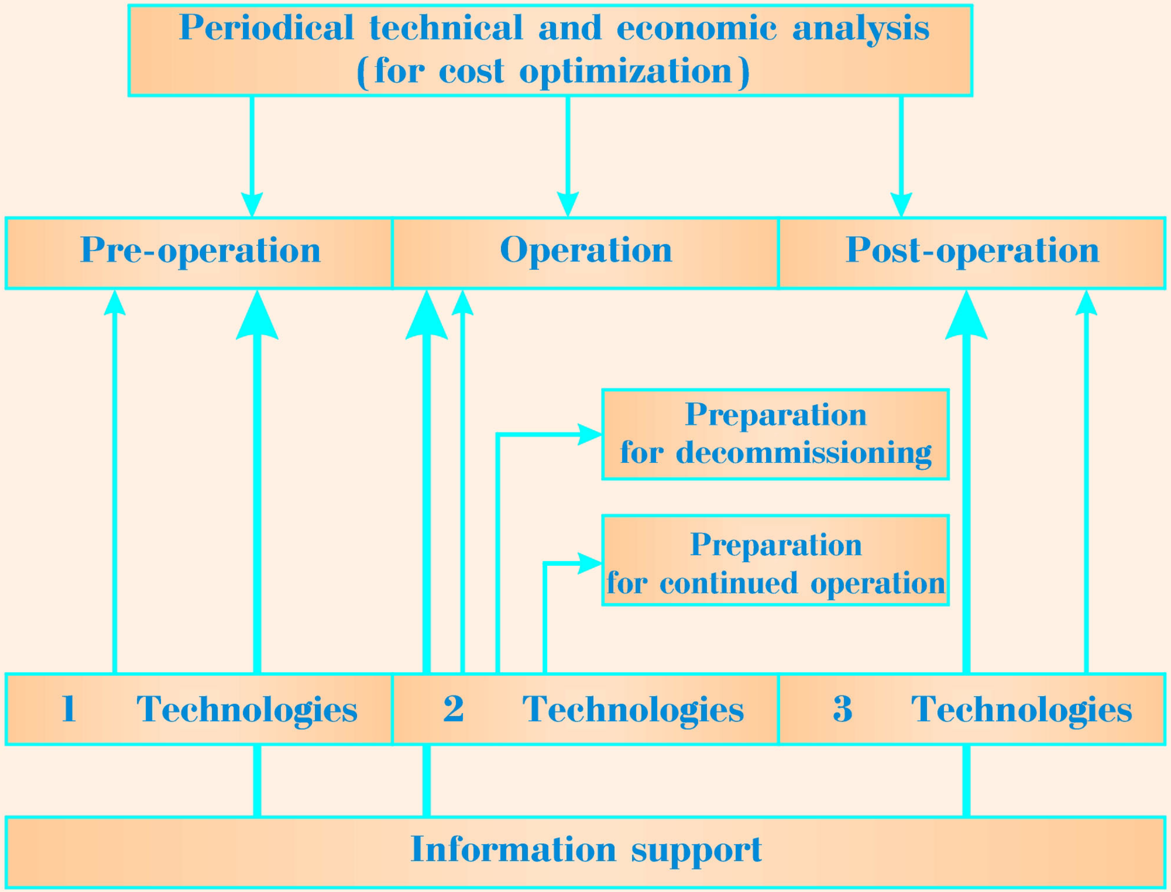
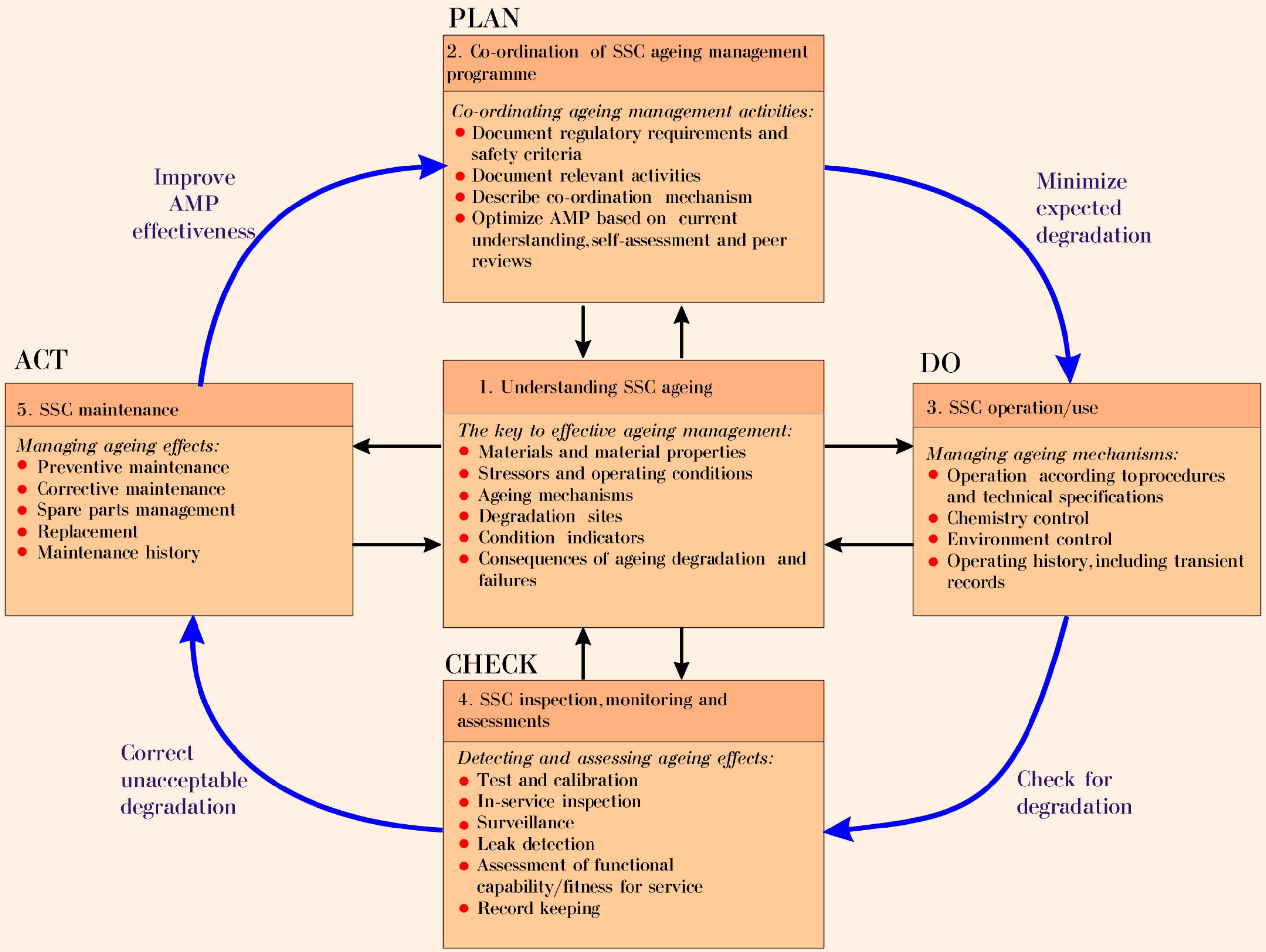


Fig.3. Management of plant equipment service life and life-limiting characteristics



Panel Session

Topic: Role of International Organizations in the Activities on NPP Life Management

P. E. JUHN
IAEA

The issue of PLIM became of interest about one and a half decades ago when a sufficient operating experience in nuclear power was acquired. At that time it became evident that a NPP during its operation needs a special programme, which would help in planning long-term safe and reliable operation. Later on as the nuclear power inventory was getting older the question of economics came to the scene to reveal the necessity to remain competitive at the electricity production market.

Economic analysis of NPP operation then showed that the activities to assure the optimised NPP life cycle should be given an integrated approach and be considered as a Programme for the whole NPP Life Cycle.

After a certain period of studies it became clear that the role of the IAEA in activities on PLIM was permanently growing because of its unique nature of the organization capable for the collection and exchange of wide range information in the area of nuclear power.

Activities of the IAEA Technical Working Group on Life Management of NPPs traditionally were aimed at the studying the ageing phenomena, however with the time its activity expanded to O&M programmes, economic aspects PLIM and decommissioning of NPPs.

The TWG with the time became an important international tool for the information exchange and technology transfer in the area of PLIM.

A variety of technical as well as major meetings in the subject area were organized during the last decade providing international forums for information exchange on different topics of PLIM programmes.

Cooperation with other international organizations (OECD/NEA) allowed for elaboration of the approach to defining the major terms of PLIM and ageing management.

The IAEA provides also a wide opportunity for the technology transfer to developing countries just embarking the nuclear power option as well as to those from former Soviet Union. Technical Cooperation projects are the means for the transfer helping the recipient countries to develop their infrastructures to support own nuclear power programmes.

It is self understood that all this assistance is arranged on the basis of highest international expertise, which is made available to the recipient countries through the Agency. Procurement of equipment, international workshops, training courses, direct expert services-this is a short list of the Agency's activities in information exchange and technology transfer.

Another important area of activities is the support of decision makers and preservation of knowledge in nuclear industry.

The most important tools in implementation of this activity are the development of relevant databases and collection of different data emerging from the operational experience worldwide.

From 1992 the IAEA is involved in the development of the International database on NPP Life management. Four software packages for different modules of this database were developed, one module is active and is used worldwide for the purposes of RPV integrity assessment, and three others are in process of data collection.

It should be mentioned that sometimes the data is either of proprietary or commercial nature and therefore not very easy available however the Agency has managed to protect the data from unauthorized access and effectively provides the Database members with the available data.

The future of the training activities at the international level may be seen in further support of so called “training centres of excellence” which could provide the opportunity for training and qualifications of different NPP personnel using the international experts in relevant areas.

International effort in achieving the goals of developing countries with the time will receive increasing attention since nowadays the IAEA Member States use more and more wide the guidance and standards developed within the IAEA and therefore begin to be more and more attached to the polices and strategies in PLIM Programme implementation at the internationally “unified” level.

The role of the Agency thus is to continue, in collaboration with other international and regional organizations dealing with nuclear power, pursuing the integrated approach to NPP life cycle management and optimisation.

Session Summaries

Session 2
PLANT LIFE MANAGEMENT (PLIM)

SUMMARY OF SESSION 2.1

Co-Chairpersons: R. Tapping / J.P. Hutin

The five papers presented in Session 2.1 developed several common themes:

- Integration is the key to PLiM
- Dababases are an essential tool for PLiM
- Documentation is critical

Integration: integration of all aspects of NPP planning, design, manufacture, operation and decommissioning, including license renewal and life extension activities, is necessary to build and operate an NPP both safely and cost-effectively. Such integration must encompass the utilities, vendors and associated resources responsible for the necessary knowledge base required to operate the NPP, including design/build and decommissioning, over several generations of staff while generating sufficient revenues. to cover these „ cradle to grave" costs. Excellence in operation and maintenance, with a PLiM focus, is required to be cost-effective in a deregulated electricity market, as has been demonstrated by those NPPs now operating at very high capacity factors yet with baseload power prices competitive with any source of electricity. Commitments from both staff and management are an essential first step.

Database: PLiM cannot be effective without comprehensive databases that provide operators, regulators and support staff with the information necessary to ensure that an integrated PLiM program is focussed an the correct issues. The databases which must themeselves be appropriatelly integrated, must cover operating data (especially reliability data), cost cause data, transienst information and, in particular, should be developed using probabilistic and risk-based assessments required to optimize operations such as maintenance. Such systematic data collection provides the rationale for license several and life extension arguments, both safety-related and economic, and, importantly, provides a knowledge base for future staff needs. These data, combined with diagnostic analysis capability, allow for correlations between operating experience and accumulated knowledge of aging impacts/mechanisms relevant to the NPP or system of concern. Such data would include external databases of operating experience as well as accumulated R&D results. An important product of this activity would be the ability to compare design specifications, for instance for fatigue, with the actual transient data. It may be inappropiate to use design codes for PLiM, for instance.

Documentation: it is critical to document „ the rules of the game" for PLiM, for the NPPs, the regulators and others involved in the political and economic decision-making for NPP operation and life extension. As noted earlier, life management, or aging management, includes people as well as plants, but must be accompanied with comprehensive documentation of the requirements, process and applications of all aspects of PLiM. These documents provide for consistent interpretation and regulation, as well as a knowledge source. As with all documentation, these would be living documents subject to periodic review and revision.

SUMMARY OF SESSION 2.2

Co-Chairpersons: K. Miya / K. Simola

In this session, four papers were presented. The first paper addressed the management of plant specific information and uncertainties in ageing analyses. Good, integrated and updated data collection systems are essential in efficient life management. As an example, a piping information databased under development for Olkiluoto NPP was introduced. The role of uncertainty analyses was discussed, and it was pointed out, that their importance grows when probabilistic analyses are considered instead of over-conservative deterministic analyses.

The second paper (Nopper et al.) summarised a PLIM approach developed at Framatome ANP. The concept includes tools and strategies for ageing management of all plant equipment. A selection process for components takes into account both availability and safety aspects. Depending on the possibilities to quantify the degradation, lifetime prediction is done either deterministically, with probabilistic methods, by monitoring methods, or by combining these.

Third paper (Tóth) described the regulatory programme for renewal of the operating licence for Paks. The preparatory activities of the safety authority, HAEA, included e.g. studying U.S. and IAEA documents related to ageing and preparing national ageing management guides. For license renewal - and even in the frame of current operating licence - HAEA as determined four areas that have to be properly managed both by the utility and the authority. One of these is ageing management programme.

The last paper described the Russian Minatom's R&D programme to support PLIM. The starting point is that PLIM activities can be divided in two constituents: a plant specific paert and a universal one. The development activities are focud on the more universal part, also called supporting basis. Normative and methodological documents, algorithms and databases are developed to support life management decision making.

All the papers clearly reflected the complexity of life management, but at the same time they showed that there are systematic approaches to cope with it by structurising the life management problem. Life management is a set of multi-criteria decision making situations, and for good decision making we need good practices and procedures, advanced tools, integrated data collection and management systems, and approved analysis methods. E.g. the development of software to support life management is an active field. All the efforts help to make the the ageing management processes more transparent and traceable.
It might be worth to further explore the current activities to identify good practices, taking into account economical, methodological etc. constraints.

SUMMARY OF SESSION 2.3

Co-Chairpersons: R. Caro / F. Gillemot

The subsession provided a very interesting overview on the PLIM activity on different type and differently aged NPP-s.

The first paper was presented by Mr. S. Vodenicharov. It introduced the situation in Bulgaria. Four WWER-440/230 type units are operated at the Kozloduy site. Unit-1 and unit-2 will be shut down recently, and extensive modernisation are going on to reach the required safety levels and extends the life time to reach at least the design life.

Mr. M. Brumovsky presented a paper on the PLIM activity at NPP Dukovany. At Dukovany WWER-440/V213 units (belonging to the second generation of the WWER-s) are operated. The units are only 15-20 years old and are in good shape. Intensive PLIM program have been implemented to operate the plant beyond the design life.

Mr. Brumovsky also presented one paper on the PLIM activities at Temelin site. At Temelin WWER-1000 units are operated. The plant is new, but PLIM already started.

Mr. S. Azeez introduced the PLIM system recently elaborated for CANDU reactors. This PLIM system would assist the CANDU units to provide electricity beyond the design lifetime.

Mr. D. Vidican shown examples of the practical use of the new CANDU PLIM system. It is applied on the Cernavoda unit 1, which one is a new CANDU reactor.

The conclusions of the subsession:

- PLIM should be started as early as it possible, preferably during the construction or at the start of the unit.
- Different type and differently aged units are having different technical problems, but they use the same PLIM methodology to extend the lifetime. This underlay the necessity of further international exchange of information and of further co-operations.
- Reactors can be operated until they fulfil the safety requirements. These requirements are not harmonised among the different countries. In some cases plants are shut down not for economical or safety reasons, but due to political decisions. Further activity is suggested to harmonise the safety requirement (especially outside of USA) and to inform the governments and the public on the existing situations of the operating plants.

SUMMARY OF SESSION 2.4

Co-Chairpersons: J. Stejskal / L. Debarberis

For the Plant Life Management of Nuclear Power Plants many factors are of importance. In this session the following contributions have been presented and discussed:

- In the first presentation Mr. Bedrose pointed out the importance of the quality and qualification of manpower and training for the plants management. This is the case not only for operating plant but also for new NPP Projects. A good basic education and additional specialized training of manpower throughout the project is essential for future safe operation and PLIM of the plant.
- In the second presentation Mr. Bhatnagar explained how the experience gained in research reactors could be feedback to NPP PLIM programmes. One of the most important elements of aging management is to provide condition-monitoring information on the present state of components and assure NPP safe/competitive operation during the reactor life.
- In the last presentation Mr. Rucareanu showed the very important role of establishing an aging management program for plant life management. This program should provide satisfactory evidence that for all safety relevant systems and components (SSCs) all known aging mechanisms have been taken into account in planning maintenance activities and quality assurance and also ensure that measures will be taken if any omissions are revealed. Aging management also provides a sound technical basis for optimizing maintenance programs and ensuring the reliability of components. Early PLIM programme is promoted and criteria suggested based on safety as well as availability criteria.

- <u>Conclusion</u>:
    - PLIM diverse aspects have been treated in this sub-session, well established plant life management must consider a large range of contributions relevant for a long, safe and competitive operational life
    - Clear understanding of relevant/critical components and their ageing stressors kinetics is of fundamental importance for PLIM. Possible guidelines for components ranking are welcome (IAEA).
    - Monitoring tools/diagnostic and staff competence and training are essential for PLIM.
    - Early PLIM involving safety as well as availability criteria is required.
    - It has to be demonstrated that at any point of plant operation the safety margins are not affected.

Session Summaries

Session 3

SPECIAL ASPECT OF PLIM: ECONOMICS AND END OF LIFE CYCLE

SUMMARY OF SESSION 3.1

Co-Chairpersons: J. R. DeMella / B. Gabaraev

Session 3.1, in focusing on “special aspects” of nuclear power plant life management, in particular, dealt with the “End of Life” of the PLIM cycle as well as various perspectives of assessing and evaluating the economics of nuclear plants throughout the PLIM cycle.

The three papers presented specifically addressed the identification of a number of issues and mitigating strategies associated with companies’ decisions to either decommission or extend the life of certain nuclear power plants and in the case of the paper on nuclear economics, various approaches for assessing the economics of nuclear power plants throughout the PLIM cycle were presented with particular emphasis on the economics of nuclear plants operating in both regulated and unregulated electric markets.

Mr. Jean-Jacques Grenouillet of Electricite de France (EDF) presented the case supporting EDF’s decision to decommission, employing the Safe Store/Final Dismantlement approach, nine first generation French nuclear units which had been previously shut down at various times since the late 1980’s. In particular, Mr. Grenouillet emphasized that the basis for EDF’s decisions and subsequent strategies to decommission were founded upon improved economics, favorable public opinion and efficient ways to transition EDF resources to replace these units with a new generation of more efficient and safer reactor designs.

Mr. Gary Young of Entergy Nuclear USA presented information supporting the decisions for license renewal by several US utilities. Based on Mr. Young’s presentation, there is a preponderance of opinion by US nuclear utilities, regulators and nuclear advocate organizations that extending the life of US nuclear units is certainly justified based on the merits of favorable economics, prospective demand growth and is consistent with US public opinion which according to Mr. Young, supports the sustenance of nuclear power as a key energy option within the US.

Mr. DeMella, in addressing the economics of nuclear power throughout the PLIM cycle, emphasized the need to distinguish between approaches used to evaluate the economics of nuclear plants which operate in regulated Vs unregulated, competitive electric markets and addressed a number of measures of economic value typically used in both. In particular, Mr. DeMella pointed out that from a financial/economic perspective, regulatory situations for nuclear plants were substantially different throughout the world. As a representative example, Mr. DeMella briefly presented and discussed a nuclear plant economic valuation study which demonstrated potential economic risk and opportunity for operating nuclear plants in competitive electric markets.

By way of questions and comments from participants, it was certainly confirmed that political, social and economic drivers underlying the economic evaluation and justification of nuclear power varied markedly throughout the world. In addition through participants comments and subsequent discussion it was emphasized that with regard to the end of life PLIM cycle, there was a need to distinguish between the process of “License Renewal” verses that of “Plant Life Extension”. It may be of interest as a subject for future papers and presentations to contrast and compare in greater detail how two countries (France Vs. USA) in applying PLIM concepts to end of life nuclear plants arrive at dramatically different conclusions (Decommissioning vs. Life Extension), justified for apparently similar reasons.

SUMMARY OF SESSION 3.2

Co-Chairpersons: $\boxed{\text{S. Bugaenko}}$ / S. Goates

At the present time practically all NPP-s are commercial enterprises the main goal of which is to develop financial profit for their owners.

I recent years several countries have deregulate their electricity markets thereby starting the route to competitive supply and pricing. Deregulation of electricity markets is a trend that is expected to be followed by many countries. This process has an influence on the performance of the existing NPP-s as well as\ on the structure and quality of new NPP-s and the nuclear power industry as a whole.

The ability of new NPP-s to compete essentially depends upon their capital costs and associated research being carried out to identify the various means for reducing capital costs of NPP-s.

It is known that time and operation leads to decline in the useful properties of the NPP as a whole due to component materials ageing, degradation and obsolescence. In turn this causes a detoriation in the profitability and safety levels. Therefore there is a need to continue with research aimed at mitigation of such impacts.

All of these aspects are highlighted within the session 3.2. presentations. Dr. Grusha in his presentation "Prospects of Nuclear Power development in Belarus" considers ways of archiving economic optimisations of electric energy development in Belarus. The authors reach the conclusion that the only way is through nuclear power.

The presentation "Power up rating of NPP-s – Solutions, Experience, and Economical Aspects" focuses on the analysis of options for increasing electrical power output from the existing NPP-s. It explores options for gaining power increases by core developments and by improving the efficiency of secondary systems and components. In most cases the combination of both approaches leads to maximum power output, although a careful analysis of the economics is needed.

In the presentation "BN-350 NPP Regulatory Aspects of Decommissioning" Mr. Shiganakov provides information on the development of regulatory documents needed for carrying out BN-350 decommissioning work in accordance with the laws and regulations applicable in the Republic of Kazakhstan.

The presentations "Economic analysis of NPP for decision making in Thailand" describes the main results of comparative economic analysis of options for increasing electricity production in Thailand by use of nuclear, oil or coal fuel. The author reaches the conclusion that nuclear power plant will offer the lowest interest rate return, but recognises that there are other factors including price stability, environmental benefits, energy independence and improvements in the domestic economy that make the nuclear options attractive.

Prof. Kuznetsov in his presentation considers the use of an unconventional approach to the extension of service life of aged units through the use of combined cycle nuclear-gas power plant. The proposed technology is examined for Russian WWER-440 reactor and also as a pilot project for the Russian VK-50 reactor. The authors believe that there are significant economic benefits from this approach.

Overall the sub sessions shows that the differences within individual countries with respect to the social, economic climate promotes a variety of creative solutions on how best to address these external market influences. Within the context of Plant Life Management this results in various ideas and considerations as reflected within the presentations. In particular this highlights that the use of modern technology or application of old technology in a new way can provide the required economic improvements.

Commentary:

The presentations demonstrate how the Nuclear Industry is applying creativity to provide solutions to influence external factors. How is the industry applying this creativity to influence the political decision makers and the public?

Session Summaries

Session 4

TECHNOLOGICAL AND OPERATIONAL ASPECTS OF PLIM

SUMMARY OF SESSION 4.1

Co-Chairpersons: M. Ishikawa / J. Figueras

Two of presented papers are related to specific PLIM works, one from Finland and other from Canada.

The Finland presentation refer to the Loviisa NPP (WWER 440, 2 units) PLIM technological aspects and is based on a mixture of operation & maintenance plans, replacement strategies, SSC technical modifications and ageing effects mitigation.

The Canadian paper show the preparatory work undergone for a big refurbishment in Gentilly 2 NPP (PHWR, CANDU-6 model) based on a PLIM program for a few critical SSC and a Condition Assessment analysis for other SSC of the plant.

Third paper was a very specific, detailed and technically sound presentation of on-going R+D performed by EdF in France in relation with the issue of RPV pressurized thermal shock in PWR from three aspects: thermo hydraulics, materials and structural integrity assessment.

Last paper was presented by Russian Federation and contain a ^think-tank^ of ideas to develop at national level, NPP level and at SSC level for a better and effective way to implement PLIM methodologies in this country.

SUMMARY OF SESSION 4.2

Co-Chairpersons: G. Young / S. Cepcek

Session 4 and sub session 4.2 were devoted to the technology and operational aspects of PLIM. Three papers were presented which provided research results that improve the understanding of material ageing effects due to both transients and normal plant operation.

Mr. A. Martin from EdF in France presented the results of research and development efforts aimed at the investigation of behaviour of fluid flow in the down-comer of the RPV during pressurized thermal shock (PTS) transients. These studies were done using the thermal hydraulic finite element codes N3S and SYRTHES. The results of the analysis were presented including animated thermal fluid flow results that provided excellent visual representations of the phenomenon. This research provided more accurate fluid temperature distribution and heat transfer results that will allow for more accurate fracture mechanics calculations for the PTS transients.

Mr. M. Tsukada from JAERI in Japan presented the results of research on degradation o RPV materials due to irradiation. The areas investigated included:

- Fracture toughness evaluation using the Master curve approach
- Grain boundary embrittlement caused by phosphorus segregation
- Gamma-ray effect on irradiation hardening
- Irradiation assisted stress corrosion cracking (IASCC)

Although this work contributed to better understand these ageing related effects, more research is needed to better understand these ageing related material behaviours.

Mr. G. Dundulis From L.E.I in Lithuania presented the results of investigation of IGSCC in the Ignalina nuclear power plant austenitic piping welded joints. The results show that the primary reason for cracking is overheating during the pipe welding process. Corrective measures including new welding technology and water chemistry improvements have been introduced to reduce the potential for future cracking. A risk-based ISI pilot study aimed at optimisation of the ISI program has been completed. At present the IAEA extrabudgetory Program aimed at mitigation of IGSCC is underway.

In summary the results of research and investigation presented in these papers contributed to development of more accurate assessing and calculating procedures, which are needed for better PLIM of nuclear power plants.

SUMMARY OF SESSION 4.3

Co-Chairpersons: I.S. Yeong / M. Hoda

The presentations in Sub-Session 4.3 are focused on solving ageing issues in nuclear components by the way of localisation of technology and national approach like JAPIEC.
Mr. Miller presented their experience in manufacturing aged in-core instruments with domestic R&D efforts.

This could be a good example of PLIM strategy in terms of technology localisation.
Mr. Tapping presented how he had performed ageing assessment of CANDU 6 steam generator, which usually had have good service record with little or no significant active degradation.

He emphasized in two cases the role of R&D on degradation mechanism and the role of continuous feedback of experience from operations.

This kind of approach will be well utilised when we are confronted to a case of life evaluation for good service components.

Mr Koyama from JAPIEC presented very good approach in ageing and degradation mechanism R&D that is currently performed by JAPIEC and PLEC by the financial support of Japanese government.

When we consider general tendency of NPP utilities for developing technology negative, rather positive in PLIM activity.

Good co-operation of government and industry in Japan would be highly praised in the worldwide ageing and PLIM society in the future.

Dr. Hwang for Korea Atomic Energy Institute presented his efforts to verify the service integrity of the local made steam generator tubing material. It is a good approach to assure the operating safety and to predict the ageing condition and environment in terms of life management of steam generator.

In order to perform PLIM, condition assessment and ageing management strategy applicable to the aged plant are so important that all Member States should concentrate on the phenomena of ageing and degradation mechanisms.

Especially technology information exchange and cooperation are encouraged for enhancement NPP safety and preventing component failure incident.

SUMMARY OF SESSION 4.4

Co-Chairpersons: C. Belinco / C. Bell

The activities of the Nuclear Energy Plant Optimization Program (NEPO)are described in the paper "NEPO Cable system ageing management programs" presented by G. Toman. In the program various techniques have been developed for modelling and monitoring the aging cable polymers used in nuclear plant cable systems. Advanced techniques are to be applied e.g. identify cables that have been aged significantly, and training aids based on these techniques have been proposed.. The ageing fragility point has been determined in the program for composite EPR/CSPE insulation with respect to LOCA function.

The suitability of an Arrhenius type model for predicting aging is being studied using data from highly precise oxygen consumption experiments. The linearity of the model at low temperatures has been evaluated. The results show that ageing data can be obtained even for room temperature experiments.

The wear-out technique, allowing highly non-linear ageing behaviour to be made linear, has been developed to achieve a linear indication of the remaining service time for cable polymers. In this technique, the wear-out points of naturally aged and reference polymers are determined using high-rate aging. The resulting ratio of ageing periods is used for estimating that remaining service life. A condition monitoring technique is applied to establish the end points.

Also, techniques suitable to evaluating aging of large number of cables, needed for planning and scheduling cable replacements, have been studied in the programme. For example, a new technique based in magnetic resonance testing has been developed.

The role of reliability in the standby safety systems of nuclear power plants and the demonstration of it is discussed in the paper "The advantages of reliability centred maintenance for standby safety systems" presented by R. Dam. Due to the extensive testing programs required for demonstrating statistical reliability and, on the other hand, because these programs often provides scrutiny on standby safety systems, the usability of a maintenance optimisation strategy, such as reliability centered maintenance (RCM), is questioned.

The role of reliability is considered as part of a system maintenance program, as a level of redundancy, driven by reliability requirements and, on the other hand, for ensuring that the system operates reliable. It is emphasised that RCM provides a broader context where testing is only one part of an overall strategy focused on ensuring that component function is maintained. Each strategy is targeted to identify known component degradation mechanisms.

A maintenance program driven by reliability requirements can thus be expected to include testing defined at a frequency intended to support the needed statistics. The testing, needed for demonstrating that the desired function is available, should in this strategy be regarded as part of an effort to ensure that the desired function will be available also in the future.

The paper considers the application of a streamlined form of RCM the emergency core cooling and standby generator systems of a CANDU plant and the studies performed on standby safety systems.

Use of the RCM analysis to demonstrate its usability in the optimisation of the testing program dictated by reliability as well as application of condition monitoring and predictive maintenance techniques in testing are discussed. The importance of identifying and linking that different plant activities within a well integrated plant culture is shown.

The background and implementation of the reactor protection system refurbishment project accomplished at the Paks NPP are described in the paper "Experience with safety I&C modernisation in Paks NPP" presented by T.Turi and M Klein. In this project the Russian reactor protection system was replaced by a western digital system based on Siemens Teleperm XS technology. The systems and their installation and commissioning, done during 1999 and 2002 for the 4 units, experiences during the project and features specific to the systems of the Paks plant are described. Also the importance of people training was pointed.

The background and basis of the project launched by EPRI on NPPP electronic component and board ageing management are explained in the paper "Instrumentation and control ageing management: a focus on electronic parts and boards" presented by C. Crocombette. The objectives are to reduce the likelihood of losing plant availability due to such failures and to reduce the global cost of maintenance.

By analysing the existing data it was found that the devices more sensitive to ageing were electronic capacitors, relays, potentiometers, various power devices, optocouplers, onboard connectors, printed boards and integrated circuits. As part of the study, field failure data have been collected from several NPPs in France and the U.S. Potential root causes and degradation mechanisms are analysed.

The monitoring of electronic systems, based either on ageing tests to assess the residual lives or continuous monitoring of key indicators, are considered as part of ageing management. Previous (IAEA) studies on the management of ageing show that work is still needed for some components. Besides gathering existing methods, new methods are been developed and assessed in the project.

Guidelines for the ageing management of I&C electronic boards have been developed. Several actions of aging management, like those helping to diagnose the state of aging of systems, are covered. These include failure and destructive analysis, aging tests, continuous monitoring and visual inspections. Actions to be undertaken after these diagnoses, like preventive replacements of parts or refurbishment of boards and, on the other hand, measures enhancing their residual lives, are also included.

As a conclusion, it is recommended that even under favourable circumstances the ageing mechanisms of I&C electronic systems, especially parts sensitive to ageing, are controlled using ageing tests or continuous monitoring of end-of-life indicators, or both.

Finally we can point out that this sub-session makes a contribution, with various techniques, methods and concepts, in the technological aspects of PLIM, and also is a demonstration that we must fulfil requirements in SSCs and human fields.

SUMMARY OF SESSION 4.5

Co-Chairpersons: G. Bezdikian / P. Vidican

The structure of sub-session has 3 main parts:

- Vessel Head penetrations; French approach for maintenance,
- Development of inspection and evaluation guidelines for LWR,
- Human Resources conditions for life extension.

It emphasize ;

- How a defect in a critical component should be evaluated an experience applied to other similar. It was demonstrated the effectiveness of a good management on a ageing problem, like defect detection – root causes analyses – perform research and expertises – technical / economical decision, in a fast moving company.

  In evaluations of mitigation activities necessary, it is important to apply As Reasonable Achievable Concept.

- The necessity to develop and to improve sound inspection and evaluation guidelines for nuclear reactors is one of major part for most critical components having a great influence on the Plant Life.

  Also the necessity to prepare in advance contingency plans, to be used for the case of appearance of similar cases as in other plants.

- The general trend is to arrive at 60 years for NPP life. This need a huge amount of activities, like;

  - Research and Development,
  - Issue solutions,
  - Validate procedures,
  - Perform application,

  and at the end, to assure margins for uncertainties.

- Assurance of young good trained people for nuclear industry is another important part of life extension. This means: Training, Screening of people, creating an adequate environment for their work, to remain in nuclear field. Maybe to create a human resource database for each country policy will be a good way.

  Assurance of good trained people could represent a serious challenge in a deregulated job market.

- Nuclear energy and PLIM, imply management in an area of high-level technology, which need very good trained people.

  The actual costs of training should be recovered in the future by attractive access of representative countries to above area.

To cover the increasing demand of energy there are not other alternative solution. The solution is to have public acceptance, high-level condition of operation via major programme of maintenance and operation and make attractive the nuclear technology for young people.

Session Summaries

Session 5

CURRENT NATIONAL APPROACHES TO PLIM

SUMMARY OF SESSION 5.1

Co-Chairpersons: I. S. Hwang / P. Tipping

Paper 70: B. Gueorguiev; **Integrated NPP Life Cycle Management – Agency's Approach**

This paper indicated the importance of nuclear power, but pointed out that practically no new nuclear capacity is coming onto line. From the 438 NPPs in operation, 230 have already reached 15 years of production. Forecasts show that there will be considerable need for qualified personnel to assure the safe use of nuclear power under cost-effective conditions. Plant life management requires management from the design to the decommisioning of a NPP. Optimisation of NPP operation may be likened to providing additional generating capacity whilst maintaining safety. Accordingly, an integrated approach to NPP lfe management is becoming increasingly important. The IAEA is accordingly implementing several programmes, including the project on engineering and management practices for optimisation on NPP service life, including final decommissioning. The project scope includes different possible modes of IAEA operations for information exchange and technology transfer. Issues under consideration include economic analysis and considerations related to decision on continued operation or decommissioning, optimisation of operation, maintenance, in-service inspection, ageing management and mitigation, control and instrumentation, creation and developing databases, retention of know-how, training and qualification of NPP personnel etc. The paper concluded that the IAEA is following all developments on NPP life management and implementation. The Agency is concentrating on analysing existing national approaches and experiences on policies and strategies of NPP life management as well as the information exchange on development and implementation of constituent parts of PLIM.

Paper 54: P. Manolatos/G. Van Goethem; **Euratom Community Co-Sponsored Research in NPP Life Mangement**

The paper concentrated on plant life extension management (PLEM). There are 143 NPPs in E.U. Member States, and despite the good safety record there is still needed research to further increase safety and performance. The European Commission's Framework Programmes dedicate funding to enhance safety and improve the competitiveness in the nuclear industry. The beneficiaries are the NPP designers, operators and regulatory. It was noted that a further challenge lay in achieving the same safety culture amongst E.U. and Central and Eastern European Countries, especially as E.U. expansion continues. To date some 30 projects have been financed under PLEM. These activities focussed on three key issues, namely, 1) Integrity of equipment and structures; 2) On-line monitoring and maintenance and 3) Organization and management of safety. Apart from stringent rules for operation, it is necessary to understand the way materials can degrade (corrosion, embrittlemeent etc.) With on-line monitoring and maintenance it is essential to prove the reliability and capabilities of innovative inspection techniques. This is done using Round Robins. Applications are focussed on irradiation damage in RPVs, thermal fatigue in piping, corrosion of internals and radiation field monitoring. In particular, weld repair procedures to minimise internal stresses and shorten repair times are issues. Further, risk-informed in-service inspection is becoming more important. The implementation of digital instrumentation and control tools and necessary training of staff is an important challenge. It was recognized that organisational factors are often at the root cause of incidents or accidents. Accordingly, methods for managing change, obtaining tools for supporting decision on inspection and management, safety performance indicators review and evaluation are all associated aspects.

Other projects include embrittlement, corrosion, fracture mechanics, concrete, welds, thermal hydraulics and harmonozation of practices in E.U. and Eastern Europe, for example.

Paper 55: J.M. Grandemange; **Safe Ageing Management of Nuclear Power Plants – A European Synthesis**

The paper distinguished between 1) changes in materials and 2) changes in personnel skill and procedure adequacy as operating time is accrued in a NPP. It was recognized that with time both aspects may not be compatible with required saftey provisions or with respect to economic issues. Either repair or replacement or changes in operational procedures (conditions) can help to mitigate such ageing effects. The paper went on to summarise the results of a E.C. supported studies concerning 1) the synthesis of work done under international auspices, and in the European context, the comparison of ageing management approaches ised in several European countries, with international recommendations, the summary of various potential phenomena and their governing parameters, the methods of in-service ageing identification and possible mitigation methods and illustrative ageing management practices, taking material ageing aspects as examples. It was noted that there were 56 OECD and IAEA reports on ageing management issues. This reflects the importance of the subject. Again it was stressed that safety is the liability of the operator and this was under surveillance from the regulator. Practical ageing management methods were identified as 1) Periodic safety reviews (PSRs), 2) Implementation of lifetime management programmes for non-replaceable components and 3) Generic evaluations on main ageing phenomena. The IAEA AMAT Guideline was used to compare ageing management approaches. A comprehensive Table was given, showing the most important ageing mechanisms and where they appear. Not only metals, concrete and electronics age, but also polymers. All require monitoring according to their importance to safety and function. The issues of RPV embrittlement, thermal ageing, creep, fatigue (high, low cycle and thermal, corrosion (all types), wear, environmental effects etc. are likely to remain important to ageing of NPPs.

Paper 9: L. Debarberis, et. al.; An Integrated View on Plant Life Management; EC-JRC-IE Projects and Programmes

This paper gave an overview of the current E.C. programmes and JRC-IE activities with respect to the issues of NPP plant life management. The key results so far were presented and discussed. The plant life management integrated JRC approach covers the whole spectrum of necessary activities. The assessment of integrity, residual stress evolution, corrosion, practices,material properties evolution, and the finding of defects all have their individual programmes and give the complete approach to all aspects of NPP life management. Results achieved to date include knowledge gained about irradiation embrittlement, material degradation assessment and monitoring, surveillance data analysis, structural integrity and fracture mechanics and qualification of in-service inspection methodologies etc.

The SAFELIFE 6th Framework Programme JRC Project was presented.

SUMMARY OF SESSION 5.2

Co-Chairpersons: G. Sgarz / S. Vodenicharov

For the 8 NPPs of Czech an integrated requirement of ageing and plant life management is given by the legislative and regulatory body for Nuclear Safety. The independent central state administration body need to be involved in all safety relevant activities, as are life time assessment or modifications or modernisation programs.

For the fleet of 58 PWRs the French utility EDF had started a lifetime Management program. For one of the major components the RPV EDF is going to evaluate the margins of different important parameters such as toughness or the entirety of integrity assessment in all plant conditions. The maintenance strategy and the global results of integrity evaluation assure a lifetime of the RPVs of at least 40 years.

Due to the German safety regulation requirements ageing management activities have been performed since the beginning of plant operation. Although no specific ageing management program exists, so far, the German utilities understand that the relevant ageing mechanisms are technically and administratively covered by the entirety of measures already taken.

An approach towards a more systematic framework concerning the direct link between safety related ageing management requirements and the related remedial actions will be appreciated and promoted by the German Reactor Safety Commission (RSK). The German utilities intend to demonstrate how and where the AM-actions have been carried out in terms of technical and administrative issues based on an overall ageing Management concept. The harmonisation process inside Germany will be finished till the next months.

In Hungary's domestic generation the nuclear operation plays an important role, and it will do it also in the future of the next two decades. A feasibility study showed that a lifetime extension of 20 years has a realistic perspective. Only measures like heating up of some vessels are needed in order to decrease thermal stress levels caused by transients.

Additional replacement of the main-condenser and some other components sound very cost-effective to reach the extended life of 50 years of operation. The business analysis shows that the plant lifetime extension is a reasonable business decision and prospects electric power generation at low market price conditions. The plant lifetime extension project shall be completed in 2007.

In the earlier Pressurized Heavy Water Reactors of India zircalloy-2 has been used as coolant tube material. Studies and experience showed an earlier limitation of lifetime than originally estimated. This meant that several reactors have to fulfil a huge program for the replacement of coolant channels. During that long lasting work of 18 months the time is used to life extension and upgrading works.

To meet latest safety requirements an extensive program of back fitting was introduced. This gives a great challenge for the utility and regulatory authority especially in the areas of configuration control, documentation, training and re-qualification of operations staff.

In Japan after more than 30 years of operation in some power plants the issues of ageing management became more and more of interest. So the utility was forced by the Nuclear Safety Commission to implement effective counter-measures against ageing. The authority determined to incorporate the technical evaluations of ageing and the preparation of long-term maintenance plans into the periodical safety review in the future. In the Report of the Nuclear and Industry Safety Agency the related activities on current approaches to cope with ageing are outlined.

After more than 20 years of operation with the Korean's plant Kori 1 a plant life time management program comprehensively assessed the current physical condition of the plant. Subsequent program of PLIM had focused on evaluating in detail integrity of major structures and components.

Korean utility deals physical ageing with the plant lifetime management and more physical ageing with the periodic safety review.

The Korean utility are going to screen systems, structures and components to perform engineering evaluations to do system performance monitoring and to perform field tests to support life evaluation. A well-harmonized PLIM program ensures to keep plant safety at the level of regulatory requirements and help to increase industrial competitiveness of nuclear power production.

SUMMARY OF SESSION 5.3

Co-Chairpersons: J. C. Hennart / M. P. Sharma

In this session 5.3, we have heard many different approaches to Plant Life Management. These approaches, methodologies, programmes, or whatever they may be called have differences and similarities

Among the differences we have noted:

- The Regulatory context
  - In some countries the licence renewal approach is applicable, in other countries the periodic safety assessment approach prevails.
  - In some countries the requirements are general guidelines, in other countries more detailed rules are recommended.
- The key distributors to the PLIM programmes, in other words the actors who are playing on the stage of PLIM also differ from county to country. They are:
  - The Regulatory Body
  - Utilities and Utility groups
  - Reactor Suppliers
  - Research Institutes
  - International Organization (IAEA)
  - or combination thereof.

Among the similarities we have noted:

- The PLIM approaches almost always include the same steps:
  - Prioritisation
  - Analyses and reviews
  - Definition and implementation of actions, etc.

  In other words they tend to stick to the "PLAN/DO/CHECK and ACT" of the IAEA recommendations.
- They rely on the same resources, like
  - Organization
  - Operation feedback
  - Data collection
  - Human resources
- They are subject to the same type of control and review:
  - Organization
  - Quality of data
  - Ageing management review, Documentation, etc.

What emerges from the presentations of this sub session is that in all countries the PLIM programmes:

- have been implemented in due time
- they have contributed to ease the licensing process, whatever it is.
- they are based on criteria.

The most frequently mentioned criteria are:

- the safety criteria
- the environmental criteria
- the economical criteria.

Safety on environmental criteria is generally fully addressed in the papers. The economical criteria are still far less developed.

If I may add a personal comment, I would express the wish that in future papers more emphasis be put in describing the ways economics is taken into consideration at the various stages of the PLIM programmes.

I am convinced that economics plays an even important role as safety and that economics and safety are complementary not competitors.

Session Summaries

POSTER SESSION

Summary of Poster Session

M. Brumovsky

Poster papers cover most of the important parts of Plant Life Management/Extension, specific to individual countries as well as individual type of reactors even though VVER type reactors prevails.

Research and development programmes are still an important part for integrity/lifetime evaluation of main structural components (P6, P16) especially as a support for reactor pressure vessel emrittlement and integrity evaluation. Specific programmes dealing with cladding behaviour and neutron fluence/ attenuation effects are now running for VVER plants but with a general application to PWR type vessels.

Dosimetry is still a non-negligible part of reactor pressure vessel lifetime evaluation and improved methods are applied for BWR as well as VVER type reactors (P3, P9).

Integrity problems and assessment of main structural components need new inspection (P14) procedures, like risk-informed in-service inspection as well as new, updated evaluation method for Russian VVER reactor pressure vessels during operation when a re-definition of postulated defects, design fracture toughness curve and integrity evaluation procedure have been approved (P12).

Corrosion problems are still of high importance especially for ageing NPPs - fuel (P1) and coolant (P7) channels are subject of study and evaluation as well as Alloy 600 (P8) used for steam generator tubes or RPV nozzles.

Special efforts have been given also to specific problem of diagnostics like protection against boiling nucleation or modernization of control rooms (P15) of NPP in accordance with newest requirements.

Personnel problem is becoming more important with ageing not only of NPPs but especially with technical and scientific personnel (P21).

Finally, licensing procedure (P20) is specific for individual countries and even though it is not a part of PLIM, it has a very close connection with the whole process, as PLIM must result in re-licensing of the plant for next period.

International co-operation is a very effective tool for dissemination of experience and knowledge between countries and especially from nuclear developed to developing countries. It also could serve to a harmonization of approaches and procedures with the aim of increasing safety and effectiveness. Good examples in this field are technical-cooperation activities leading by the IAEA (P19).

Large number of electrical motors is operated at NPP-s. The preventive maintenance and life management of them reduces the malfunctions and repair costs. (P4).

Several methods exist for RPV lifetime extension. The most promising ones: study the clad properties and use real data for assessment of the aged clad behaviour, use the Master Curve, and finally to anneal the vessel. All of them require further R&D. By selection of proper RPV life mitigation methods, the RPV in the WWER-440/V213 type NPP-s will not be the life limiting element (P6).

CONCLUSION OF SYMPOSIUM/CLOSING REMARKS

SYMPOSIUM CHAIRMAN'S FINAL REPORT-L. M. Davies

In this Final report I will identify some of the major points that have emerged from this Symposium, some of which have been continuing themes throughout the meeting. To do this I have drawn on the session Chairmen's reports and the separate report on the Poster Session and I have also added some points of my own.

The venue and dates for this symposium were timely in that it was held in Hungary at the stage when the up rating and definition of the Operating Lifetime of the PAKS plant were announced approved by the Station Directors.

The terminology of Nuclear Power Plant life and its operational life is confused and still requires attention. The Plant Life Cycle describes the period which includes "pre-conception" through the stages of choices of plant type and site, design, financing, procurement, commissioning, operation, shutdown, decommissioning and the return of the site to a "greenfield". The period of "operational life" is not defined (so it can't be "extended"). There is the "design life" which is the target aimed at the plant designers and which together with other factors incidentally used for determining the notional cost of electricity generated by the plant. It is during the operational phase that the plant makes electricity for sale and that income is the source of funding all the other activities. At some stage during operation the design operating life can be reviewed and the operating life can be established-coincidentally, the current stage of the PAKS plant.

Initially the main objective in Plant Operation is Plant Life Assurance. The Plant has to meet at least its Amortization Life and performance targets – to ensure profitable operation, but as I mentioned earlier there are examples when this has not happened-and there have been some dramatic failures.

In most countries and at some stage, usually about halfway in the design operational life the plant starts being subjected to a 10-year interval Periodic Safety Review (PSR approach) – which allows for the continued operation of the plant *but the operational life is not defined*. There are exceptions to this approach. In the USA, the plant is licensed to operate for 40 years and Owners are now well embarked on a national programme of renewing this licence - covering a period of a further 20 years. This is a "Re-licensing" process.

It is clear that the harmonization of definitions in this area is necessary – to reduce the current confused vocabulary.

The importance of integrating the resources needed for plant life management and operation was stressed – this process should also encompass the efforts of the utilities and owners. The complexity of these aspects was highlighted by the need to operate at very high capacity factors with low periods of shut down-but with electricity base-load prices, which are competitive with other plant and fuel sources of generation.

Documentation and relevant archive material need to be identified and stored in a comprehensive and retrievable way from the earliest stage of the plant life. These needs should be specified at an early stage and International guidance is sought.

Databases are an important adjunct to both the operation of plant and as a depository for external experience of data for current evaluations. Guidance is sought on operational record data storage, its management system, and to review requirements.

PLIM approaches seem to be harmonising in their approach in different countries and there appears to be advantages in comparing and contrasting these approaches to provide a harmonized guidance. It was also stressed that the PLIM process should commence at the earliest part of overall plant life. Exchange meetings - such as this Symposium were thought to be an important requirement.

The need to better understand the underlying mechanisms of degradation and associated "acceptable level of degradation" of components was needed. In particular the rate of degradation was a crucial requirement, so that appropriate and timely action can be taken. This was again an area where exchange of information was highly desirable and is likely to continue as ongoing underlying areas of study and exchange for mutual benefit.

Monitoring of degradation was important to provide early information on component condition. Automatic techniques with computerized record keeping could provide a more comprehensive evaluation of component condition. However the 100% inspection of key areas of Reactor Pressure Vessels has shown the large benefits that can derive from the Structural Integrity Analyses.

The benefits of significant increase in the operational life from PLIM considerations with regard to, for example, the increased generation of electricity and the benefits to the consumer need more public emphasis and explanation. This is to spell out the benefits of electricity and nuclear power.

Indeed, participants thought that the whole area of 'public relations' in Nuclear Power acceptance required increased emphasis by the Industry.

An allied problem was to do with the difficulties in recruiting young entrants into the Nuclear Power Industry and their perception of the Industry. It was thought that the IAEA in particular could address and perhaps provide assistance for particular countries.

Concerns were expressed about the falling staff levels in the Industry and the need to address this worldwide problem. Specialised training needs should also be addressed. Political decisions on Nuclear Power were shown to have a dramatic effect on the infrastructure of the educational system requirements.

It was noted that the political, social and economic drivers underlying economical evaluation (which are used to justify Nuclear Power) vary markedly in the world. But, fundamentally the operating period of Plant Life and its efficiency (LF) determine the revenue for the NPP. The higher these criteria then the cheaper the electricity. Similarly, capital charge and the Amortization Life have an impact. It was noted that Amortization Life varies in different countries and sometimes for different plant in the same country.

Increasing power output and increasing the plant efficiencies also contribute to decreasing the cost of the electricity produced. Other factors such as environmental benefits and the cost of fuels, which may have to be imported, are also important factors.

The anticipated plant behaviour is important in assessing its structural integrity of Category 1 components – whose failure could have catastrophic consequences. Judgements on ongoing assessments and the input data requirements are obviously important.

International Organisations continue to contribute knowledge to understanding and application of these key degradation mechanisms. There are important improvements in both techniques and methodology. It is also important for these improvements to be propagated. While this help could be in the form of training and support for 'kit', the role of Coordinated Research Projects should be stressed and perhaps utilized. To emphasize these projects there should be continuation of exchanges of programme information between International organizations.

Let me finish by saying that the recent dramatic technological developments are illustrated by:

1. The Re-licensing of plants in the USA to give an operational life of 40 years,
2. The assessment of French plant operational life to be 40, 50, 60 or even more years,
3. The considerations in Japan, which could lead to 60 years operational life.
4. The current developments in Hungary.

These should lead to lower nuclear electricity prices for the consumer and also help to meet the need for increased electricity for the future.

P.E. Juhn's closing remarks

I hope you have had a good meeting and good stay during last four days. I observe that the Symposium has been very successfully and fruitfully preceded with your active participation and discussion. I anticipate that you will agree with me in bringing the words of gratitude on your and the IAEA behalf to those who particularly made this event successful.

First of all, I would like to thank the Members of the International Steering Committee and its Chairmen Messrs. Davies and Hutin who did a great deal of work in guiding the Symposium and bringing its ideas and content to the attention of national respective organizations what resulted in a very good attendance and high quality papers.

I should mention also the work of the Hungarian Local Organizing Committee, which has performed excellent job in assuring the success of the Meeting. I would like to express my gratitude to its members, Messrs. Istvan Vidovszky and Ferenc Gillemot as well as to the technical supporting staff of KFKI in particular, Messrs. L. Sagi and P. Tamas. Special thanks go to Mr. Levante Tatar who has been very helpful during the Symposium and in particular with Power Point presentation arrangements.

Thanks to the "Contours Congress and Travel Bureau" and it's representative Ms. Judith Burjan who supported the timely arrangements for the hotel accommodations and organisation of the technical tour tomorrow to Paks NPP.

We are grateful to the Authorities of NPP Paks for their involvement in the process of Symposium preparation and kind hospitality provided to its participants.

I would also like to express special thanks to Ms. Karen Morrison of the IAEA who put a tremendous effort in almost every activity connected with the preparation of and running the Symposium.

Thanks to Messrs. G. Soukhanov and V. Lyssakov, who acted as a Symposium co-ordinator and a Scientific Secretary, for their work on the preparation of and during the work of the Symposium.

Finally I would like to thank all of you for being here until this moment and contributing to the work of the Symposium and making it a great success.

Let me wish you all the best in your future activities on PLIM & PLEX and have a safe journey back home, and I hope meet you all again in the next PLIM meeting to be organized by the IAEA hopefully in next 4 years.

With these words, I declare the International Symposium on NPP Life Management closed.