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Technologies for improving current and future light water reactor operation and maintenance: Development on the basis of experience

Proceedings of a Technical Committee meeting held in Kashiwazaki, Japan, 24–26 November 1999

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TECHNOLOGIES FOR IMPROVING CURRENT AND FUTURE LIGHT WATER REACTOR OPERATION AND MAINTENANCE: DEVELOPMENT ON THE BASIS OF EXPERIENCE

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FOREWORD

Application of efficient technologies for improving operation and maintenance of nuclear power plants is an important element for assuring their economic competitiveness with other means of generating electricity. The competitive environment, which nuclear power plant operators face in many countries as a result of de-regulation of the electricity market, imposes cost pressures that must be met while at the same time satisfying stringent safety requirements. Further, as currently operating plants age, proper management includes development and application of better technologies for inspection, maintenance and repair. For future plants, the opportunity exists during the design phase to incorporate design features for performing efficient inspection, maintenance and repairs.

This Technical Committee meeting was hosted by the Nuclear Power Engineering Corporation at the Kashiwazaki-Kariwa Nuclear Power Station of the Tokyo Electric Power Company in Kashiwazaki, Niigata Prefecture, Japan, 24–26 November 1999, to exchange information on technologies for improving operation and maintenance for current and future light water reactors (LWRs). The meeting was convened within the frame of activities of the IAEA's International Working Group on Advanced Technologies for Light Water Reactors. Topics addressed within the frame of this International Working Group focus on technology developments for improving the economic competitiveness of current and future LWRs while meeting safety objectives.

The IAEA officers responsible for this publication were J. Cleveland and T. Mazour of the Division of Nuclear Power.

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SUMMARY

BACKGROUND AND OBJECTIVE

Despite the prevailing low prices of fossil fuels, the generation costs of nuclear electricity continue to be competitive with electricity generation costs from fossil-fuelled plants for base-load generation in several countries. For nuclear power, the capital investment component of electricity generation cost is relatively high, while the nuclear fuel cycle cost is — and is expected to remain — relatively low. The prices of fossil fuels are fairly low today but are likely to increase over the long term because the resource is limited. Moreover, governments may introduce incentives to reduce the use of fossil fuels in order to protect the environment.

In many countries, nuclear utilities are experiencing increased competition with other sources of electricity production due to deregulation of the electricity market, and nuclear plant operators can no longer pass along the generation costs to consumers through regulated electricity rates. This competitive environment has significant implications for plant operations to achieve efficient use of all resources, and to effectively manage plant activities including outages and maintenance.

Over the past several years, steady improvements have been experienced in nuclear power plant operation and maintenance. Worldwide, the average energy availability factor has increased from approximately 70 percent in 1989 to 79 percent[1] in 1998, with some utilities achieving significantly higher values. This is being achieved through integrated programmes, including personnel training and quality assurance, from improvements implemented in plant systems and components, and from improvements in outage duration for maintenance, refuelling, and other scheduled shutdowns, as well as forced outages.

International co-operation is a key role in this success. The various programmes of the World Association of Nuclear Operators (WANO) to exchange information and encourage communication of experience, and the activities of the IAEA including projects in nuclear power plant performance assessment and feedback, effective quality management, and information exchange meetings on technology advances, are important examples of international co-operation to improve the performance of nuclear power plants.

This IAEA Technical Committee meeting (TCM) was convened to provide a forum for information exchange on technologies for improving plant operation and maintenance which can contribute to enhanced economic competitiveness of LWRs, and on on-going or planned technology development expected to achieve further improvements. Information exchange of recent positive experience on development and/or implementation of technologies that have resulted in reduced operation and maintenance costs of LWRs, and information on on-going or planned activities, was strongly encouraged.

CONDUCT OF THE MEETING

The TCM was attended by 40 participants from 12 IAEA Member States (China, Czech Republic, France, Germany, Hungary, Japan, Republic of Korea, Russian Federation, Slovak

[1] Based on IAEA Power Reactor Information System (PRIS) data. In PRIS, the energy availability factor is defined as 100 [1-EL/E_m] with E_m being the net electrical energy which would have been produced at maximum capacity under continuous operation during the reference period, and EL is the electrical energy which could have been produced during the reference period by the unavailable capacity. (The numbers reported here are for plants with capacity greater than 100 MW(e) and with more than one year of commercial operation).

Republic, Sweden, United Kingdom and the United States of America), and the World Association of Nuclear Operators (WANO).

Welcoming addresses were given by K. Nakamura, Deputy Director of the Nuclear Power Division, Agency of Natural Resources and Energy, Ministry of International Trade and Industry, Japan, and by M. Idesawa, Associate Director, Superintendent, Kashiwazaki-Kariwa Nuclear Power Station, Tokyo Electric Power Company (TEPCO).

The meeting was chaired by H. Ogasawara, Director and General Manager of the Systems Safety Department of the Nuclear Power Engineering Corporation (NUPEC), Japan.

A total of 23 technical papers were presented in the following areas:

- Programmes for achieving high performance and reliability;
- Equipment and techniques for component inspection, maintenance, repair and replacement;
- Methods for reducing refuelling outage duration; and
- Advances in design and technologies for improving plant operation and maintenance.

A technical tour of the Advanced Boiling Water Reactor (ABWR), Kashiwazaki-Kariwa Unit 6, was provided by TEPCO.

RESULTS

Programmes for achieving high performance and reliability

Key factors determining the economic competitiveness of a nuclear power plant (NPP) include the operational lifetime of the plant and its operational costs. “Plant operational life” is not a defined period, with different countries taking different approaches. In some countries, there is a “licensed life” for nuclear power plants with possible consideration of life extension, while in others there is a “periodic safety assessment” to approve the plant for a further fixed period of operation. Political decisions to end the operation of a plant before it reaches its technical or economic lifetime also occur.

In Japan, the government and the industry have close interaction in managing NPP lifetimes and in establishing the necessary inspection and maintenance activities and requirements for countermeasures against ageing. The Ministry of International Trade and Industry (MITI) has prepared a basic policy on aged nuclear plants. Some of the key elements of the policy focus on:

- the importance of periodical inspections of major components such as the reactor vessel and primary piping;
- thorough preventive maintenance;
- the further development of inspection and repair technologies, and the acquisition of material data and operation data related to ageing.

Technical assessments have been conducted by the Japan Atomic Power Co., the Kansai Electric Power Co., and TEPCO for three units which are approaching 30 years of commercial operation: the Tsuruga Unit 1, Mihama Unit 1, and Fukushima Dai-ichi Unit 1. These assessments have considered all components and structures related to safety performance. The assessments have been based on a 60-year goal. These assessments have been reviewed by MITI and have resulted in plans for long term periodic inspection and maintenance programmes to incorporate ageing countermeasures. The conclusion for these three plants is that safe operation for lifetimes as long as 60 years is possible with correct and adequate inspection, maintenance and repair. Other plants will be assessed in the future.

On the other hand, an example of a planned shutdown of a plant before it reaches its design lifetime can be found in the Slovak Republic where two units of the VVER-440 V-230 series have been operating at the Bohunice nuclear power station since 1978 and 1980, respectively. Based on the design lifetime of 30 years, the originally planned decommissioning dates were 2008 for Unit 1 and 2010 for Unit 2. Since the plant commissioning, many safety upgrades and operational reliability improvements have been made on these units. Improvements, with completion scheduled in 2000, are largely implemented and include: reconstruction of several systems (e.g. the emergency core cooling system, electrical systems) to achieve separation, redundancy and independence, addition of an emergency feedwater system, reconstruction of the instrumentation and control system, and seismic improvement. Another improvement to both of the Bohunice units was the annealing of the reactor pressure vessels in 1993 so that brittle fracture and fatigue damage would not reduce the reactor lifetime below the 30-year design life. Nevertheless, in autumn, the Slovak government decided, for political reasons, to decommission these units in 2006 and 2008, respectively, two years before reaching their 30-year design lifetime.

Four units of the newer design VVER-440 series 213 began operation in the Czech Republic during 1985–1987. As a result of intensive internal and international safety reviews, considerable modernization and upgrading of these plants have been, and are being, conducted to achieve a higher level of safety. Modernization includes, for example, improved fire protection, an improved instrumentation and control system, modification of the emergency feedwater system, and measures to control hydrogen buildup during accident conditions. Improved economics is also an important goal. This includes the goal to operate all four units to the end of their planned lifetime (2015) with possible extension of this lifetime to 2025, and use of design margins for increasing the power rating of each unit by approximately 20 MW(e).

An example of successful experience in importing nuclear technology together with technology transfer is provided by the Guangdong Nuclear Power Station in China where two 984 MW(e) PWR units of French design began commercial operation in 1994. The Guangdong station has reached the top quartile in over half of WANO’s 10 performance indicators. This has been achieved while at the same time reducing the cost of power generation at Guangdong and while increasing the self-reliance of China’s nuclear power programme through technology transfer. This positive experience is being fed into the LingAo Nuclear Power project at which construction started in 1997 for two PWR units that are mainly copies of the units at Guangdong, but with some improvements. While there is a need for continued involvement with the plant manufacturer, this experience feedback is resulting in China being able to locally fabricate some components and to have a high level of self-reliance in project management.

Designers of new plants have incorporated technical features for improved operation and maintenance into the design of the plants. The two ABWRs at the Kashiwazaki-Kariwa site provide a good example. These two ABWRs (Kashiwazaki-Kariwa Units 6 and 7) began commercial operation in 1996 and 1997, respectively. Outage times have been very short at these two units even though they are the largest capacity units in Japan, and incorporate new design features compared to earlier BWRs in Japan. This success is largely due to the “test-before-use" approach that was applied for ABWR components and systems that had not been used previously in Japanese BWRs, even if the components and systems have been used in plants outside of Japan.

Overhaul and inspections of new design features of these ABWRs (including reactor internal pumps, advanced control rod drive mechanisms, and high efficiency steam turbines) were conducted during the first two outages at each plant and confirmed the integrity of these components. Furthermore, the occupational radiation exposure during the ABWR outages has been low compared to other plants at Kashiwazaki. This good experience results in part from the training that maintenance personnel gain at the full-scale reactor maintenance training facility at the Kashiwazaki-Kariwa site. This experience will be useful not only in future outages at the two ABWR units at Kashiwazaki-Kariwa ABWRs, but also at new ABWRs planned in Japan and elsewhere in the world.

With regard to plant reliability, an important effort has been the assessment and solution of the Y2K problems at NPPs. In this regard, TEPCO began their assessment in 1996 and examined both the plant monitoring systems and the control systems for Y2K readiness. While no control system modifications were required, it was found that some monitoring systems required software modification. TEPCO completed Y2K tasks in October, 1999.

Equipment and techniques for component inspection, maintenance, repair and replacement

New equipment and techniques have been developed in several countries for component inspection, maintenance, repair and replacement. To support the goal of carrying out applications of these techniques as economically as possible, considerable laboratory testing and testing in full-scale mock-ups have been performed before their applications. This is the case, for example, for

- chemical decontamination of the reactor system to prepare for replacement of core internals;
- holographic inspection methods for sizing and recognition of cracks;
- water-jet peening for mitigation of stress corrosion cracking; and
- development of manipulators to carry out in-pipe inspection, grinding and for repairing cracks in welds between the vessel nozzle and pipe.

Intergranular stress corrosion cracking has affected the core shroud and other reactor internals components made of SS type 304 at plants in Japan. As a result, a programme of replacement with components made of SS type 316 is underway at TEPCO. The first unit to undergo core shroud replacement was Fukushima-Dai-ichi Unit 3. The experience gained in this operation was applied at Unit 2 with the result of a reduced outage and reduced total radiation exposure. Prior to replacement, chemical decontamination of the reactor system has been conducted using a combination of oxalic acid and per-manganic acid which was circulated throughout the reactor vessel by operating the recirculation pumps at minimum speed. TEPCO is currently planning the core internals replacement work at Fukushima-Dai-ichi Unit 5.

To support this extensive core internals replacement programme in Japan, an extensive programme has been underway since 1995 at NUPEC to establish the reliability of reactor internals replacement techniques using full-scale mock-up models. Tests of replacement methods for BWR core housing, core shroud, control rod drive housing, and jet pump riser brace, and for PWR core barrel and bottom mounted instrumentation adapter are included in the eight year programme, which concludes in 2002.

Reliability tests are currently underway in Japan for the method of replacement of the core barrel of PWRs. A full-scale mock-up test will be carried out in early 2000. Additional mock-up tests of the PWR bottom-mounted instrumentation adapter and the BWR jet-pump riser brace are also planned.

Methods for reducing refuelling outage duration

Considerable improvement in outage time and generation costs may still be achievable at plants that have been operating for several years through technical and administrative measures, and application of computer tools for planning and managing plant outages. Two examples of plants which have been operating for several years for which improvements in outage time and generation costs have recently been realized are provided by the Indian Point 2 plant in the US and the Paks Nuclear Power Station in Hungary. Indian Point 2 has shown that replacement of two complex safety systems with new passive systems can reduce or eliminate maintenance and testing requirements during outages, and thereby shorten the outage duration. The modifications involved (a) replacement of conventional hydrogen ignitor systems for control of hydrogen in the containment during a hypothetical loss of coolant accident with passive autocatalytic hydrogen recombiners, and (b) replacement of the conventional containment spray additive tank with baskets of tri-sodium phosphate. The conventional system had the function of removing iodine from the containment atmosphere in the event of a loss of coolant accident, but required significant maintenance and testing. The replacement system requires no maintenance or testing, as the chemical will dissolve into the spray when the spray system is activated. These two new passive systems replaced the conventional systems which comprised several hundred components that depended on electric power and other complex control and support systems.

The Paks NPP provides the second example. Through a modified fuelling scheme, Paks has reduced fuel costs significantly (more than 16%), and changes in outage planning have reduced outage duration. Further economic gains in the fuel costs are expected from qualifying fuel from a second manufacturer to promote competition.

For future plants, features to reduce outage duration can be incorporated at the design stage. Within the CIDEM project, Electricité de France, together with some German utilities, has reviewed the operating experience of French NPPs as well as foreign units to identify means of reducing the outage duration, particularly for the European Pressurized Water Reactor (EPR). Several design features which influence outage duration, including accessibility of the reactor building during operation, number of safety trains and others have been identified. Logistics support including requirements for special tools and availability of spare parts also influence outage duration, as does outage management. Some results could be adapted also in current plants, and a study will be carried out to examine applicability to EdF's 1300 MW(e) series plants.

Advances in design and technologies for improving plant operation and maintenance

For new plants, the opportunity exists to incorporate features for improved operation and maintenance at the design stage. Specific design features which can contribute to improved operation and maintenance in new plants include:

- redundancy – reducing vulnerability to single component failure, and enabling maintenance and repair during plant operation;
- diversity – reducing the sensitivity to "common-mode" failures;

- careful planning of plant layout and installations – to ensure accessibility for inspection, maintenance, replacement and repair; and
- design for on-line testing and maintenance.

Such features and others are being incorporated into the advanced LWR designs.

Technologies which can contribute to improved operation and maintenance in new as well as operating plants include:

- digital instrumentation and control, including self-diagnostic systems; and
- control room and man–machine interface improvements with due consideration of human factors engineering.

The advanced control room design with the validation results confirming the improvement of the operability for the next PWR in Japan and the control room design for the Korean Next Generation Reactor are two examples of these technologies.

Such technologies and others are being incorporated into the advanced LWR designs, and implemented in a number of operating plants in the world.

PROGRAMMES FOR ACHIEVING
HIGH PERFORMANCE AND RELIABILITY

(Session I-a)

Chairperson

T. PEDERSEN
Sweden

CURRENT STATUS OF LIFE MANAGEMENT POLICIES FOR NUCLEAR POWER PLANTS IN JAPAN
(Summary)

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Almost 30 years have now passed since the initial commercial nuclear power plants started operation in Japan. It is natural, therefore, that concerns related to the aging of these nuclear power plants have increased. Under these circumstances, in April 1996, the Ministry of International Trade and Industry (MITI) studied how countermeasures in response to future aging should be provided. The results of this study were summarized in the report titled "Basic Policy on Aged Nuclear Power Plants", and publicized accordingly.

The key points in the report are described below.

(1) Technical assessment was applied to the major components (reactor pressure vessel, primary piping, etc.,) of nuclear power plants. It was concluded that complete periodical inspections/examinations would enable safe operation assuming a long term operation.

(2) The safety of Japanese nuclear power plants is sufficiently ensured by the execution of periodical inspections/examinations and thorough preventive maintenance measures. Considering further aging, however, the items and content of the periodical inspections conducted by the national government and examinations conducted by the electric utility companies shall be enough established aiming at a 30-year operation period as a standard, thereby enabling further upgraded safety management.

(3) It is appropriate for the electric utility companies to apply detailed technical assessment to all nuclear power plant components after a 30-year operation, and to produce detailed maintenance schedules thereafter.

(4) The enhancement of structural codes/standards in response to the toughness change due to aging shall be constantly studied in the future by reflecting the knowledge accumulated so far and by referring to the codes/standards of the US.

(5) Technology development is required to enable the further reliable management of aged nuclear power plants. It is important, therefore, to continue to develop inspection and repair technologies. It is also important to acquire the material data and operation data of power plants in relation to their aging.

In response to the MITI report issued in April, 1996, the Japan Atomic Power Co., the Kansai Electric Power Co., Inc. and the Tokyo Electric Power Co. conducted technical assessment for all components and structures related to safety performance at the Tsuruga Power Plant Unit-1, Mihama Power Plant Unit-1, and Fukushima Dai-ichi Nuclear Power Plant Unit-1. Considering the aging, also, their new maintenance measures to reinforce the current maintenance activities were extracted and summarized in the long term maintenance schedule.

The MITI has studied and assessed the technical assessments and long term maintenance schedules of the electric utility companies while listening to the technical advisers' opinions of the Examination Subcommittee on Countermeasure for Aging Plants under the Advisory Committee on Comprehensive Preventive Maintenance. The technical advisers for nuclear power generation are scholars assigned by the Minister of the MITI. The MITI has also carried out studies regarding the future approaches to management of plant aging. The results of these studies were summarized in the second report titled "Assessment of Aging Countermeasures of Electric Utility Companies for Nuclear Power Plants and Future Approaches to Management of Plant Aging", and were published in February, 1999.

Major concluding remarks of the second report are as follows:

(1) The latest detailed technical assessment of components practiced by the electric utility companies has started with confirmation by going back to their design stage and following their long-term maintenance activity records thereafter. And, also including future aging prediction based on the latest knowledge, it was a large and diversified assessment process. Similar technical assessment shall be applied to all plants in order according to aging progression. Such assessment shall be repeated in approximately 10-year intervals. In this case, it is important to accumulate, manage and succeed to diversified technical information appropriately, including the design concept.

Regarding the aging countermeasures, long term and steady efforts are required including long term technology development, establishment of technical standards/private standards, and accumulation of material/component data. It is, therefore, recommended that the electric utility companies, manufacturers, universities, research institutes and the national government share their appropriate roles in cooperation and continue their activities accordingly.

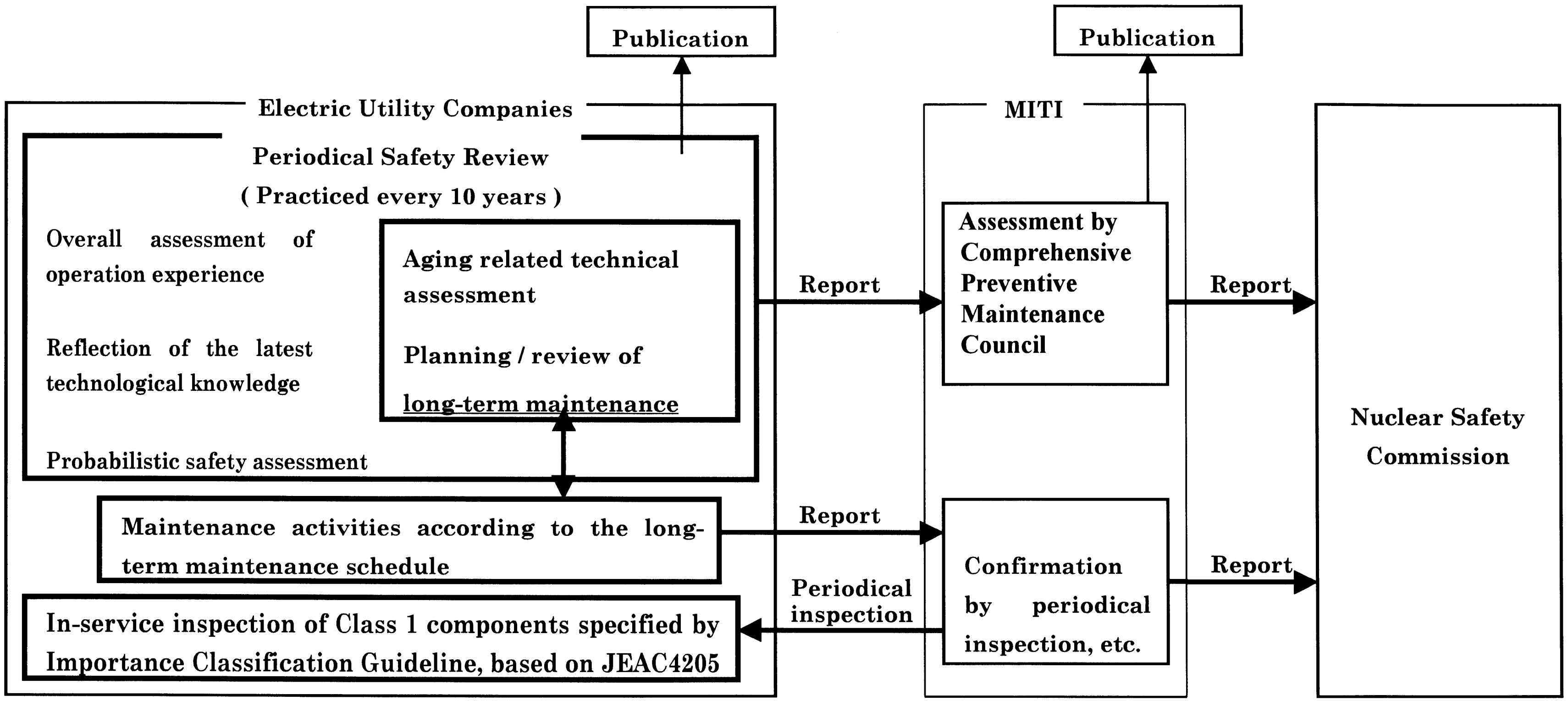
(2) The total consensus of the nation is also vital to practice aging countermeasures. Regarding the activities of the electric utility companies and national government it is, therefore, important to ensure their transparency by opening information to the public and providing with such information in a comprehensive manner.

(3) Japanese nuclear power plants have accumulated safety assurance records by practicing periodical inspections and other maintenance activities precisely. It is necessary in the future, while considering aging factors, to practice the measures described in the second report appropriately and to constantly focus on safety assurance.

(4) We shall continue in our efforts to improve aging countermeasures by reviewing them flexibly as required, based on the aging related experience and knowledge gained in Japan and overseas.

It was considered appropriate to thoroughly provide the periodical safety review (PSR) and periodical inspection as a comprehensive facility management system in response to the aging of nuclear power plants in the future. The entire scheme of the comprehensive facility management system is shown in the Figure 1.

Keywords: plant life management, technical assessment of aging degradation, long term maintenance program, regulatory aspects, periodic inspection, structural standards, R&D activity



1. Periodical safety review (PSR) shall be conducted at each plant at approximately 10-year intervals.
2. Technical assessment and establishment of long term maintenance schedule shall be included in the PSR for plants subjected to PSR based upon 30-year operation as a standard. Such technical assessment and long term maintenance schedule shall be reviewed at 10-year intervals thereafter.
3. Technical assessment and long term maintenance schedule shall be assessed by the MITI and reported to the Nuclear Safety Commission.
4. Progress state of long term maintenance schedule shall be followed up at periodical inspection, etc.

FIG. 1. Comprehensive facility management system

TECHNOLOGIES FOR IMPROVING CURRENT AND FUTURE LIGHT WATER REACTOR OPERATION AND MAINTENANCE: DEVELOPMENT ON THE BASIS OF O&M EXPERIENCES — THE WANO PERSPECTIVE

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Abstract

World Association of Nuclear Operators (WANO) plays a role of promoting safety and reliability to nuclear industry after Chernobyl accident. Four programmes or so called cornerstones, operating experience, peer review, professional and technical development, and technical support and exchange have significantly promoted the nuclear operating performance in the past years. A WANO biennial general meeting was recently held in Victoria, Canada, which disclosed apparent achievements on higher unit capability factor, lower unplanned automatic scram per 7000 hours critical, lower collective radiation exposure and lower industrial safety accident rate. In more practical, exchange visits to learn good practices and measures in operation, maintenance, and management have shown benefits among WANO members. Recurring events can be minimized when members learn lessons from significant operating experience reports, significant event reports, and those events have posted on the WANO Web site. Particularly, plant managers' meetings that Tokyo Centre host have created an environment, which allows plant management to exchange ideas by such a face-to-face channel. WANO aims at nuclear safety and reliability. Economic and public acceptance are regarded as pillars to support WANO's mission as well.

1. A BRIEF HISTORY OF WANO

World Association of Nuclear Operators (WANO) established on May 15, 1989, in Moscow, by 144 electric utilities with nuclear power plants in operation or under construction. 30 countries sent representatives to the Inaugural Meeting and signed the WANO Charter. As a result, such a worldwide non-government, non-political organization has been promoting nuclear industry on seeking excellence on nuclear safety and reliability.

A WANO Biennial General Meeting was held in Prague in 1997. Mr. Remy Carle, a former WANO Chairman, described plans for an Internal Review – an essential peer review for WANO itself. WANO's Internal Review was led by Mr. Bob Franklin and Mr. Ray Hall. They interviewed WANO staff and WANO member executives from the CEO level to the plant manager level among four WANO regions. On May 1st, 1998, Dr. Zack Pate, the Chairman of WANO, sent a letter to all members with a comprehensive description of actions planned in response to the Internal Review.

The most essential indication of the Internal Review was the confirmation that the WANO mission established at the time of inauguration is still valid. In addition, more efforts should be made to attract the attention of plant managers so that WANO programmes penetrate into members more deeply. The Internal Review also outlined various areas to be further improved in reinforcing individual programmes. As a result, a new WANO programme realignment is noted as four cornerstones:

- Operating experience
- Peer review
- Professional & technical development
    - Workshops/Seminars/Courses
- Technical support & exchange

— Good practice
— Operator exchange visits
— Performance indicators
— Technical support missions
— WANO network

2. HOW WANO PROGRAMMES HELP ITS MEMBERS

2.1. Operating Experience (OE) Programme

OE programme alerts members to events that have occurred at other nuclear plants and enables members to take appropriate actions to prevent event recurrence. When an event occurs at a plant, management at that plant analyses the event and completes an event report, which is then sent to a WANO regional centre that the plant belongs to. After a regional centre review and necessary iteration, the report is posted onto the WANO Web site to make it available to all WANO members. By the middle of 1999, over 1830 event reports were posted.

After the Internal Review, two significant operating experience reports (SOERs) have been issued to date. One on “Safety System Status Control” was issued in 1998 and is based on nine related events that had occurred in the previous two years. The second SOER on “Loss of Grid” was issued in mid-1999 and is based on six worldwide events. In addition, earlier this year WANO issued its first significant event report (SER) on an event that related to the main steam safety and relief valves being unavailable during a plant transient. Later, the second SER on the spurious actuation of containment sprays that resulted in a severe plant transient was published as well.

WANO respects its members to review these significant documents in the light of their own plant procedures, policies, and practices to determine how these operating experiences can be applied to further improve safety and reliability. Implementation of SOER recommendations is reviewed in all peer reviews commencing six months after the issue of the document.

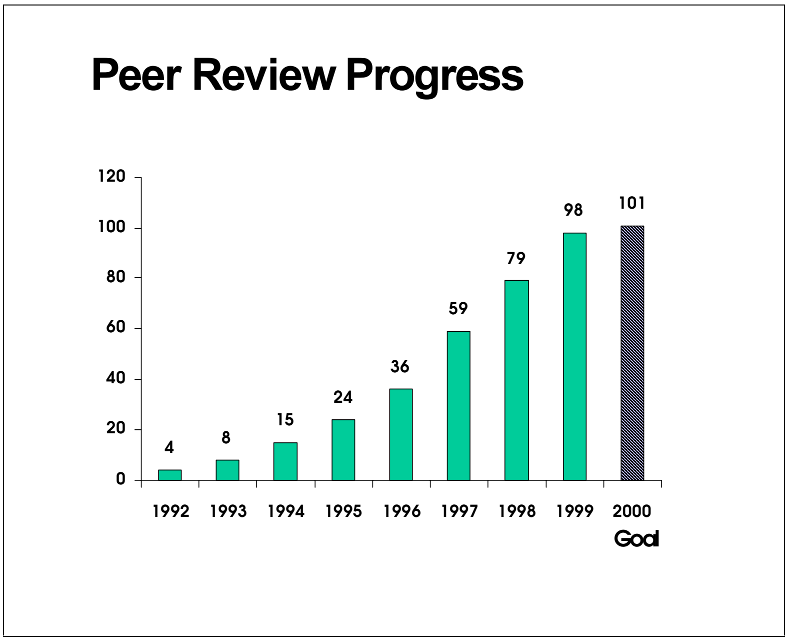
An on-line event database has been developed and made accessible to all WANO members via the WANO Web site. Members can search this database in a variety of methods to obtain necessary OE information.

2.2. Peer Review Programme

WANO Peer Review Programme is a unique opportunity for members to learn and share the best worldwide insights into safe and reliable nuclear operations.

WANO peer review teams cover a review to areas of organization and administration, operations, maintenance, engineering support, radiological protection, operating experience, chemistry, training and qualification, emergency preparedness, and fire protection. By the performance objectives and criteria (PO&Cs), cross-functional areas, such as safety culture, human performance, and self-evaluation are also reviewed by worldwide peers.

WANO peer review identifies aspects of a plant’s safety and performance that need attention when viewed against world best standards. As such it enables an action plan and priorities to be established. Experience has shown that even excellent plants can learn from a peer review, as well as passing on strengths to the team members for them to take back for emulation. Through to the September of 1999, 100 WANO peer reviews had been conducted. As the peer review goal, nearly 50% of all sites will have had a WANO peer review by the end of this century.



With 100 peer reviews, the most two frequently occurring areas for improvement (AFI) of management issues, each with their own underlying causes, that the teams identified are:

- Managers are not sufficiently in the plant and when in the plant, do not correct improper practices.
- Management expectations are not clearly established, not communicated, not understood or not reinforced.

Solutions to these types of performance problems need to reflect local policies, practices, culture, and site history. However, WANO is looking at the full analysis to identify ways in which WANO can focus its programme activities in the light of the frequently recurring themes. Examples of how WANO might do this include technical support missions concentrating on specific frequently occurring topics and addressing common issues in workshops and courses under the professional and technical development programme.

2.3. Professional & Technical Development (PTD) Programme

WANO PTD programme provides a means for WANO member personnel to exchange information and experiences through a variety of workshops, seminars, and training courses. Through to the end of 1999, WANO will have conducted a total of 200 such events.

During 1998, WANO held four inter-regional seminars on the topics of “How to conduct WANO Peer Reviews” in Russia, “Usage of Performance Indicators” in Ukraine, “Human Performance Enhancement System” in Czech Republic, and “Fast Breeder Reactors’ Group Meeting” in Kazakhstan. In addition, 16 workshops were held in the same year.
In 1999, 19 workshops, six expert meetings, two seminars, and 14 training courses are provided to WANO members, which include workshops on:

- The Application of Probability Safety Analysis – Moscow Centre
- Nuclear Plant Transition from Operation into Decommissioning – Paris Centre
- Public Relations in the Local Community – Paris/Atlanta Centres

- Upgrades and Reliability Enhancements of NPP's Instrument and Control Equipment – Tokyo Centre
- Plant Modification and Dose Reduction – Tokyo Centre
- Nuclear Power Plant Life Management – Moscow Centre

Inter-regional training courses are being continuously conducted among regional centres in 1999, which include topics on:

- Control Room Teamwork Development Training
- Conservative Decision Making
- Improving Human Performance
- Supervisors Professional Development
- WANO Peer Reviews Team Leaders
- WANO Peer Review Exit Representatives

2.4. WANO Technical Support & Exchange (TSE) Programme

WANO TSE programme includes a variety of methods for members to exchange nuclear plant operating information and experience to enhance safety and reliability. Subjects of the programme include good practice (GP), operator exchange visits (EV), performance indicators (PI), technical support missions (TSM), and WANO Network.

2.4.1. Good Practice

Good practices enable WANO members to learn from each other's best practices and thereby improve operational safety and reliability. A good practice describes one means of addressing an issue, and it may be a specific action or a comprehensive description of complex activities. Once identified, good practices are made available to members through WANO Web site and annual reports.

A total of 200 plus good practices are available to be retrieved at the database in which 39 good practices have been selected and posted onto the WANO Web.

2.4.2. Operator Exchange Visits

Based on the broad acceptance of WANO exchange visits, members continue a high level of activity on their own initiatives including exchange of personnel, documentation, and any other exchange or cooperation between members' nuclear power stations, usually independent of direct involvement of WANO regional centres.

Through the end of 1998, a total of 443 exchange visits and twinning activities have been conducted.

2.4.3. Performance Indicators

WANO performance indicators provide a means by which members can assess the performance of their plants objectively. Use of such indicators supports the exchange of operating experience information by collecting, trending, and disseminating nuclear plant performance data in 10 key areas. With each member providing data on its performance, WANO members can then compare their performance to that of other plants around the world. Performance indicators are mainly used as a management tool so each member can monitor its own performance and progress; set challenging goals for improvement. 10 indicators are noted as follows:

- Unit Capability Factor
- Unplanned Capability Loss Factor
- Unplanned Automatic Scrams per 7,000 Hours Critical
- Thermal Performance
- Collective Radiation Exposure
- Volume of Solid Radioactive Waste
- Industrial Safety Accident Rate
- Safety System Performance
- Fuel Reliability
- Chemistry Indicator

WANO members report on these 10 indicators on a quarterly basis. The data is collected by the Atlanta Centre, where it is entered a computer database, trended and redistributed to WANO members via the WANO Web and annual reports.

2.4.4. Technical Support Mission

Technical support visits are in response to members' specific requests for assistance in areas such as control room staffing, reactor engineering, work management, outage scheduling, and problems encountered during construction stage. During a technical support mission, a team of experts from regional centres and members plants work closely with host plant personnel to find solutions to known problems and methods to correct. Through the end of 1998, 69 technical support missions have been conducted. Additional 29 technical support missions either have been conducted or will be completed in 1999.

2.4.5. WANO Network

Electronic communications among WANO members is available via the WANO Network. In addition to its use for sharing industry event reports, the private global computer network allows all WANO member personnel the opportunity to exchange information on plant issues and share experiences.

To help members share information about year 2000 related computer issues, a special section on Y2K was added to the WANO Web site.

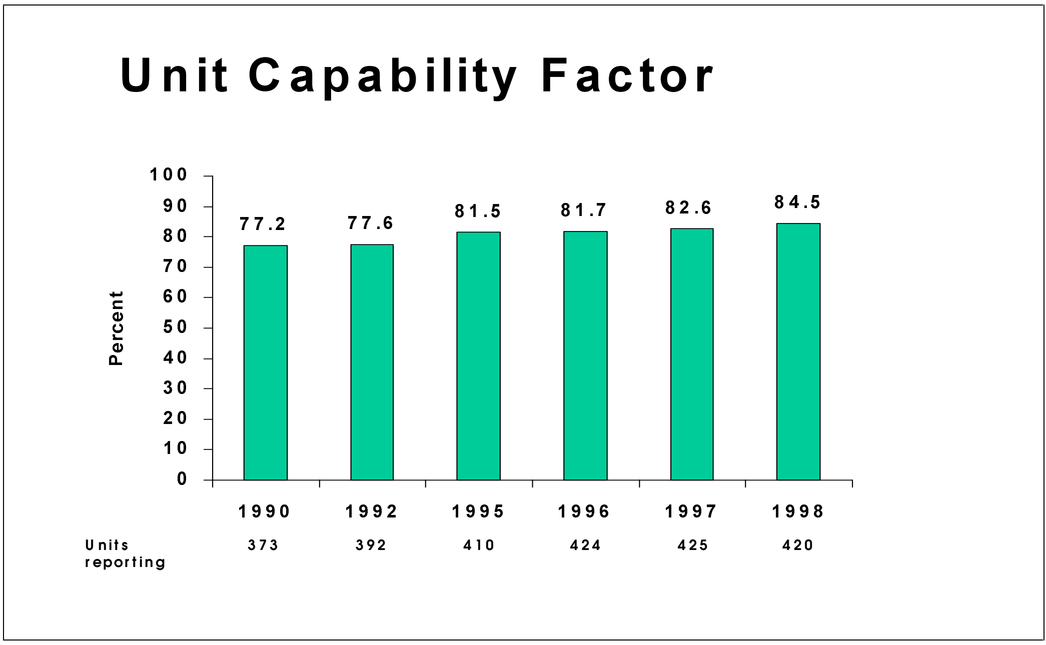
3. VALUES OF WANO

Performance Indicators have become a widely accepted measure of plant performance and progress. The WANO Performance Indicators are now in use at every plant in the world. Here is some good statistical evidence:

Performance Indicators

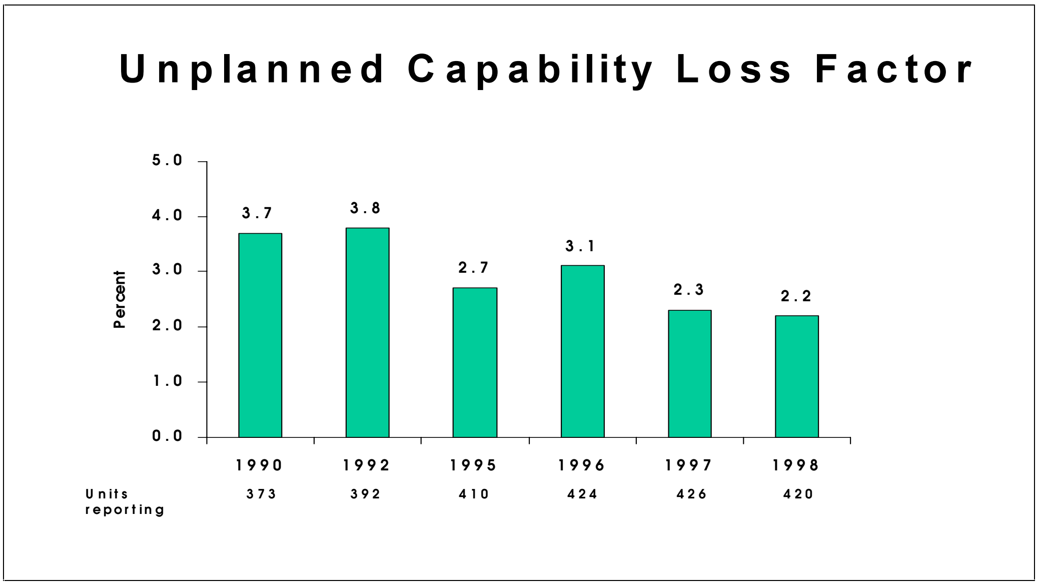
- 100% of plants are reporting data.
- 98% of plants are reporting data for at least 7 indicators.
- All utilities have agreed to make their plant-specific data available to the WANO community.

The indicators are widely used by WANO members to set goals and to monitor progress. Some worldwide trends are presented as follows.

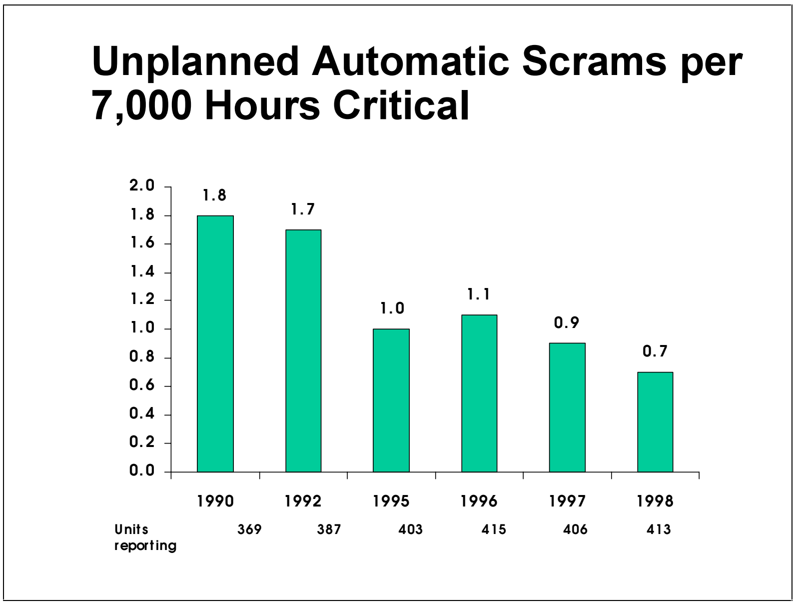


WANO now has collected nine years of data. Note that 420 units reported Unit Capability Factor data for 1998. And, of course, note the steady trend of progress.

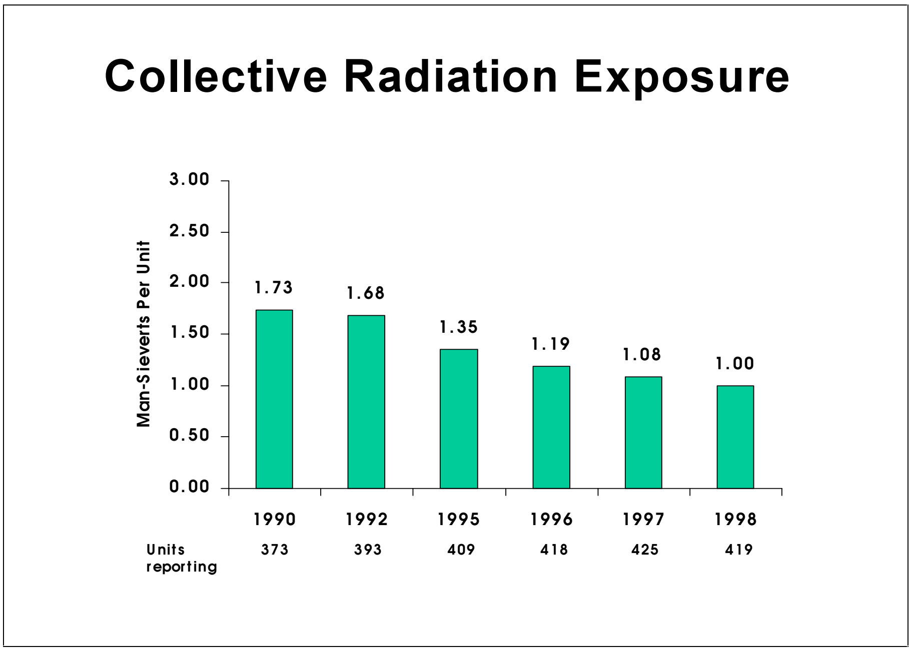
The following figure shows good progress for Unplanned Capability Loss Factor.



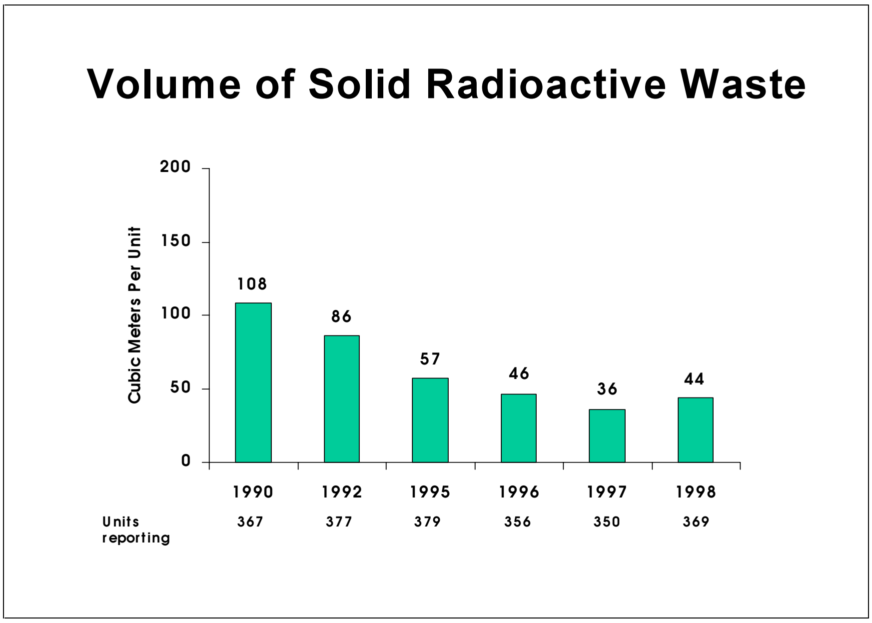
The Unplanned Automatic Scrams. Note that WANO members are now averaging well under one scram per year per unit worldwide.



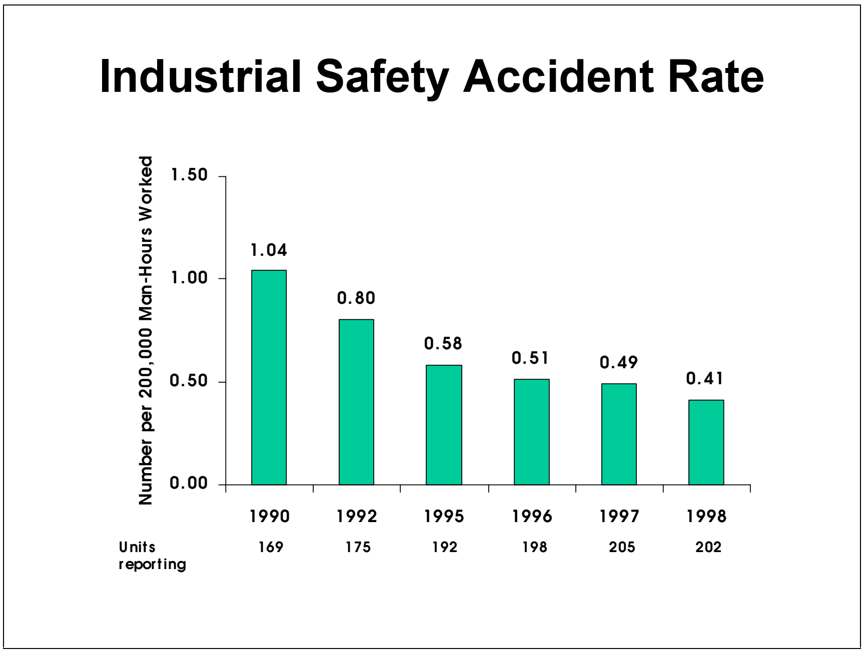
The Collective Radiation Exposure; 419 units reported data for 1998. Quite an impressive downward trend in worldwide radiation dose.



The Volume of Solid Radioactive Waste. Note one went up slightly in 1998, but the long-term trend is favorable.



And finally, Industrial Safety Accident Rate.



Some words that we quote from Dr. Zack Pate the Chairman of WANO who concluded such values of WANO during a WANO Biennial General Meeting in Victoria, Canada; "I think we can all take a great deal of pride in the progress shown by these slides. This progress is a direct result of applying the WANO mission. When plants communicate, compare, and then emulate the best performers; improvement is the clear result. These words – communicate, compare, and emulate – are taken directly from the mission statement, but it is the plants and not WANO that have done the work that has led to these improving trends. On the other hand, WANO has facilitated and enabled the communication and comparison through its Performance Indicator program."

4. CONCLUSION

At Victoria WANO Biennial General Meeting held in September 1999, which was also on the occasions of 20^{th} anniversary of Three Mile Island, and the 10^{th} anniversary of WANO, Dr. Allen Kupcis the former WANO President questioned that "Why is it that WANO isn't just merely existing, but is positively thriving in most areas?" WANO members' senior officers all agree to what Dr. Kupcis believes: "The reason WANO exists and thrives today is that WANO continues to have only one fundamental focused mission – maximize safety and reliability in nuclear operations."

WANO can contribute to the future of the nuclear industry by providing a forum through which nuclear utilities worldwide can improve their safety and reliability.

THE SIGNIFICANCE OF PLANT LIFE MANAGEMENT

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Abstract

The paper carries a definition and describes Plant life and plant life management. It also describes the procedures and defines the categorisation of components giving examples and referring to key components. Examples of 'good practice and guidance' are given for the establishment and implementation of plant life management programmes. A description is given of recent and current IAEA activities under the aegis of the International Working Group on Nuclear Power Plant Life Management (IWG-LMNPP). Some of the future activities in this field are described.

1. INTRODUCTION

This purpose of this paper is to focus on plant life management as a process to combine ageing management and economic planning to allow the optimisation of operation, maintenance and operational life of Systems, Structures and Components (SSCs) in Nuclear Power Plants (NPP) — in order to maintain an acceptable level of performance and safety. Safety is paramount. The objective is to maximise the return on the investment during the operational life of the NPP. [1, 2]. To this end the definitions of plant life will be discussed against the background of the current NPP capacity in the world.

The purpose of a NPP is to generate electricity for sale. This has to be done in many parts of the world against a background of increasing competition from the use of fossil fuels. However, the overall sales of electricity continue to increase at a rate higher than the population increase. The world's population increase and the increasing electricity use per head of population means that the people of world will need all the electricity that can be generated in the future in a cost effective way-in order to sustain the development of social conditions. Major constraints on fuel usage are the impact of an international agreement on limiting 'greenhouse gas' emissions and the location/availability/location of fossil fuels — which will be used in increasing quantities in the future.

In an increasingly liberalised market nuclear power has to be competitive, and so attention will continue to be given to the operational costs of already operating plants (because not many new ones are being built) to maintain or compete with fossil burning plants.

2. DEFINITIONS OF PLANT LIFE

The definition of PLIM and PLEX are apparently familiar and well known. However, what is not generally understood is that the full scope of Plant Life includes the pre-operational activities including the initial choice of Nuclear Power Plant (NPP) and its construction, through to the post-operational activities of de-commissioning and the return of the site to a 'green field'. Indeed a practical description of *'Nuclear Power Plant Life'* is the period when charges can be made against the NPP. It is also not generally understood that what is generally termed 'plant life' usually refers to 'plant operational life' — the period when electricity is made and sold and the plant is earning money (Figure 1). However, end-of-life activities and de-commissioning are increasingly considered as part of the process of plant life management.

'Plant operational life' is not a defined period. The approach used in different countries varies. In some countries there is a *'licensed life'* and in others there is a *'Periodic Safety Assessment'* — to clear the plant for a further fixed period of operation (see Figure 2 for some national examples). If there is no 'fixed' end of operational life then the period remains undefined. It also follows that *'Plant Life Extension'* is a misnomer because plant life itself is undefined. The term PLEX is now falling out of favour. One term in use is 'design life' and this should be viewed as a 'target minimum operational life'.

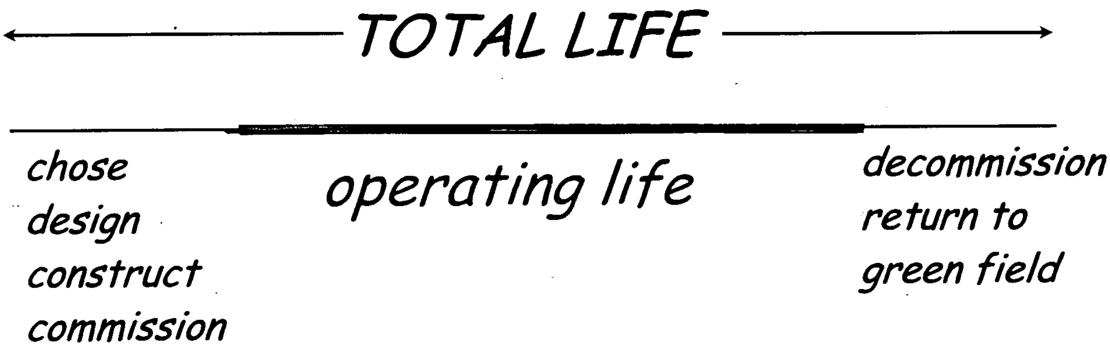


FIG. 1. Nuclear Power Plant Lifetime.

License period	Unlimited period P S Review
Canada (0.5-3 years)	Belgium (10 years)
Finland (10 years)	France (10 years)
Hungary (12 years)	Germany (10 years)
Netherlands (10 years)	Japan (10 years)
United States (40 yrs)	United Kingdom (10 years)

FIG. 2. Examples of period of operation of NPP.

Plant life management was referred to as PLEX in earlier times — which was to do with the adjustment of the 'design life' by re-evaluating the actual plant, its actual operation, the impact of ageing phenomena, the impact of improved maintenance and the role of inspection.

The initial aim is to operate NPP for the period that allows the full recovery of the capital costs. Sometimes this is known as the amortisation period. But the capital cost recovery is only part of the total costs. Allowance has also to be made for the, for example, post operational liabilities for de-commissioning, fuel storage/reprocessing and the costs of eventually returning the site to a 'green field'. Some NPP have been shutdown prematurely because of accidents (e.g. TMI, Chernobyl) or by Government decision (e.g. Zwentendorf). Some have been shutdown because they have been assessed as being too old or un-economical (e.g. Trawsfynydd, Tokai Mura). However most Owners of NPP would wish to continue to operate their plants as long as they were safe and economical.

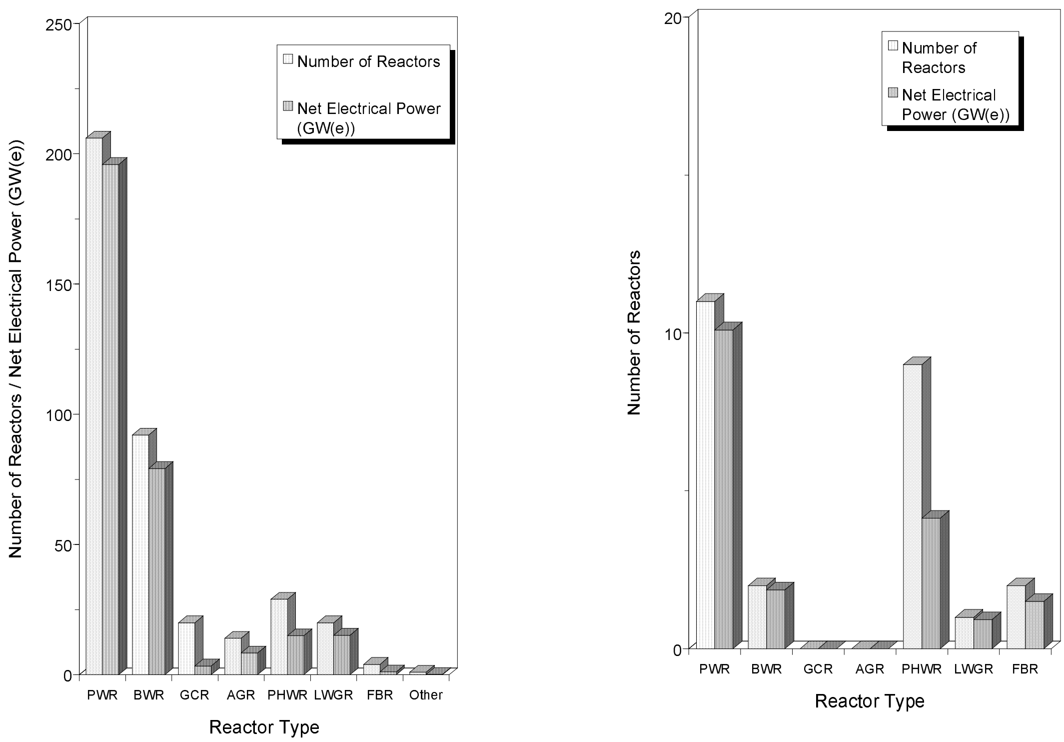
There are different approaches to plant life in different countries. In my own country we have the oldest Gas Cooled NPP — one of which has provisional clearance to operate for fifty years. Our Sizewell B PWR has an amortisation period of forty years. In Japan the studies by JAPEIC indicate that a target life of sixty years for NPP is feasible. In Russia the design life of NPP is thirty years. In the US NPP were designed to operate for forty years (coincidentally, the same as the initial licence period). There is no internationally agreed harmonised plant life period.

3. NUCLEAR POWER IN THE WORLD

At the end of 1998 there were 434 plants [3] having a capacity of 348, 864 MW(e) in the world (Figure 3). The commonest type was the Pressurised Water Reactor (PWR) and the second most popular was the Boiling Water Reactor (BWR). The Pressurised Heavy Water Reactor (PHWR) was second to the PWR in NPP being constructed.

With the decrease in the construction rate of NPP over the past 15 years in the world there has been an increase in the number of older plants in service (Figure 4). But there aren't many really old NPP in service. The average age of NPP is about 15 years but the age distribution is 'skewed' because of the rapid decrease in the introduction of new plants in the past decade or so.

'Life Extension' of existing NPP increases the nuclear capacity (see for example, Figure 5) [4], generates more electricity and provides a better return on the capital invested. In Japan the approach seems to be to identify the 'life' from a technical base and to eventually operate the NPP on this basis. After comprehensive study and the development of a plant life strategy, a target 'life' for NPP of sixty years seems practicable.



Reactors in Operation and Net Electrical Power (as of Dec. 1998)

Reactors Under Construction and Net Electrical Power (as of Dec. 1998)

FIG. 3. Nuclear power capacity [3].

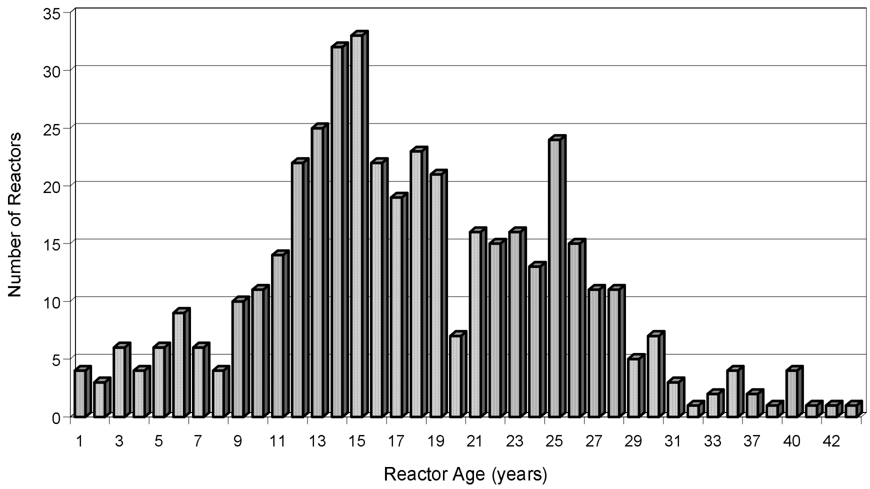


FIG. 4. Number of reactors in operation by age at the end of 1998 [3].

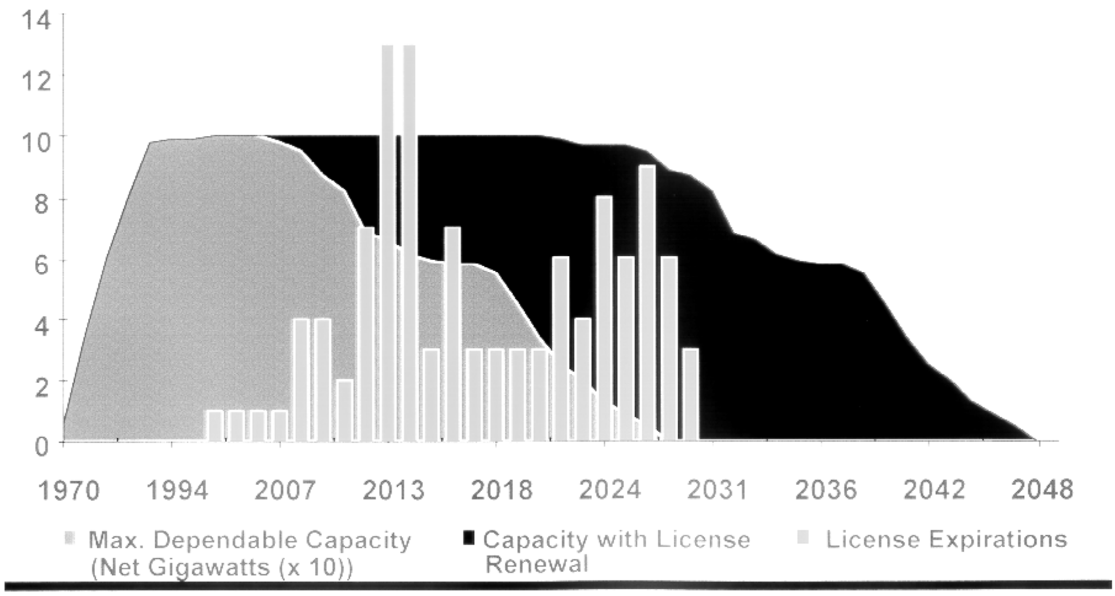


FIG. 5. US generating capacity with license expirations assuming construction period recapture.

4. PLANT LIFE MANAGEMENT PROCESSES

The total lifetime cost of a NPP includes the choice of the NPP, its construction and other pre-operational costs, the fuel cycle costs, the system requirements and other factors. These aspects are not discussed here but for a particular country it will be realised that the decision will include political, financial and other considerations. However the overall process described below includes examples of recommended practice and guidance for the establishment and implementation of plant life management programmes.

In the operation of a Plant Life Management Strategy it will be remembered that certain inspection, surveillance, engineering, maintenance and other activities will also need to be carried out on the NPP whether a PLIM programme is in place or not. These activities are usually considered as being distinct from PLIM. However it has been realised that it is the same personnel who discharge the bulk of both activities at the NPP. There are advantages in considering 'normal' and PLIM activities under the same classification of work. Organisational arrangements involving 'central' staff are necessary for considering the totality of these activities in an Utility-and also the generic impact on NPP operation and planning on other plants of a similar type.

After recovering the capital charges, the electricity generating costs component consists of operation, maintenance and fuel costs. Of course, provision for the post-operational costs will also need to be made. A recent illustration [5] of generating costs of nuclear coal and gas costs (Figures 6 and 7) and shows the competitiveness of nuclear power even with the inclusion of some large refurbishment costs.

An outline of the NPPLM processes are shown in Figure 9. There is no prescribed limit on the operational life. Age related degradation mechanisms could cause deterioration in component properties. It is therefore necessary to have a high level of understanding of ageing to ensure that plant safety and reliability are maintained as components age. All components 'age', but it is the rate of degradation and its significance that determines its importance with regard to failure. The main degradation mechanisms include metallurgical phenomena such as irradiation embrittlement, fatigue, corrosion, interaction of mechanisms and so on (see Figure 8 for stressors, ageing mechanisms and consequences). The details of the degradation mechanisms for key components are not given here.

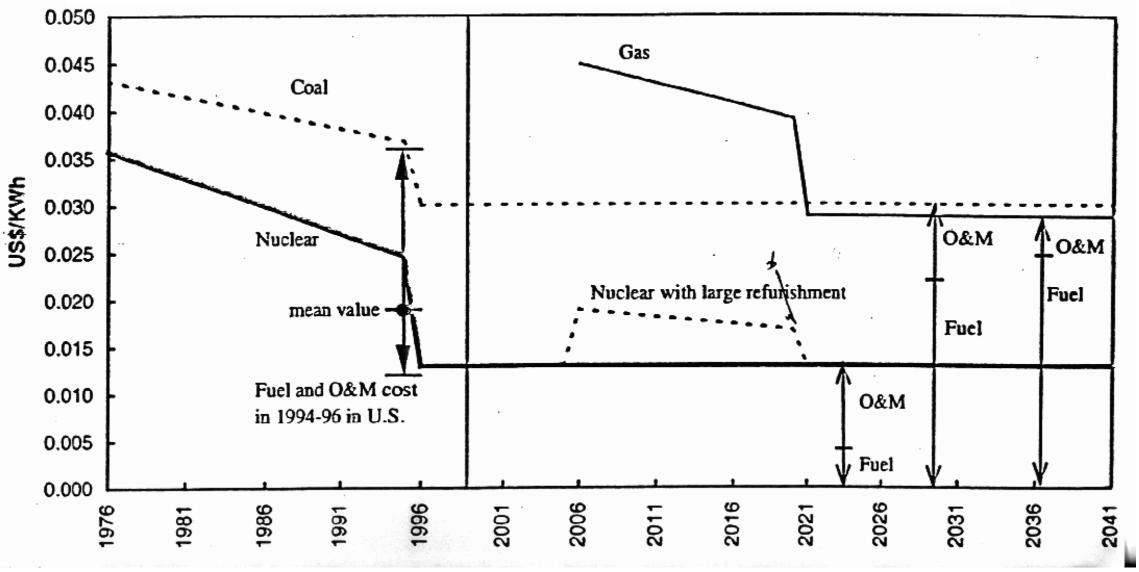


FIG. 6. Generating costs of nuclear and coal power established in 1976 and gas power in 2006 [5].

	Nuclear	Coal	Gas
Investment at 1976 (a)	1 540 US$/kWe	880 US$/kWe	
Investment at 2006 (b)			880 US$/kWe
Amortisation period	20 years	20 years	15 years
Refurbishment at 2006 (c)	300 US$/kWe		
Interest rate	5%	5%	5%
Load Factor	75%	75%	75%
Thermal Efficiency	34%	34%	52%
O & M cost (b, d)	58.6 US$/kWe/year	52.5 US$/kWe/year	27.0 US$/kWe/year
Coal & Gas cost (b)		2.09 US$/Gjoule	3.58 US$/Gjoule
Uranium cost (b)	50.2 US$/kg		
Enrichment cost (b)	103.8 US$/SWU		
Fabrication cost (b)	310.5 US$/kg		

(a) Assumed the average in OECD countries in 1976 is same in 1981(OECD 1983).
(b) Average in OECD countries(OECD 1998b).
(c) Assumed large refurbishment costs 300 US$/kWe (NDC 1999) and amortisation period is 15 years.
(d) The data of power plants expected to be commercially available by 2005-2010 (OECD 1998b).

FIG. 7. Assumptions adopted in the generating cost calculation in Fig. 6 [5].

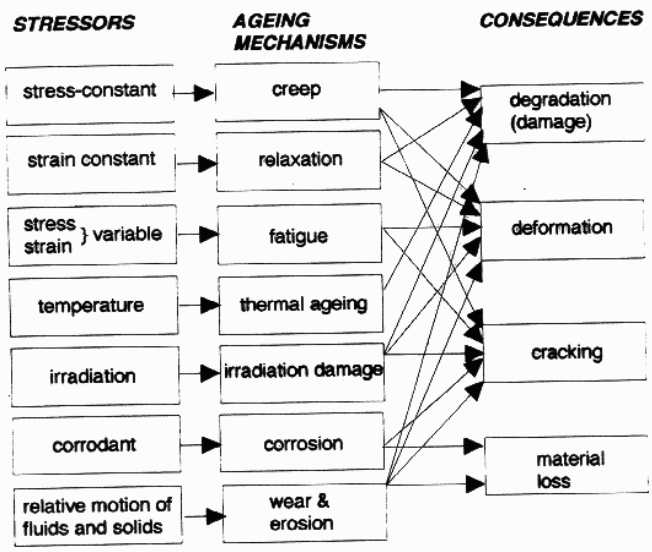


FIG. 8. Lifetime-ageing factors, basic ageing mechanisms and possible consequences.

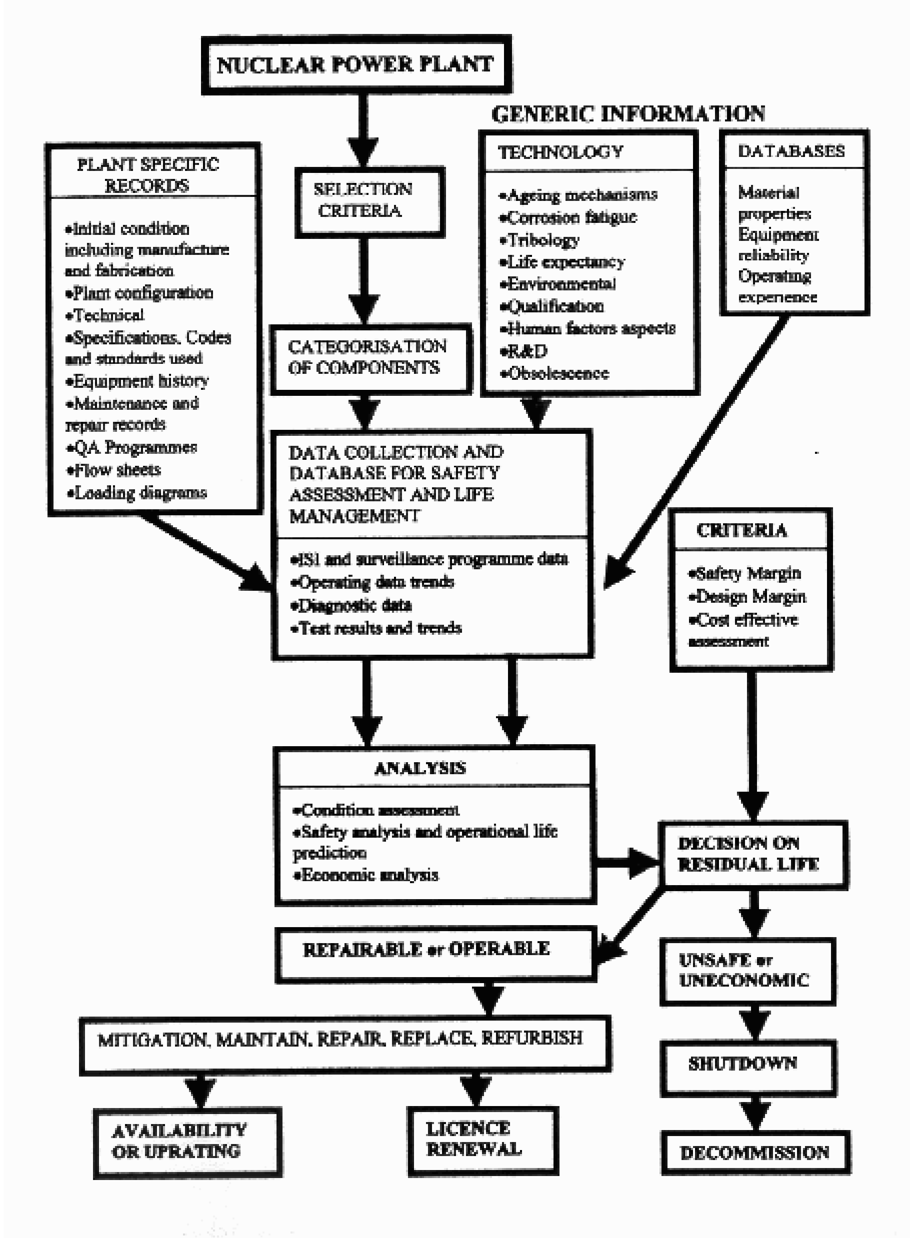


FIG. 9. NPP lifetime management processes.

The important features in the process of Plant Life Management are:

(i) the selection of key plant components is made using some prioritisation principles.

(ii) the remaining life of each component is determined using the available data from the original design documentation and the relevant Codes and Standards, relevant degradation mechanisms, degradation data, operational and maintenance history data and the present state derived from inspection, surveillance, condition monitoring data and relevant records.

(iii) the estimated remaining life is compared with the target extended life of the plant.

As a result of this evaluation three possible courses of action are available:

(i) if the estimated remaining life is greater than the target plant life then action is needed by the Utility.

(ii) if the estimated remaining life is close to the target NPP life then measures for mitigating the effects of ageing degradation would be required. New actions in the areas of preventative maintenance, improvements to operational procedures, record keeping and R&D might become necessary.

(iii) if the predicted remaining component life is less than the target plant life, measures for slowing down ageing and for restoring reduced component performance would have to be initiated. These measures could include increased/enhanced inspection and maintenance, repair or replacement. The schedule of the repair, or large-scale replacement initiatives must be based on an evaluation of safety, economics, reliability and other factors. For some plants such an evaluation could lead to the decision not to continue operation of the NPP.

Of course, these courses of action are not mutually exclusive. A mixture of possibilities exists, for example, some complex components may have sub-components to which different alternatives apply.

5. KEY COMPONENT SELECTION CRITERIA. IDENTIFICATION AND CATEGORISATION

Each NPP has thousands of components and to evaluate each of these in term of its life would be a daunting task. Therefore it is desirable to categorise or 'rank' these components in terms of their importance in order to prioritise the work and to maximise the effective use of resources. The first step is to identify those key components which would be important if their failure had a major impact on Safety or Plant Operational Life. In considering the economic assessment the key components are those whose repair or replacement would cause a major addition to the maintenace budget or an abnormally long shutdown such as to adversely affect the cost of generating electricity from that plant. The second step is to classify the components into four categories on the basis of the economic criteria. Then a prioritisation of items is attempted in order to rationalise, optimise and identify resources. There can be a large number of factors (Figure 10) to categorise components and differing weighting will be given inside a particular Utility in a particular country:

- Cost to replace/refurbish
- Impact on plant availability
- Loss of revenue (economic impact
- Radiation dose
- Regulatory importance
- Modifications required
- Leading plant influence
- Replacement precedent
- Generic applicability (User Group Influence)
- Mode of failure
- Consequences of modification on plant safety
- Implication for overall plant safety

FIG. 10. Examples of "categorisation" factors.

It follows that there may be special site features which could produce peculiarities in these lists-for example 'wet'/ 'dry' sites in evaluating concrete, climatic or seismic factors may also be enhanced for specific sites. National lists would possibly reflect the Utility's operation experience and maintenance practices.

The categorisation of components in the order of priority falls into four main areas (Figure 11):

•**CATEGORY 1 COMPONENTS** are those which are generally considered to be irreplaceable. Examples of such components include the Reactor Pressure Vessel (RPV)and also the Containment Structure (with regard to replaceability it can be argued that even the RPV and the containment structure could be replaced-but at a great cost. Also with regard to irradiation embrittlement of RPVs in older NPP it is noted that more than fifteen WWER RPVs have been annealed. For the larger PWR vessels there is not an immediate need for the mitigating action of annealing. So operational life limiting degradation is not a problem area.

•**CATEGORY 2 COMPONENTS** are those which are replaceable, but are costly in terms of capital expenditure and outage time requirements. An example of this type of components would be steam generators which have been replaced on many plants.

•**CATEGORY 3 COMPONENTS** are those which are 'key' in terms of plant safety and reliability and are susceptible to ageing, but which are replaceable on a routine basis.

•**CATEGORY 4 COMPONENTS** are all other components not included in the earlier categories and are not related to 'life' considerations.

FIG. 11. The four categories of components.

Many Utility and National studies have been carried out on the identification and prioritisation of key components of different NPP. Two examples are given below.

The first example (Figure 12) is for eleven key components of the US PWR.

- Fuel channels
- Steam generators including their internals
- Calandria vessel
- Reactor headers
- PHT piping pressuriser
- General nuclear piping
- Calandria supports
- Secondary piping
- building
- Calandria vault and end shield c.system
- Cables (power, control and instrumentation
- Reactor building
- Turbines
- Generator
- CW intake structure
- Spent fuel bay/liner

Sixteen key components for CANDU

- Reactor Pressure vessel
- RPV Internals
- Reactor coolant pressure boundary piping
- RPV safe ends
- CRD housings and guide tubes
- Drywell metal shell
- Suppression chamber and vent system
- Reactor vessel support
- Concrete structures; RPV pedestal, drywell foundation, biological shield, fuel pool slabs and walls, reactor building basemat sacrificial shield wall, reactor building floor slabs and walls and turbine pedestal
- Plant control centre
- Emergency diesel geerator

Eleven Key components for the US PWR.

FIG. 12. NPP lifetime management processes — Key components.

The second example is from JAPEIC where the study included both the PWR and BWR. (see Figure 13) [6].

The full evaluation then usually follows the scheme outlined above.

BWR (7 CSs)

- **Reactor Vessel**
- **Reactor Internals**
- **Main Coolant Piping**
- **Primary Loop Recirculation Pump**
- **Primary Containment Vessel**
- **Cable**
- **Concrete Structure**

PWR (9 CSs)

- **Reactor Vessel**
- **Reactor Internals**
- **Main Coolant Piping**
- **Reactor Coolant Pump**
- **Pressurizer**
- **Steam Generator**
- **Containment Vessel**
- **Cable**
- **Concrete Structure**

FIG. 13. Major components and structures [6].

6. DATA AND RECORDS

Data and accurate records are crucial for NNPLM evaluation and to establish safety margins. The information required for these assessments needs to be established at the earliest stage. Not only is there a requirement for records but there is a need for representative archive material for possible future testing. Utilities will store data on key components. Even if it is assumed that the Utilities in particular countries have the ability to provide appropriate data on key component performance they will also need to provide data on component repair and replacement (even if the original supplier has moved on), associated hardware and software, operational changes and events etc.. In order to achieve a better understanding of ageing phenomena and the impact on NPPLM, exchanges of information on relevant topics, which could be organised under the framework of international organisations, Users Groups and Specialist meetings like the Bi-annual PLIM/PLEX meetings are particularly important.

'Proper' record keeping requires the identification of specific data. Only then can the impact of ageing on the availability and reliability of components be followed and evaluated. The data set represents a general approach to the assessment and evaluation of ageing in components and may vary with regard to the Categorisation of the specific component. The data set could be based on information which is necessary for detecting and following faults or degradation. The data set, in general terms, should include the groups of data given in Figure 14. A 'proper' format for record keeping needs to be established at the outset of a NPP project.

1 **Component specification Data (baseline data)**
- Material (manufacturing details, archive material)
- Initial properties
- Codes and Standards
- Design loading
- Anticipated ageing mechanisms (ageing sensitivity, methods of assessment)

2 **Age or Failure tracking data**
- Operational history
- In-service inspection
- In-service monitoring

3 **Stressor and root cause data**

4 **Failure or degradation mechanism data**

5 **Test and maintenance data**

6 **Measures to improve material choice, design and operations**

FIG. 14. Scope of data set requirements.

In the component description the influence of Codes, Standards and Regulations and the underlying rationale for the choice of materials, design, fabrication, inspection and operational validation testing of the system will be recorded. Deviations from those in current use will need to be identified. Qualification data including materials tests, pre-operational testing, pre-service 'fingerprint' inspections in-service inspections and repair information will form part of the data set. The design basis will provide the functional criteria, design assumptions about the operational requirements, structural integrity assessments and material condition. The operating conditions and the maintenance history will provide the basis to assess degradation. Owners should collect and maintain records of all significant testing and plant transients to evaluate Pressurised Thermal Shock events and also fatigue processes.

The process of Plant Life Management will continue to require data in the areas of:

- Materials data
- Development of models for component residual life prediction
- Component repair, and
- Investigations in the field of feasibility of component replacement

Category 3 components generally include those that are part of the NPP Safety related systems. Programmes for their ageing management usually address the potential risk of failure from ageing degradation. Because these are components that can be replaced their economic impact are smaller than Category 1 or two components. The main questions to be addressed are to do with the timing and before there is a loss of functional capability.

The full evaluation of components for operational life assessment is a large task requiring the application of a large resource of manpower and data. However, the potential benefits in terms of plant life assurance and in determining the operational life of the NPP s great. Reducing the cost of generating electricity from nuclear power is the target.

7. INTERNATIONAL ATOMIC ENERGY AGENCY ACTIVITIES

The IAEA programmes on NPP Life Management activities are implemented in the Division of Nuclear Power (NENP) and Safety Aspects in the Division of Nuclear Installation Safety.

IAEA Publications on 'good practice and guidance' are available in this area of Plant Life Management (for example see [7–11]).

A large scope of activities are pursued under the aegis of the International Working Group on NPP Life Management, in the Division of Nuclear Power Division. Specialist meetings are held, publications are commissioned in specific areas, International Co-ordinated Research Programmes are conducted to harmonise technology, techniques and to pursue 'hot topics' , the development of a large scope multi-module data base is being pursued. The work of the IWG has been described most recently [12] and copies of this will be made available at the TCM at Kashawazaki. A guidance document on Plant Life Management is being prepared at the present time [13].

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THE BOHUNICE NPP V-1 UNITS NUCLEAR SAFETY UPGRADING

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Bohunice NPP, Slovak Republic

Abstract

Safety upgrading and operational reliability improvement was carried out by the Bohunice NPP V-1 staff continuously since the plant commissioning. By now, more than 1200 minor or major modifications have been implemented, either by the NPP maintenance staff or by the contractors. Based on findings of safety assessment missions invited by Bohunice NPP in 1990 - 1991, the Czecho-slovak Nuclear Regulatory Authority (CSKAE) issued the decision No5/91 of 81 safety upgrading measures to be taken in different areas. These improvements are referred to as the "**Small Reconstruction** of the Bohunice V-1 NPP". Realization of measures during Small reconstruction of the Bohunice NPP V-1 became a power plant, which further operation is acceptable from safety point of view, but it is also necessary further safety improvement. During the period of the Small Reconstruction the development of a Safety Report for the Gradual reconstruction has been completed. Based on this report the SR Nuclear Regulatory Authority issued the Decision No.1/94, in which requires 59 upgrading measures in different areas to be addressed. The development of Basic Engineering of the **Gradual Reconstruction** has been contracted to the Siemens AG. Implementation of safety measures are provided through contract with the consortium REKON (which consists of Siemens AG company and Nuclear Power Plants Research Institute Trnava) and other Czech, Russian and Slovak companies. The Gradual Reconstruction of Bohunice NPP V-1 will be finished in 2000. By implementation of the measures carried out during Gradual Reconstruction achievement of an internationally acceptable nuclear safety level will be reached.

1. Introduction

1.1. Units V-1 at Bohunice site consist of two units of VVER-440 type V-230 series. Unit 1 was put into operation at the end of 1978 and the second one at the beginning of 1980. The original Russian (Soviet) design of units VVER-440/230 type originates from the end of sixties and the beginning of seventies. Rupture of primary coolant circuit with equivalent diameter of 32 mm was stated as maximal design basic accident, for managing of which capacity of safety systems was designed in compliance with those days valid standards.

1.2. The design of NPP V-1 units, which are youngest ones of V-230 series, took into account experience and knowledge from previous units, what reflected in higher safety level and operational reliability in comparison with other units of VVER-440/V-230 series. Among the most important improvements of V-1 units design there are:

- supplementing of steam generator super emergency feedwater system
- higher capacity of emergency core cooling system
- supplementing of automatic links between primary and secondary circuit systems
- higher level of secondary system automation.

2. Small reconstruction of Bohunice NPP V-1 units

2.1. Process of safety improvement and operational reliability of NPP V-1 units began immediately after commissioning. There were realized more than 1200 design modifications of bigger or smaller scope, which resulted from operation status evaluation, operational experience and from various international recommendations and regulations. The modifications were realized by own workers and also in contractual way. Among the most important arrangements belong:

- supplementing of shielding assemblies into the core to decrease neutron flux on reactor pressure vessel, to reduce of reactor pressure vessel brittle fracture risk
- supplementing the emergency power supply of the most important consumers of emergency systems
- reconstruction and supplementing of operation computer systems and supplementing the monitoring of parameters after failures
- supplementing the diagnostic systems for monitoring the status of power plant main components
- reconstruction and replacement of lower reliable I&C components supplementing of more automatic functions for improvement of safety and operational reliability.

2.2. Further more significant step in safety improvement of NPP V-1 units was made after 1990 as a result of several national and international expert missions. Expert missions were focused on evaluation of NPP V-1 safety status level and operational reliability. Short term and long term measures were the results of evaluation of expert missions which were realized, while no one mission approached the conclusion that it necessary to shut down the NPP V-1 units.

2.3. Based recommendations of individual expert missions the Czechoslovak Nuclear Regulatory Authority issued the decision No. 5/91 dated January 11, 1991. There were defined 81 measures concerning further safety and reliability improvement of NPP V-1 units and decision No. 213/92 dated June 23, 1992. There were stated another 14 measures. These measures were realized during the period 1991 – 1993 and they are known as Bohunice NPP V-1 "Small Reconstruction". Significant operational safety and reliability improvement and fixing the dominant position of NPP V-1 among the V-230 units became the reality mainly in following fields:

(a) Decreasing the probability of serious accident to one half (decreasing the probability of core damage) from the 1.7×10^{-3} per year to 8.89×10^{-4} per year based on elaborated probability safety analyses.

(b) Significant decreasing (15x) of core damage risk caused by possible fire (NPP V-1 fire protection status improvement, which decreased from original 30% probability of core damage contribution to 2%).

(c) Demonstration of extremely low probability of reactor coolant system piping rupture (10^{-6} per year) by using the method "leak before break" on primary circuit, which confirmed the integrity and reliability of systems and main components of primary circuit.

(d) Significant increasing of safety and reliability by modifications of reactor protections, increasing the power of emergency power supply and separation of redundant power supply systems and safety related control system of V-1 units. There were provided two independent AC and DC power supply systems for each unit, supplementing the new diesel generator (No.5) and accumulator batteries, what enabled to manage to leak from primary circuit to rupture with equivalent diameter 100 mm, what is more in comparison with original design basic accident (diameter 32 mm).

(e) Increasing the tightness of confinement of V-1 units, which purpose is to contain radioactive materials released at hypothetical accident caused by reactor coolant circuit rupture, more than 25 times in comparison with the status before small reconstruction.

(f) Build up emergency control room for emergency reactor shut down, checking and control of safety related systems and monitoring main unit parameters in case of accident and impossibility to use the main control room.

(g) The seismic improvement of civil structures and other technological equipment.

(h) To make possible further operation of reactor pressure vessels by performing the annealing on both units of NPP V-1 in 1993. The conclusions demonstrated, that regarding brittle fracture and fatigue damage it is possible to operate reactor pressure vessels till the end of their designed lifetime. Further annealing is possible, what is demonstrated by annealing of NPP V-1 units in 1993 and of Loviisa (Finland) in 1996.

(i) Supplementing diagnostic systems for monitoring the immediate status of safety related systems and components. The scope of NPP V-1 diagnostic systems is larger than in western units and offers more information on equipment status.

(j) Providing over-standard In-service Inspections on NPP V-1 main equipment with aim to ensure their actual status and safe operation. (Some of these are performed by SE a.s. employees together with prestigious companies like e.g. Siemens – Germany, RTD – Netherlands, Skoda – Czech republic and VUJE – Slovakia for other NPP operators – e.g. NPP Loviisa – Finland, NPP Paks – Hungary, NPP Borselle – Netherlands and NPP Dukovany – Czech Republic).

Decreasing the probability of important safety related system malfunction 5 to 62 times according to following table:

Improvement	Probability of malfunction per year		Improvement coefficient
	before small reconstruction	after small reconstruction	
Emergency reactor shutdown caused by low pressure	1.92×10^{-2}	3.1×10^{-4}	62 ×
Emergency reactor shutdown caused by loss of feed water	9.1×10^{-3}	1.8×10^{-4}	51 ×
Emergency reactor shutdown caused by opening pressurizer safety valves	1×10^{-3}	2.2 × 10-5	47 ×
Loss of emergency power supply of the 2nd category	2.4×10^{-2}	1.24×10^{-3}	19 ×
Malfunction of emergency feedwater system	3.2×10^{-1}	7.44×10^{-2}	5 ×

Realization of measures during SMALL RECONSTRUCTION NPP V-1 became (from point of view of core damage probability) a power plant, which further operation is acceptable from safety point of view, but it is also necessary further safety improvement.

3. Graduall reconstruction of Bohunice NPP V-1 units

3.1. Preparation of NPP V-1 gradual reconstruction.

Regarding decision No.5/91 of Czecho-Slovak Nuclear Regulatory Authority on possibility to operate NPP V-1 after 1995 under condition, that its nuclear safety will be increased to European standards, the preparation of "Principal reconstruction" started with working teams of EBO, VUJE, Skoda and EGP since 1991. Result of mentioned above is issuing the "Safety report for NPP V-1 principal reconstruction" dated June 1992, which was re-elaborated in February 1993 to the form of "Safety report for NPP V-1 gradual reconstruction" and which was approved by Slovak Nuclear Regulatory Authority decision No. 1/94 dated February 24, 1994. Slovak Nuclear Regulatory Authority decision

No. 110/94 dated August 25, 1994 elaborated the original Czecho-Slovak Nuclear Regulatory Authority decision No. 5/91 in the field of NPP V-1 operation conditions after 1995 and has changed the original license for NPP V-1 operation till the design lifetime to approval for operation only for the next year under the condition of positive evaluation of works performed last year and refuelling and approved scope of reconstruction works for next year refuelling. Slovak Nuclear Regulatory Authority decision No. 1/94 supplemented and made more precise the safety report for NPP V-1 gradual reconstruction in 59 measures (37 measures for analyses and evaluations and 22 measures for realization of reconstruction works). This decision of Slovak Nuclear Regulatory Authority measures No. 1/94 concern following safety related functions:

1. Primary circuit integrity
2. Core cooling during operation and in case of accident
3. Core cooling in case of coolant leakage from primary circuit
4. Confinement
5. Auxiliary systems
5.1. Service water
5.2. Power supply
5.3. I&C
5.4. Fire protection
5.5. Seismic improvements

Further organisational measures were stated by Slovak Nuclear Regulatory Authority decision No.110/94.

3.2. Elaborating BASIC ENGINEERING for NPP V-1 gradual reconstruction.

Regarding importance and scope of NPP V-1 reconstruction works there were selected several foreign companies for submitting their tenders. The Westinghouse and Siemens submitted satisfactory tenders at the beginning of 1993 as offers for NPP V-1 principal reconstruction, which in December 1993 were re-elaborated to the form of offers for NPP V-1 gradual reconstruction.

Selection of tenders finished in the 1st quarter of 1994 for the benefit of Siemens KWU, and on May 5, 1994 there was signed the contract for Basic design for NPP V-1 gradual reconstruction.

Implementation of this NPP V-1 gradual reconstruction is focused on fulfilling following probabilistic and deterministic targets:

- Managing the newly defined design basic accident (coolant leakage through diameter 200 mm) by conservative approach and beyond design basic accident (coolant leakage through diameter 500 mm) by best estimated methods.
- Confinement tightness and localising systems must ensure, that in case of coolant leakage through 200 mm the dose equivalents will not be exceeded 50 mSv for whole body and 500 mSv for thyroid in monitored power plant area and in case of coolant leakage through diameter 500 mm by best estimated methods 250 mSv on whole body and 1,500 mSv for thyroid.
- Safety related systems must fulfil the requirement on reliability with malfunction probability 10^{-3} per demand or less.
- Safety related systems will provide, that probability of serious core damage is 10^{-4} for reactor per year or better.
- Reliability of reactor trip system is as least 10^{-5} per demand
- Completion of seismic resistance of all safety related systems and equipment of the unit and corresponding buildings and systems to 8° MSK-64 (250 cms^{-2} horizontally and 130 cms^{-2} vertically).

– Separating of redundant trains of safety systems and supporting systems.

By realization of the measures mentioned above is assumed achievement of such NPP V-1 nuclear safety level, which will be internationally acceptable and will be in compliance with international standards. Nuclear safety level assessment is in competence of Slovak Nuclear Regulatory Authority, which on the basis of its status allows further operation of NPP V-1 units.

Elaborating the BASIC ENGINEERING for NPP V-1 gradual reconstruction was provided mainly by the company Siemens KWU in co-operation with Slovak companies like VUJE Trnava, VUEZ Tlmace, PPA Bratislava, EZ Bratislava and other.

BASIC ENGINEERING solved reconstruction of following fields, systems and subsystems:

– Primary circuit integrity improvement.
– ECCS reconstruction.
– Reconstruction of safety and relief valves (from pressurizer to relief tank).
– Confinement strength and tightness improvement including supplementing isolating valves and flaps on HVAC pipings penetrating trough the confinement boundary.
– Supplementing accident localisation system in confinement.
– Supplementing of venting system.
– Confinement spray system reconstruction.
– Building up the new system of service water to ensure cooling of safety important consumers.
– Reconstruction of electrical systems:
  - emergency power supply 6kV, 0.4kV, 220V DC, 220V AC, 24V DC - reconstruction of distributions and switchgears.
  - exchange of motor generators.
  - reconstruction of DG control system
  - adding the third electric source to supply home electric consumption from the near hydroplant Madunice (in case of blackout).
  - providing of two independent power supply systems of safety related consumers.
– I&C reconstruction:
  - RTS and ESFAS reconstruction including reactor power control and limitation systems.
  - neutron flux measurement system reconstruction.
  - reconstruction and supplementing of post accidental monitoring system (PAMS).
  - completion of emergency control room.
  - ensuring of two independent systems for checking and control of safety related consumers.
– Supplementing SG additional steam generator emergency feedwater system.
– Supplementing steam dumping valves to atmosphere.
– Fire protection improvement in existing areas.
– Reconstruction of HVAC systems in areas where there are located important safety related equipment.
– Building modifications in existing areas and building up of new buildings for:
  - steam generator emergency feedwater system
  - essential service water system
  - ventilation systems for cooling of rooms with electric and I&C systems.

There were also solved following items in the frame of contract for BASIC ENGINEERING:

– preliminary safety analyses report (PSAR).
– equipment labelling (identification).
– creating the set of standards, regulations and decrees, which will followed during NPP V-1 units gradual reconstruction.
– equipment seismic resistance
– re-qualification of existing NPP V-1 equipment.

Basic Engineering was finished in November 1996. The results assumed by elaborators were assessed and commented by EBO and independent organizations and finally they were submitted to Slovak Nuclear Regulatory Authority for approval.

During Basic Engineering elaborating the activities of suppliers and customer were co-ordinated in 15 expert working teams, which discussed and solved the problems on common meetings at least once per month or according to necessity.

Once per month there were organised two day CO-ORDINATION MEETINGS (STATUS MEETINGS) of design team representatives on behalf of Siemens and Bhunice NPP, where there was evaluated work status on Basic Engineering, they solved problems arising from working teams and made more precise the overall program. The work status was evaluated in monthly reports (Progress Reports).

3.3. Elaborating the DETAIL DESIGN and REALIZATION.

Based on approved results from BASIC ENGINEERING and PSAR the documentation for each individual system is completed up to level of realization documentation and consequently according to this provided realization of proposed and approved reconstruction works.

Elaborating of realization documentation, equipment production, realization and putting into operation for whole gradual reconstruction is provided by MAIN SUPPLIER - consortium REKON consisting of SIEMENS KWU and VUJE Trnava , according the contract, that was signed in April 1996.

Realization of modifications and reconstructions of individual systems will be performed gradually in scheduled unit refuelling outages. Unit refuellings will be extended according to the necessity.

Assumed deadlines for realization of NPP V-1 gradual reconstruction are from the end of 1996 and last modifications are supposed during refuelling on unit 1 in 2000.

Full scope of the Bohunice V-1 NPP gradual reconstruction is divided into sixteen technological systems as follows:

(1) Reconstruction of the **SAFETY AND RELIEF VALVES** (from pressuriser to relief tank) for Bleed & Feed Procedure.

(2) Adding another **SUPEREMERGENCY FEEDWATER SYSTEM** to steam generators.

(3) Adding the **STEAM RELIEF VALVES TO EACH STEAM GENERATORS** (steam from SG to the atmosphere).

(4) Adding the **THIRD ELECRICAL SOURCE TO SUPPLY SELF ELECTRIC CONSUMPTION** from the nearby hydroplant Madunice (in case of blackout).

(5) Reconstruction of **EMERGENCY CORE COOLING SYSTEM** into the two separated and independent redundancies with increased capacity for new defined DBA.

(6) Reconstruction and improvement of **FIRE PROTECTION** (14 measures).

(7) Reconstruction of **ELECTROTECHNIC SYSTEMS** to have two independent and separated redundancies:
- exchange of the obsolete motor generators
- reconstruction of diesel generators control systems
- reconstruction of 6kV, 0.4kV, 220 DC + AC switchgears

(8) Reconstruction of **I&C** SYSTEMS:
- exchange RTS + ESFAS.
- exchange Reactor Control Power System.
- exchange Ex-core Neutron Flux Measures.
- exchange In-core Temperature Control Measures.
- adding Reactor Power Limitation System.
- adding PAMS (Post Accident Monitoring System) - more parameters.
- adding Radiological Monitoring System.

(9) Reconstruction of **CONFINEMENT SPRAY SYSTEM** into the two separated and independent redundancies.

(10) Adding the **PRESSURE SUPRESSION SYSTEM** after LOCA inside confinement.

(11) Assure **CONFINEMENT TIGHTNESS** by adding isolation valves into ventilation pipes crossing trough confinement boundary.

(12) Increasing **CONFINEMENT STRENGHT** to assure confinement resistance against maximum over and under pressure
+100/120 kPa over pressure DBA/BDBA
- 20 kPa under pressure DBA/BDBA

(13) Reconstruction of **SERVICE WATER SYSTEM** for cooling safety systems — to separate and create essential service water system.

(14) Adding and reconstruction of **VENTILATION SYSTEMS** for cooling of rooms where are placed new I&C and Electrical systems.

(15) **SEISMIC IMPROVEMENT** a lot of equipment that can have negative influence interaction to safety systems.

(16) Implementation of **ACCIDENT LOCALIZATION SYSTEM** to decrease pressure in confinement after DBA/BDBA.

Following systems and subsystems were finished or partially implemented within the scope of refuelling outages till now (unit 1 in 1997, 98 and partially in 1999, unit 2 in 1996, 97,98 and 1999 - actually during current outage with refueling from September 18 to December 16).

(1) Pressuriser safety valves - finished at the unit 1 and 2.

(2) Steam generator super emergency feeding - finished at the unit 1 and 2.

(3) Stations releasing steam to the atmosphere - finished at the unit 1 and 2.

(4) Electric supply from the hydroelectric power station Madunice - finished at the unit 1 also at the unit 2.

(5) ECCS – finished at unit 2. New equipment ECCS on unit 1 will be realized in 2000.

(6) Fire protection upgrading — it has been carried out during each outage and it will be finished at the unit 1 in 1999 and at the unit 2 in 2000.

(7) - Motor generators - both systems were changed at the unit 1 and 2.
- Diesel generator control system — finished in unit 2. 1st stage finished at the unit 1.
- 6 kV switchgears - finished at the unit 2. For unit 1 it will be finished in 2000.
- 0.4 kV and 220 V switchgears finished at the unit 2. At unit 1 will be finished in 2000.

(8) - RTS, ESFAS (Engineered Safeguards Actuation System), ex-core neutron flux measurement, reactor power control system, reactor power limitation system — 1st stage is finished on unit 1. It includes realization of systems and connection input signals, but the consumer control is not connected. System is operating under testing mode (open loop), that means in parallel with existing systems. After its positively evaluation it will finally replace the existing system in next outage (2000). Unit 2 systems were put into active operation in 1998.
- PAMS - finished at unit 2. Implementation at unit 1 will be finished in 2000.
- Temperature in-core measurements and radiological system were implemented at Unit 1 and 2.

(9) Confinement spray system — finished at unit 2. Reconstruction at unit 1 will be finished in 2000.

(10) Pressure suppression system inside confinement - 1st stage finished at unit 2. Implementation will be finished in 1999 (during current outage). At unit 1 will be finished in 2000.

(11) Confinement tightness - finished at unit 2. Reconstruction at unit 1 will be finished in 2000.

(12) Confinement strength - finished at unit 2. Reconstruction at unit 1 will be finished in 2000.

(13) Service water system - temporary solutions are finished. Installation of all equipments for both units will be finished in 2000.

(14) Ventilation systems - reconstruction at unit 1 and 2 was finished. Other improvements will be finished in 2000.

(15) Seismic resistance upgrading - partially implemented at unit 1 and 2. Completing at unit 1 and 2 will be finished in 2000.

(16) Accident localization system - partially implemented at unit 2 confinement. Completing of design change at unit 2 will be finished in 1999 and at unit 1 in 2000.

4. Conclusion

Carrying out the gradual reconstruction we assume to reach the internationally acceptable level of nuclear safety and operational reliability of the V-1 plant. The designed lifetime for both units V-1 is 30 years. The originally planed decommissioning of unit 1 is 2008 and for unit 2 is 2010. In autumn 1999 the Slovak government released the decision about decommissioning NPP V-1 units. According to this decision unit 1 will be decommissioned in 2006 and unit 2 in 2008.

PROGRAMMES FOR ACHIEVING
HIGH PERFORMANCE AND RELIABILITY

(Session I-b)

Chairperson

L. MYRDDIN DAVIES
United Kingdom

YEAR 2000 READINESS FOR TEPCO'S NUCLEAR POWER PLANTS

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Abstract

In line with our policy of positioning the Year 2000 (Y2K) problem as a major management task, we have performed Y2K readiness. We started to survey the influence of Y2K problem at Nuclear Power Plants (NPP) on February, 1996, and completed Y2K ready of NPP on October, 1999. This paper presents Y2K readiness of our NPP. Our NPP instrumentation and control can be roughly divided into two types (Monitoring System showing NPP's status, Control System to control equipment of NPP). We surveyed hardware and software of these systems to assess the influence of these systems by Y2K problem. For survey of hardware, we picked up all chips of Real Time Clock (RTC) with check of the lists of all parts on board. And we surveyed integrity to Y2K problem of picked up RTCs with their specification, instruction manual and so on. For survey of software, we picked up the system using a time parameter in software with check of system's source program. And we surveyed whether the system using a time parameter had functions that were influenced by Y2K problem. As the result of the survey, for both monitoring systems and control systems, there was no RTC chip having Y2K problem. And there was no control system with software modification required. It was confirmed that some of Monitoring Systems were required software modification. The modification of these systems was completed by October, 1999. We performed Simulated Test of Y2K to validate the result of survey and software modification. Simulated Test was performed for the confirmation of the system integrity when the system was input critical dates of Y2K problem. As the result of Simulated Test, there was no system that was confirmed the failure of survey and software modification. In addition to these Y2K readiness, referring to the existing contingency plan for emergency situations such as a system failure and a natural disaster like earthquakes and typhoons, we have completed contingency planning for Y2K. It is decided in consideration of internal risks (Trouble of Monitoring system for Y2K problem) and external risks(A sudden drop on electricity demand). We are going to perform the following items that based the contingency plan from December 31,1999 to January 1, 2000.

- Confirmation of system integrity by check of plant parameters;
- Additional operator support staff and maintenance staff will be posted to confirm system integrity;
- Make system of communication between our and manufacturers engineers for emergency of Y2K problem.

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1. INTRODUCTION

In view of the social responsibility for providing a steady supply of electric power, an essential item of infrastructure, we consider the Year 2000 (Y2K) problem to be a major management task and are systematically making the following systems ready for the Y2K problem:

Systems for monitoring and control of the process from power generation to power supply and distribution (FIG.1)

- Monitoring System
  At Instrumentation & Control (I&C) system of Nuclear Power Plant (NPP), the system is a system to monitor parameters showing NPP status. It has no influence in operating NPP even if the monitoring system shuts down.
- Control System
  At NPP, the system is a system to control equipment of NPP. It has influence in operating NPP if the control system shuts down.

Business Data Processing System for rate calculation, procurement of materials, and accounting, etc.

Since February 1996, we have taken steps to systematically make systems ready of NPP for the Y2K problem. We reported the following items to the Agency of Natural Resources and Energy, the Ministry of International Trade and Industry. After the reports, the agency evaluated that our measures for Y2K readiness were proper and reasonable.

Status of Readiness

Report on the method of the survey for Y2K problem, the methods of modification for Y2K problem and validation, the time of modifications, etc. (submitted on June 21, 1999)

Contingency Plan

Report on measures against assumed influences of the Y2K problem (submitted on November 1, 1999)

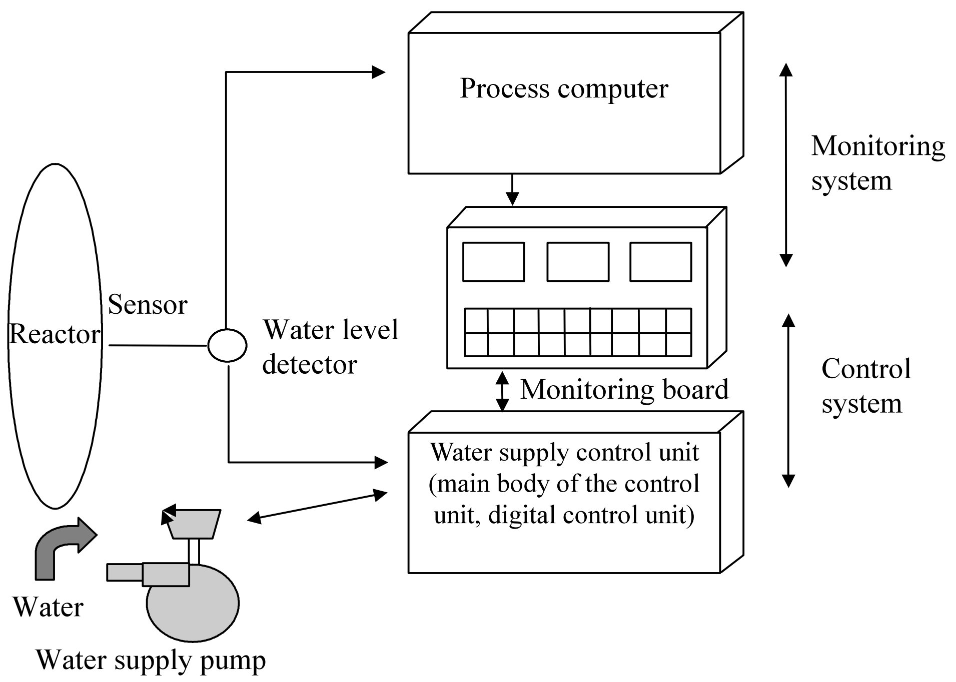

FIG.1 Outline of Instrumentation & Control at Nuclear Power Plant

2. Y2K READINESS HISTORY

We began taking measures for Y2K readiness at NPP in February 1998 when we started to survey the influence of Y2K problem at NPP. And, these measures were shown as follows:

February 1998:	Selected Systems that need Corrective Actions of Y2K problem.
April 1998:	Software Modification for Y2K problem (during an annual outage at Fukushima-Daiichi NPP No.1).
October 1998:	Conformed the integrity of our systems with investigation of all chips.
April 1999:	Starting a simulated test in public (during an annual outage at Kashiwazaki-Kariwa NPP No.6 Unit).
October 1999:	Software Modification for Y2K completed (at Fukushima-Daiichi NPP No.3 Unit).
October 1999:	Conducting a public simulated test with a process computer during plant operation (the first of such test in Japan).
November 1999:	Drill in properly responding to contingencies on the critical date (planned).

3. Y2K PROJECT

3.1. Project Organization

We consider Y2K readiness to be a priority task, and it should carry out to fulfill our responsibility and maintain public trust as a public utilities company. For this purpose, we set up the following company-wide and nuclear power division-wide organizations to share information on the Y2K problem and take interdivisional action for Y2K readiness:

Year 2000 Project Committee (Company-Wide)

- Chaired by vice president, with managing directors and directors of departments acting as members
- Established in October 1998

Nuclear Power Plant Y2K Meeting (Nuclear Power Division-Wide)

- Chaired by the group manager of the Nuclear Power Plant Management Dept.
- Established in January 1999

3.2. Feature of Instrument & Control System against Y2K Problem

The following are the features of NPP I&C system against the Y2K problem:

(1) I&C of NPP does not perform control using date information.

In principle, NPP is designed to operate at constant output and, therefore, does not have to change their output according to days and hours.

NPP control does not require information on hours because all component equipments are designed to be controlled according to the values of plant parameters (e.g., temperatures, pressure, water level and flow rate, etc.) which are always measured.

(2) I&C system of NPP is provided with exhaustive quality control that can trace events down to the level of chips on the IC substrates.

Especially strict quality control is implemented for the system to improve the reliability of plant components.

(Purposes)

- Improving the quality of individual components to secure the reliability of the plant as a whole.
- Ensuring that in the trouble of NPP I&C system, its causes could be investigated easily and that corrective action could be taken to prevent its recurrence.

To secure high quality, the system uses chips supplied from only the qualified vendors who have been found acceptable by a strict test and who can provide all necessary information on the chips they produce.

Concerning I&C delivered to NPP, all data from design descriptions to manufacturers' data reports are filed and maintained until replacement.

(3) Digital I&C system are replaced at relatively short intervals (10–15 years) because their existing technologies rapidly become obsolete.

In the digital system field, rapid progress of technologies makes it more difficult for us, year by year, to secure spare parts for repair and engineer for the system. Accordingly we have decided to replace existing I&C system with the latest ones at an interval of around 15 years.

(Example)

- Startup of 1F1 (Fukushima-Daiichi NPP No.1 Unit) 1971
- First replacement of process computer 1985 (14th year of using)
- Second replacement of the computers (planned) 2004

These arrangements enable us to obtain necessary information on the system hardware, including the types, vendors' names and specifications of chips used for the I&C substrates and on its software, including the program list.

3.3. Survey

FIG. 2 shows a conceptual diagram of surveys on the influence of the Y2K problem on I&C systems of NPP.

3.3.1. Hardware Survey

With cooperation from the manufacturers of systems, we conducted the following survey:

Step 1

We picked up all systems in service by using design descriptions that cover the whole system. And we confirmed whether these systems are digital systems or not.

Step 2

Based on the manufactures' design descriptions and parts lists[1], we picked up all chips of Real Time Clock [2](RTC) on all boards.

Step 3

Based on specifications, technical information (i.e., instruction manuals), questions to the manufacturers, etc., we confirmed integrity to Y2K problem for picked up RTCs:

Whether these RTCs are able to continue accurately dating beyond Y2K;

Whether these RTCs cover leap years, etc.

3.3.2. Software Survey

As in the case of hardware, the following survey was performed on software with cooperation from the system manufacturers:

Step 1

All program lists were checked to survey those parts of Operating Systems and Application Software which use date information.

With software[3] for detecting those commands which use dates, Program Listing was surveyed the parts that use date information.

Step 2

Concerning the parts found to use date information, their date information processing methods were surveyed on the basis of such data as the program list and design descriptions to find whether they would be influenced by the Y2K problem.

[1] A parts list shows all components of an IC and has some other entries, including the type of IC used.

[2] Real Time Clock is a chip having the function to count the time.

[3] The software to detect commands that use dates means a software package for retrieving character strings by Grep, etc.

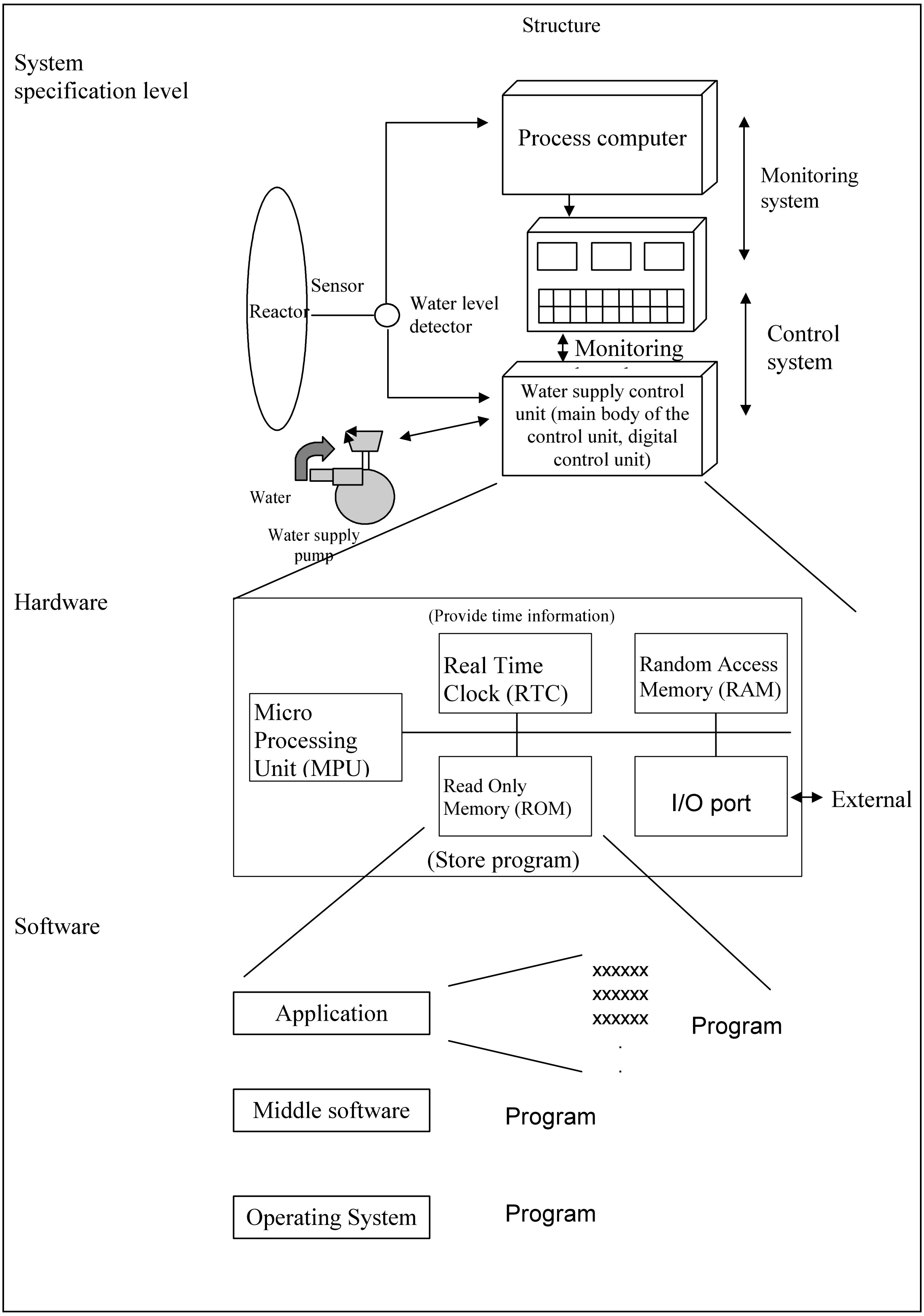

FIG.2 Conceptual Diagram of Y2K Survey

3.3.3. Survey Result

(1) Control Systems

None of the control systems was found subject to the influence of the Y2K problem as they do not use date information directly for control purposes.

(2) Monitoring Systems

Monitoring systems showing NPPs status use date information, and some of these systems were required software modifications for Y2K readiness.

Of important systems[4] installed at NPP, those which require software modification are shown in TABLE 1.

TABLE.1 IMPORTANT SYSTEMS REQUIRING SOFTWARE MODIFICATIONS

System	The Number of Systems
Process Computer [5]	22
SPDS Computer [6]	1
Computer for Rad-Waste [7]	12
Computer for Monitoring Post [8]	3

The following are major influences of Y2K problem on these systems:

Irregularity in the display of dates on the CRT screen

Irregularity in date printout

Failure of some application programs to operate, etc.

(3) Hardware

The survey found that RTCs, although outputting only the last two digits of the year, will continue to operate properly in Y2K and beyond and also properly respond to leap year. None of the RTCs was found to be subject to the influence of the Y2K problem. Incidentally all systems at NPP were found to have about 4,000 RTCs.

3.3.4. Corrective Actions

(1) Either of the following two methods is mostly used for software modifications:

Two digits represent a method of adding the program that is able to discriminate 1900’s from 2000’s. This method is used for many systems.

(Ex.) 80 - 99 = 1900s; 00 - 79 = 2000s

The other method changes year data from the last two digits of the year to full four digits.

(Ex.) 19"00" (before modification) "2000" (after modification)

(2) The cost of these software modifications totaled about \350 million (approx. $3.2 million at an exchange rate of \110 to the dollar) for all NPPs.

3.3.5. Current Status of Important Systems

TABLE 2 shows the current status of Y2K readiness for important systems at NPPs.

[4] The "important systems" means I&C systems which are important to NPP safety and operation.

[5] The Process Computer displays and prints out various parameters indicating the status of NPP, calculates plant performance and provides some other functions.

[6] The SPDS Computer displays technical important parameters for safety and operation of NPP.

[7] The Computer for Rad-Waste displays and prints out parameters indicating how radioactive waste treatment facilities are operating.

[8] The Computer for Monitoring Post measures and records radiation levels on the periphery of Nuclear Power Station.

TABLE.2 CURRENT STATUS OF IMPORTANT SYSTEMS

Items	Monitoring System (Plant Monitoring System, etc)	Control System (Feedwater Control System, etc)
Systems	66 (53)	164 (24)
Corrective Actions Not Necessary	28 (24)	164 (24)
Corrective Actions Required	38	0
Compliance for Y2K Problem	66	164 (26)
% Completed	100%	100%
Completion Date	Completed (October, 1999)	Completed

(System using times in parentheses)

3.4. Testing & Validation

3.4.1. Basic Principle

(1) Systems Required Modification
We performed simulated test of Y2K problem for all modified systems in site.

(2) Systems not required Modification
Important Systems
Simulated test is performed for systems using RTCs (capable of date setting) in site.
Other Systems[9]

System as divided by type should be validated at facilities of the manufacturer's factory or at NPP facilities.

3.4.2. Methods of Simulated Test

Using the time-setting command of the operating system or the time-setting function of the application program, the time set for each system should be changed in an attempt to find whether the system operates properly after the change.

The operation of the system should be validated in respect of all functions, including the interface with external units. Where data are transmitted between two systems, for instance, the time set for both systems should be changed in an attempt to find, among others, whether they can properly transmit and receive data after the change.

(Examples of Test Dates)
December 31, 1999 — January 1, 2000
February 28 - 29, 2000 — March 1, 2000

[9] Other systems are recorder, maintenance tool, etc.

(Items of Validation)

Capability of inputting the specified dates

Capability of accurately displaying and printing the dates

Proper operation of all programs without any failure

Never affect control signal (applicable to only control systems), etc.

3.5. Contingency Plan

3.5.1. Company -Wide Plan

In view of our social responsibility for maintaining a steady supply of electric power, an essential element of infrastructure, we have always been ready to take action in case of an emergency, such as a facility failure or natural disaster, with manuals prepared on emergency procedures and drills held regularly. In order to implement all necessary measures to maintain a steady supply of electric power, TEPCO, as an electric utility, has prepared the contingency plan for the Y2K problem based on the existing contingency plan. The contingency plan aims to prepare for internal and external failures peculiar to the Y2K problem, such as troubles with systems having monitoring and recording functions and a sudden drop in load due to internal or external factors.

The contingency plan is based on the following fundamental concept[10]:

We should do our utmost to prevent any impediment to carrying on our operations with the standard operating and monitoring procedures by completing on schedule program modifications and simulated tests of programs with monitoring and recording functions for major control systems involved in the operation of power generation facilities.

A malfunction of any of the computers with monitoring and recording functions installed at a power plant would cause no problem with the operation of the plant because it could be operated or safely shut down manually. However, we will adopt a conservative policy and enhance our stand-by system to provide operational back-up, as well as our immediate response system.

In preparation for any Y2K problems which might occur, we will set up a company-wide communications system based on the anti-disaster system which has already been developed in preparation for emergencies such as facility failures and natural disasters.

As part of prearrangements, we perform a drill in responses to contingencies.

Based on the fundamental concept described above, the contingency plan was formulated, taking into account internal and external risks. The completion of its formulation on June 21, 1999, the plan was announced publicly.

Internal Risks

Trouble of monitoring systems for Y2K problem

External Risks

Sudden drop in electricity demand due to a failure for Y2K problem at major customers' facilities

3.5.2. Nuclear Power Division-Wide Plan

The contingency plan for NPP, completed on November 1, 1999, prescribes specific measures based on the company-wide plan.

3.5.2.1. Purposes of Contingency Plan

Nuclear power plants are very carefully designed to ensure their safety. In order to secure safety in operation, personnel training course is given at regular intervals based on operation manuals during an equipment failure or accident, as part of routine safety activities. These arrangements enable NPP

[10] Only the parts relating to NPP were extracted.

to smoothly take all measures for Y2K readiness.

Based on the conditions stated above, we prescribe specific measures by the contingency plan, in addition to routine safety activities, in a bid to ensure that plant integrity will be maintained in the Y2K and that even if the plant is affected by Y2K problem, we could quickly shift to responses based on the existing manuals.

FIG. 3 shows the positioning of the contingency plan.

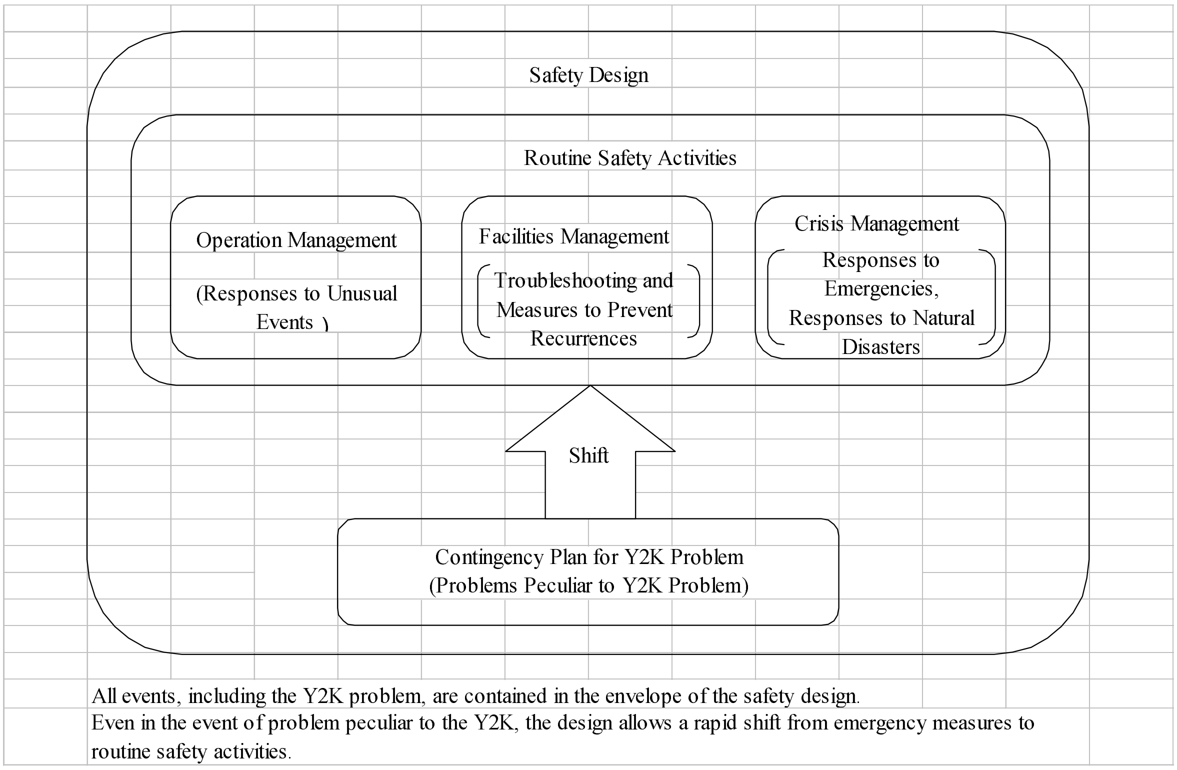

FIG. 3 Positioning of the Contingency Plan

3.5.2.2. Postulated Influences of Y2K Problem

The following risks were postulated in formulating the contingency plan for NPP. Then we assessed the impacts of the postulated risks on the plant in case they actually occur. The assessment found that any of the risks would not impair the safety operation of the plant.

(1) Internal Risks

We believe that the survey of Y2K problem on systems installed at NPP is very low error risk because these systems are provided with thoroughgoing quality control covering each part as well. Accordingly internal risks are limited to those which may arise from a failure of some important systems whose software has been modified for Y2K readiness. These systems are Process Computer, SPDS Computer and the Computer for Monitoring Post.

(2) External Risks

Among conceivable external risks are variations in system voltage and in system frequency and troubles with telecommunication facilities. Of these external risks, telecommunication facilities are considered to have a very low risk of being affected by the Y2K problem because, in addition to facilities provided by two or more common carriers, TEPCO has its own communication circuits. Therefore, external risks are limited to the instance where the load on the power system varies significantly due to the impact of Y2K problem on customers' facilities that are connected with the system.

3.5.2.3 Responses to Postulated Impacts of Y2K Problem

As stated in the preceding subsection, neither internal nor external risks impairs the operation of the plant, but a specific response plan was formulated for the following purposes:

Ensuring that plant integrity will be maintained in Y2K.

Ensuring that even if the plant is affected by Y2K problem, plant operation could quickly shift from emergency measures to responses based on the existing manuals.

(1) Prearrangements

Identification of monitoring systems

Preparation of manuals

Training in proper responses on critical dates[11]

Securing a sufficient stock of plant expendables

(2) Measures for Critical Dates

Collecting data on major equipment and checking their functional integrity by a test run

Priority monitoring of specified digital systems installed at NPP

Posting operators on standby at major equipment during the critical hours.

Make system of communication between our and manufactures engineers for emergency of Y2K problem.

3.5.3. Measures for Critical Dates

Steps should be taken to enhance measures, especially monitoring capability, and arrangements for posting assigned personnel on standby and improving their communication for the critical date from December 31, 1999, through January 1, 2000.

More specifically, a company-wide total of about 3,500 liaison and standby personnel will be added to some 1,300 workers who will be on duty by regular shift on December 31, 1999 and January 1, 2000. The personnel at our NPP will be increased by around 120 who will be posted at the plants as monitoring-support and communication staffs on these critical dates.

3.6 Open Simulated Tests

We performed open simulated tests on monitoring and control systems actually installed at NPP to provide an opportunity for outsiders to see that we had steadily implemented measures for Y2K readiness.

The aim of the simulated test was to confirm the behavior of the systems when their clocks were set forward to January 1 or February 29, 2000.

(Check List)

Monitoring system: Accuracy in date display on the CRT screen and in date printout

Control systems: No influence to control signals and accurate dating of failure information

(Record of Open Simulated Tests)

April 16: Test at Kashiwazaki-Kariwa NPP No.6 Unit (open to the local press)

June 14: Test at Kashiwazaki-Kariwa NPP No.5 Unit (open to Diet members)

June 16: Test at Fukushima-Daini NPP No.4 (open to the press)

June 29: Test at Fukushima-Daini NPP No.4 Unit (open to private Y2K study groups)

Sept. 6/7: Test at Fukushima-Daiichi NPP No.1 Unit (open to the press)

Oct. 29/30: Test at Fukushima-Daiichi NPP No.5 Unit (open to the press)

The open simulated test at Fukushima Daiichi NPP No.5 Unit was confirmed the behavior of the systems when the date set for the process computer was changed during operation. It was the first of such test ever confirmed in Japan during plant operation.

[11] Critical dates are date having possibility that a system occur failure of system by Y2K problem.

4. CONCLUSION

The Y2K problem at NPP will not affect the safe, stable operation of the plant, as described below, and we believe that all of our NPP will see January 1, 2000, without involving any problem.

The parts that are directly related to the control of equipment do not use any date information, nor do they require any modification for Y2K readiness.

Some monitoring systems were required software modifications for Y2K readiness.

These software modifications had been completed by October 1999.

Following the modifications, a simulated test was performed to verify the integrity of the modifications.

The systems which use time information were subjected to a validation test to prove the integrity of the survey for Y2K problem.

Contingency plan for Y2K problem is set up for the specified critical dates.

TECHNOLOGICAL DEVELOPMENT OF GUANGDONG NUCLEAR POWER STATION

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Abstract

After over 5 years of operations, the Guangdong Nuclear Power Station (GNPS) has achieved good results both economically and in operational safety performance. The main attributes to the success of the plant operational performances include the equipment reliability, the technical capability and management efficiency. To that the key strategy has been to adopt know-how and technological transfer and encourage self-innovation, aiming to strive for the long-term self-reliance in design, manufacturing and operating the plant.

1. INTRODUCTION

Daya Bay Nuclear Power Station contains of 2x984MW PWR units, and were inaugurated for commercial operation respectively in February and May, 1994. After over 5 years of operations, the plant has achieved good results in both economic and operational safety performances (see Figures 1-5).

The two units have not experienced any unplanned reactor scrams for nearly 800 days in average, and both units had realized OCTF (one cycle trouble free) in 1997 and 1998 respectively. The longest continuous operating record within one cycle has marked to over 310 days in the captioned year (for 12 month per cycle).

Among the WANO 10 performance indicators, over half have reached the world best quartile level. The grid off-take energy price has also been showing a decreasing trend each year, and remains below that of Hongkong thermal plant, indicating a high competitiveness in the electric market of the region. We can say, the overall performance of GNPS has reached beyond the world median level.

2. KEY ATTRIBUTES TO THE PROGRESS OF GNPS

GNPS consider the following three elements as the key attributes to the progress of the plant performance: equipment reliability, technological capability and management efficiency.

In order to enable a continuous improvement in all these three areas, the plant has adopted know-how and technological transfer plus self-innovation as the critical strategy. The ultimate goal for that is to not only strive for the best quartile performance of the plant but also realize self-reliance in technical expertise and management skills so as to prepare for the serial development of nuclear power industry in Guangdong region.

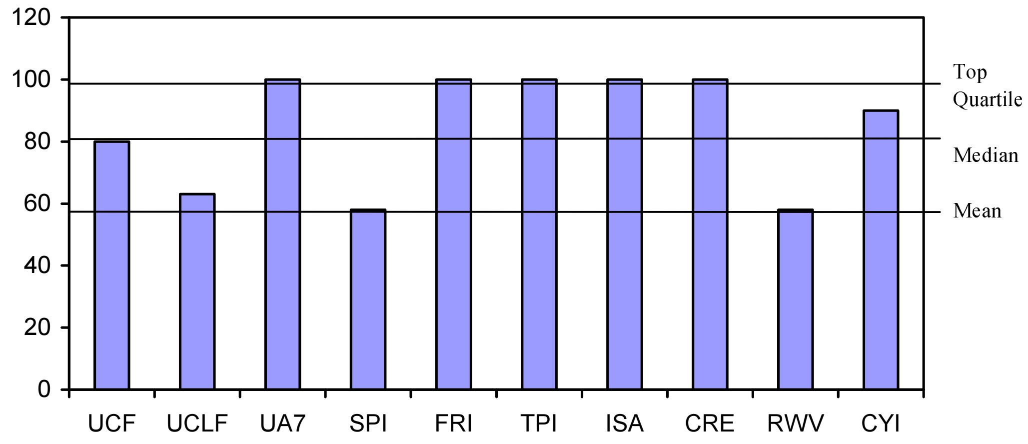

FIG. 1. GNPS 1998 performance.

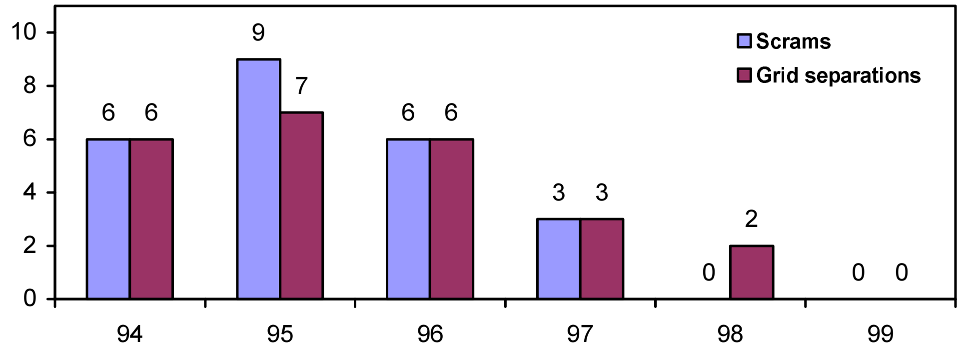

FIG. 2. GNPS scrams and grid separations.

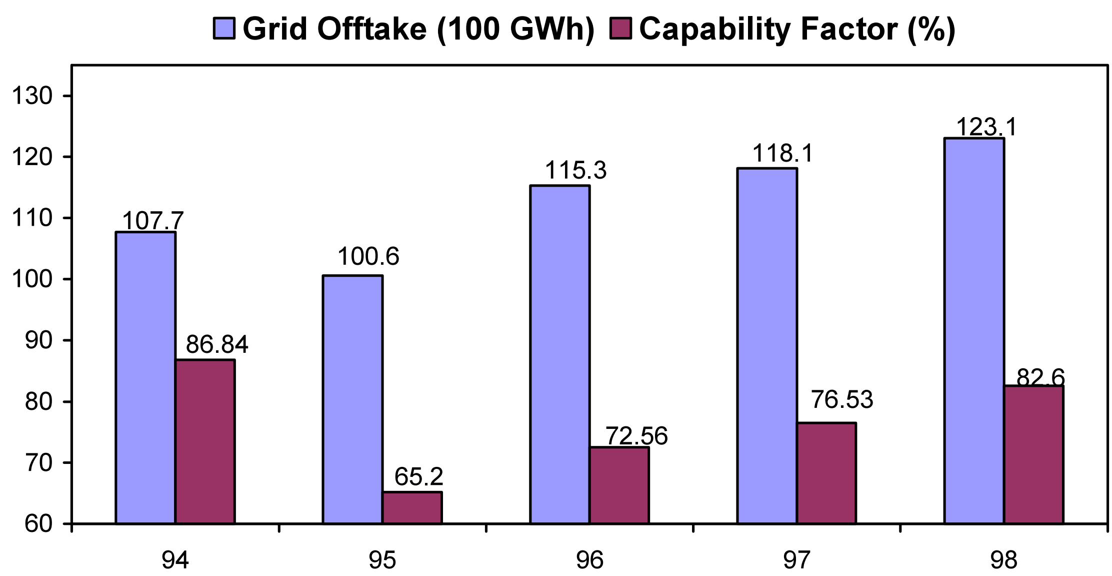

FIG. 3. GNPS capability factor and grid offtake.

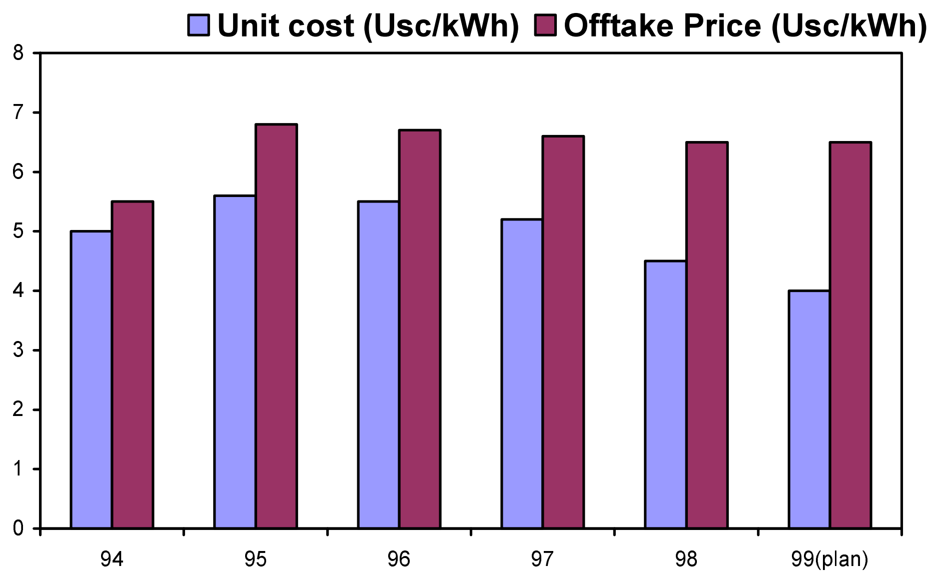

FIG. 4. Unit cost and offtake price.

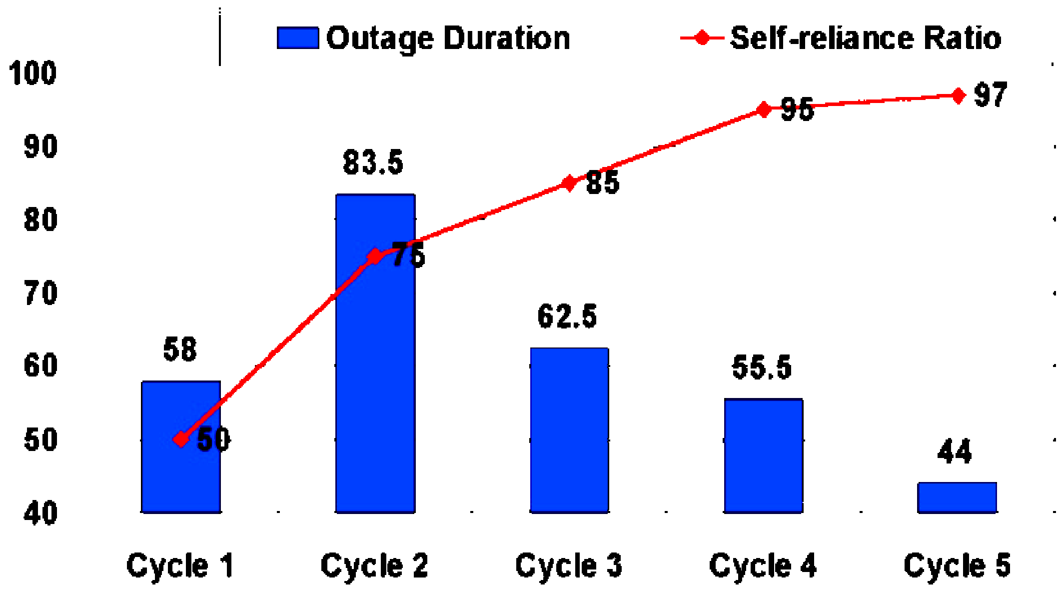

FIG 5. Self-reliance ration and outage duration.

3. IMPROVEMENT OF TECHNICAL EXPERTISE

GNPS import its major components and systems from France for nuclear side and Britain for conventional. As it is not realistic to have French and British people run the plant, one utmost important mission of the plant has been to take over the essential technology from the suppliers, and then by integrating successful practices worldwide to develop a mechanism of Daya Bay mode.

In doing so, GNPS signed a service contract with EDF (Electric De France) to seek expertise support for site technical issues. The latter has also, according to the contract conditions, seconded a batch of experienced experts to assume at the plant some key technical and engineering positions. EDF experts are also involved in the plant management, training, outage activities and procedure writing. Each expert is designated to a PRC individual as his counterpart. This has enabled the plant counterparts to grasp the technical knowledge and hands-on experiences on an on-line and dynamic basis through day-to-day cooperation.

The EDF plants in France have been the sound base for the preparation of GNPS first batch of RO (reactor operator), SRO (senior reactor operator) and staff of other key positions such as plant manager, I&C and Electric engineer etc. These people, after coming back to the plant, have become the driving force in the development of self-reliant technological capability.

The plant absorbs not only French experiences but also endeavor to integrate good practices of other countries domestic and abroad. The key method to that is to send out for training or to invite in for guiding. Especially for significant modifications and maintenance items, the plant has been in good contact with plants and utilities allover the world.

As a result, the ratio of self-reliance has been increasing rapidly year by year. Figure 5 and 6 has detailed the ratio of self-reliant outage maintenance items and the decreasing number of expatriates seconded to GNPS (see also Table I).

4. DEVELOPMENT OF MANAGEMENT TOOLS

Taking as a special kind of technology, the plant attaches high importance to the improvement of management skills. By applying the similar strategy, GNPS has developed some effective tools for technical management. These tools have tremendously facilitated the process of self-reliance in technology. Some examples are given below:

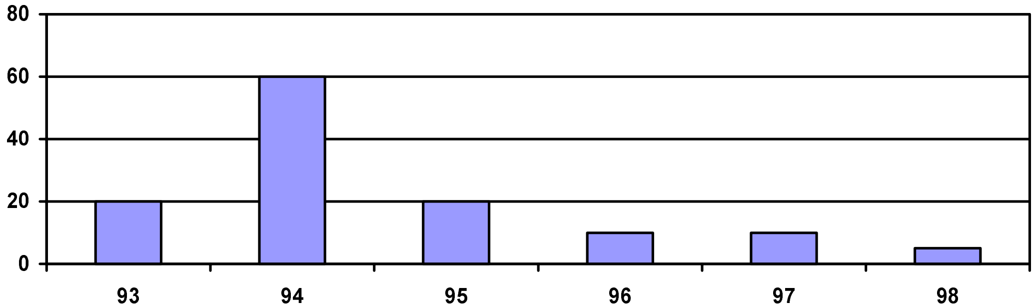

FIG. 6. Number of EDF staff at GNPS.

TABLE. I. MAJOR NI TASKS UNDERTAKEN BY PRC & FRENCH CONTRACTORS IN THE PAST OUTAGES

NI critcal items for outages	1	2	3	4	5	6
refueling	G	G	G	G	G	G
reactor vessel head open/closure	F	F	G	G	G	G
PMC pre-inspection commisioning	F	F	F	F	F	F
SG flushing	G	G	G	G	G	G
SG eddy current inspection	G	G	G	G	G	G
RCP primary pump maintenance	F	F	G	G	G	G
SEBIM valve maintenance	F	G	G	G	G	G
LHP/Q overhaul	F/5 year overhaul	F/5-year overhaul	G	G	G	G
primary steam isolation valve maintenance	F	F	F	F	F	G
NI valves maintenance	F	F	F	G	G	G
other NI pumps maintenance	F	G	G	G	G	G
reactor vessel hydrotest	—	—	—	—	G	G
electrical equipment maintenance	G	G	G	G	G	G
control&protection system maintenance	G	G	G	G	G	G
ISI	G	G	G	G	G	G
TUY53/56	F	G	G	G	G	G
RIC	F	F	F	F	F	F
DMW/DMR	F	F	G	G	G	G
P2R/SG open/close of man hole	F	G	G	G	G	G
primary steam pipe safety valve set point adjustment	F	F	F	G	G	G
reactor vessel bolt holes maintenance	F	F	F	F	F	G
in-core internals inspection	F	F	F	G	G	G
primary pump sealing inspeciton	F	F	F	F	F	F

G-items undertaken by PRC side (GUANGDONG)

F-items undertaken by French side (FRAMATOME)

GNPS applies SAT (systematic approach to training) for RO and SRO preparation. In order to maintain the qualification of the staff, the plant has integrated the "life-cycle" training approach in the SAT system. This means each personnel needs to take a batch of re-educational training as compulsory each year to keep his authorization effective.

GNPS implements rolling approach for technical issue resolution, which is usually called "Top Ten" approach. Each technical issue identified on the units is prioritized according to its significance of impact on plant safety, reliability and competitiveness. And the significance of impact is quantified to the defined weighting criteria. The ten issues of highest scores are listed as Top Ten, and receive high attention from the plant management. As soon an issue in the top ten list is closed out, the one next comes up automatically, thus ensuring that issues of more importance are resolved with higher priority.

GNPS uses risk monitor to assess plant safety risk on a dynamic basis with the theory of living PRA (probabilistic risk assessment). The project is not yet finally completed though;

the application of this tool from time to time on the intervention of safety-significant components has shown soundness for safety assurance (see Table II and III).

5. TECHNICAL INNOVATION OF LINGAO PROJECT

LANPS (LingAo Nuclear Power Station) is a copy + modification of GNPS. Because of the accumulation of technical experiences of GNPS both in hardware and software, LANPS has been able to localize a notable proportion of components and manage the project self-reliantly from the starting point.

To date, GNPS has fed back 268 items of technical improvement, and quite an amount of them have been successfully implemented in LANPS. The major locally manufactured components include steam generator, pressurizer, safety injection tank, moisture separator reheater, HP & LP reheater and deaerator etc.

TABLE II. SINGLE ITEM ASSESSMENT CRITERIA

	Unit reliability	Production cost	Nuclear Safety
5	Disconnected from grid > 24 h	Unit type, reactor scram MT trip	3R
4	Disconnected from grid > 12 h	Unit type, reactor scram	Backup power supply
3	Disconnected from grid > 4 h	Limit of power	Safety redundant
2	Disconnected from grid > 2 h	Disturbed power	Common failure
1	Disconnected from grid > 1 h	Reduced power	Monitor function

TABLE III. 3R: REACTIVITY, COOLING, RADIOACTIVITY AND WEIGHTING CRITERIA

Assessment criteria	Weighting factor
Nuclear safety	5
Unit availability	3
Production cost	1

6. CONCLUSION

GNPS has applied a good strategy in the development of technological self-reliance and has made good progress. The achievements are reflected in the improving performance of the plant and the notable ratio of self-reliant components/systems of LANPS. The progress is also evidenced by the fact that PRC staff took over the position of GNPS plant manager two years earlier than expected. We are confident to see more equipment made in China and more stations operated by mere Chinese in the near future.

MODERNIZATION PROGRAMME AT DUKOVANY NPP

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Abstract

The main goal of each NPP is to produce electricity safely, economically and without influence to environment. For Dukovany NPP it means to upgrade all documentation and perform the Equipment Upgrading Programme. All these activities are time and money consuming and therefore the determination of priority of all items was necessary. In the presentation there are mentioned some important changes in documentation, results of PSA studies and reason for Equipment Upgrading Programme performance. It was selected the most important item from the list of Equipment Upgrading Programme the I&C upgrading. Management has decided that Dukovany NPP will become among the best NPPs with VVER type of reactor. It seems this decision is the best way how to extend lifetime of the NPP.

1. INTRODUCTION

Dukovany NPP is now the only nuclear power plant in operation in the Czech Republic. The second one (NPP Temelin VVER 1000 MW) is under construction, the first criticality is expected at the end of the next year 2000.

Nuclear power plant Dukovany has got four units in operation. The units were built according to newer Russian design VVER 440/213, 440 MW each. The four units started commercial operation between November 1985 and January 1988. Dukovany NPP covers one fourth of the Czech Republic consumption of electricity.

2. OPERATION

2.1. Documentation

The NPP modifies all operating procedure to be update and in the same form. Practically each operating procedure has two parts now, a manipulation one and a description part. This year the NPP is finalising a program of a digital flowcharts and it continues a large programme of fulfilling a database for all devices, i.e. valves, pumps, protections,...
New symptom based EOPs developed under Westinghouse methodology has started this month (1.11.1999), together with new abnormal operating procedures.

Benefits from the new symptom based EOPs:

- Procedure cover large spectrum of emergency situations, combination of accidents, even low probability accident.
- Solutions have their own priorities.
- Independent checking of Critical Safety Functions.
- Several levels of diagnoses correct errors of operators (decrease human factor).
- Several levels of diagnoses allow to correct solution, when conditions are changed because of progress of accident.

Next year we will add some shutdown procedure based on PSA study. Many recommendations from Westinghouse to better performance of new EOPs are mentioned on part 5.

The same replacement of EOPs will be done in Slovakia´s NPPs Bohunice and Mochovce and Paks NPP in Hungary as well.

We plan to write the SAMG from 2001. We participated as co-beneficiary in PHARE project Beyond Design Basis Accident Analysis and Accident Management, Filtered Venting and Hydrogen Control. All PHARE projects were supplied by European Union.

2.2. Training of EOPs

From last year Dukovany has got the multifunction simulator on site. It has served for operator staff training, validation of new EOPs and transient studies. Project of full scope simulator will be finished next year. Training of all crew from a new EOPs has started two years ago.

3. MODIFICATION DUE TO EOPs

3.1 Pressure in hydroaccumulators

We decreased pressure in hydroaccumulators (HA) as a result of thermohydraulic analysis investigated different pressure in HA in many significant accident. This analysis has been done by UJV REZ research institute. Optimal pressure was different pressures for two HA injecting below core and for two HA injecting above core. This solution with different pressures was not optimal for operations. So, we use the second optimal solution and we decrease pressure in HA from 6 MPa to 3,5 MPa. Reason for decreasing pressure in HA was not only that the maximum pressure in HA was higher then opening setpoint of SGSVs but also transient during LOCA accident – main parameter for this was fuel cladding temperature.

3.2. PORV + cold overpressure protection

New PRZR PORV was added two years ago. It is the third safety valve, made by different design and principle that the old main two safety valves. New one is qualified for steam, water, mixture steam-water and gases. New PORV is equipped by cold overpressure protection system.

3.3. Hydrogen control

Two years ago we placed 17 passive hydrogen recombiners in the highest parts of containment and RCPs room. But there is a possibility, that after new recalculation during developing SAMGs we will have to place more recombiners. Each recombiner has a hydrogen concentration sensor and a thermometer, with a recorder in the Control Room. Operator can estimate a place with probability of leak or break during LOCA accident by indication of increasing temperatures in containment.

3.4. Main changes in I&C

- pressure in containment, we added two new measurements, fully qualified, based on different principle, range is from deep underpressure to overpressure 50 kPa – 550 kPa (two times higher than design pressure).

- water level in containment, we added also two new measurements, fully qualified, based on different principle, range is 2 metres.
- core exit temperature, basic range 0°C–400°C we increased until 1000°C, for the possibility to measure temperature during accident with insufficient core cooling.
- Main Steam header rupture signal modification (ESFAS signal).

3.5. Auxiliary feed water pump

Last year we changed all auxiliary feed water pumps to increase shutoff head pressure from 5 MPa to 6,3 MPa, more than SGs SV opening setpoint.

3.6 Sump protection

We changed shape of containment sumps. New sumps are practically without possibility of clogging (it was tested by current insulation during recirculation phase from the containment sump for clogging in a research institute).

3.7. Emergency venting

Next year we will finish emergency venting from reactor vessel head and SGs collectors (VVER reactors have horizontal SGs) to cover a possibility of venting gases or steam voids, especially in high pressure mode.

4. PSA PROJECTS (PROBABILITY SAFETY ASSESSMENT STUDIES)

All changes in technology and documentation are reflected in all levels of PSA.

4.1. PSA level 1

Development of results of PSA-1 model for Dukovany NPP

• 1996	$1{,}84 \times 10^{-4}$	CDF (1/year)	(PRZR PORV)
• 1997	$1{,}09 \times 10^{-4}$		(RCPs sealing qualification)
• 1998	$9{,}93 \times 10^{-5}$		(Main steam header signal)
• 1999	$7{,}93 \times 10^{-5}$		(EOPs implementation)
• target 2010	$9{,}85 \times 10^{-6}$		

The main contributors to CDF are as follows :

• Break of Main Steam Header	50,3 %
• Fires	16,8 %
• Break of main Feedwater Header	9,3 %
• Loss of external power supply	5,4 %
• LOCA < 10 mm	4,4 %
• Release into intermediate circuit	4,2 %
• LOCA 60 – 100 mm	1,7 %
• Loss of circulating water pumps	1,4 %
• Rupture of SG primary header	1,3 %
• Rupture of cooling water pipeline in turbine hall	1,3 %
• Loss of SG feedwater	1,1 %

Achievement less than the value 1,0 $\times10^{-5}$ belongs to the objective of the most of operated NPPs and this reduction could be expected also from the IAEA (the limit 1,0 $\times10^{-5}$ is today recommended value for newly constructed NPPs).

Results of **Shutdown PSA** force us to add some shutdown procedures to current EOPs. The main contributors to CDF are as follows :

- LOCA during pressure and leak test
- LOCA during test of PORV and SVs of pressuriser
- loss of natural circulation
- loss of coolant of spent fuel storage pool

4.2. PSA level 2

The PSA-2 provides probability of radioactivity release from hermetically sealed area and data concerning the size of such release. Different categories of releases according to the size and timing of release are determined in the PSA-2 Project and for each individual category the size and the probability of release is computed. Large early release frequency (LERF) is the most interesting one from the point of view of safety evaluation.
Totally eleven categories of releases were concerned in the PSA-2 study for Dukovany NPP. Containment behaviour in the course of accident with reactor core damage was analysed.
Positive features of the Dukovany NPP containment:

- due to the evacuated bubbler condenser system the small overpressure is maintained
- large volume of containment in comparison with the size of reactor core

Determination of containment damage modes belongs to the results of these analyses.

- an early large failure (rupture, break) -12,5 % of cases
- an early small release - 8,9 %
- a late large failure (rupture, large break) - 0,9 %
- a late small release - 14,6 %
- no failure (an operational leakage is concerned)- 63,0 %

Computed frequency for an early large release of radioactivity:

LERF = $6{,}8*10^{-6}$

Rough estimate taking into account successful qualification of RCPs seal as well as implementation others changes of technology is

LERF = $3{,}3*10^{-6}$

Because PSA-2 is based on older results of PSA-1 and a lot of changes was already implemented, we must recalculate PSA-2.

5. MODERNIZATION, EQUIPMENT INNOVATION AND SAFETY UPGRADING PROGRAMME

The Dukovany NPP belongs today to the most safe and reliable power source of the Czech Republic. Implementing program of the NPP modernization will meet also increased requirements on safety, determined by the IAEA, in order to be classified among the top quality nuclear units of the same type and age. Our effort is directed to allay inappropriate doubt resulting from the NPP operation and any impact on its surroundings.

The programme was entitled "MORAVA". We would like to express by this title our attitude to the region and environment surrounding of our NPP. The title of programme was created using initial letters of the following English words:

MOdernization	equipment and systems
Reconstruction	equipment and systems
Analyses	safety cases
VAlidation	result and process correctness verification

Program MORAVA – main objectives
- to obtain an operational licence for Dukovany NPP operation to year 2025
- to reach an economical competitiveness on deregulated market
- to achieve a safety level comparable with the best NPPs

5.1. Approach to the area of modernization and safety enhancement

The interest of Dukovany NPP was focused already from the beginning of operation of each unit on the area of safety enhancement. Condition of the NPP designed in the seventies and commenced from 1985 and 1987 was continuously upgraded based on needs and requirements of operation as well as on the actual condition of plant including new requirements for safety. Such review was based on the actual condition of plant for given period. During work on design of nuclear power plant the origin Russian regulative was used for the area of safety. In the course of review by the State Office for Nuclear Safety (SONS), IAEA, external as well as internal audits and other reviewers, the condition of plant was on the higher level in comparison with condition in the year of units start-up. The Dukovany was reviewed by IAEA missions as well as the technical audit from point of view of valid standard issued by the IAEA, safety regulative in force in the Czech Republic as well as internationally accepted concepts, practice and generalised national standards. Standards of this type are developed (and revised) by the IAEA. This Project called NUSS (Nuclear Safety Standards) was commenced in half of the **seventies**. In such way, implementing existing and by Dukovany NPP accepted recommendations, the Dukovany NPP reaches step-by-step further qualitatively higher grade of nuclear safety.
The Technical level of the Dukovany NPP should be concerned not only for European Union but also world-wide standards.

5.2. First modernization

The first modernization was based on the international knowledge by Government Decree in 1986 so called "Dukovany NPP Backfitting" (after Chernobyl accident):
- improvement of electronic fire signalling as well as systems outdoor of the all auxiliary buildings
- haloid fire extinguish equipment for electrical equipment of the NPP units
- back up of the 4th system of the power supply of the 1st category of reliability (non-interrupted) for computers
- cooling of the steel roof construction in turbine halls
- provision of the central oil plant by fire extinguish equipment
- hydrogen elimination in atmosphere of containment in the case of accident with release of coolant
- system of teledosimetry
- to make the Spent Fuel Storage more compact (increase numbers of spent fuel assemblies in reactor halls)
- fire protective coats of all cabling

- Signalling system for inhabitants in surroundings in the case of accident

After 1990 (after the political changes in Central and East Europe) the real level of safety of VVER 440/213 type of reactor was concerned and reviewed. The question was discussed whether these reactors can be left under operation. An international review of safety (initiated by the German government) known under the name “Green Book” performed for 5th unit of the Nord NPP (or Greifswald in the former DDR). It was shown necessary to perform thorough assessment of actual condition of plant also in the case of the Dukovany NPP. Number of analyses and supporting programmes were performed for our NPP both by the experts from Czech Republic and within the framework of international activities. These data were used for development of Equipment Reconstruction Programme of the Dukovany NPP.

It concerns the following:

- “Operational Safety Report” after ten year operation
- technical audits (internal and external)
- IAEA missions (OSART, ASSET,..)
- state regulatory body requirements
- PSA level 1
- supporting analyses, PHARE (EU) as well as IAEA Projects
- exchange of operating experience within the frame of WANO
- common activities of VVER 440/213 units (Paks NPP, Mochovce, Loviisa, Bohunice,..)
- analyses performed within framework of “Emergency Operating Procedure” development
- review of conclusions of the above mentioned “Green Book”

Conclusions from all the assessments are presented in the following areas:

- Safety Issues in the Dukovany NPP design was assessed by calculations based on probabilistic analyses. This assessment follows from the Operational Safety Issues Report and in addition from the IAEA recommendations, i.e. Safety Issues resulting from extensive evaluation of VVER 440/213 design performed by the IAEA. This assessment contains 74 Safety Issues divided into three categories according to their relations to nuclear safety as well as 13 operational recommendations. Their fulfilment in Dukovany NPP was verified by the IAEA mission in 1995. Every year NPP sends Report of progress in fulfilment these Safety Issues.
- Consequences of lifetime exhaustion of individual equipment or their components respectively in relation to the lifetime given in design or by manufacturer or estimated from operating experience (technical audit). The most relevant problem in this area is the I&C equipment.
- Impact of equipment failure rate, cost demands for maintenance of operated units
- Area of supporting analyses, programmes, upgrade of design base documentation, for example equipment qualification, check of pipeline integrity, programme of probability of core melting (PSA level 1, 2, living PSA, risk monitor, shutdown PSA), EOPs and all operating documentation upgrade.

5.3. Activities of Equipment Upgrading Programme

The above mentioned activities resulted in the set of recommendations, topics and requirements for measures leading to fulfilment of Equipment Upgrading programme objectives. The set of activities within framework of the programme Morava was established based on the analyses and the following determination of priority of individual activities according to the “Rules for modification management of Dukovany NPP” which have the nature of Equipment Renewal Programme. The objectives of the Programme Morava are as follows:

to provide safe operation also in future

- technical equipment of the NPP shall be further operated in accordance with increased safety requirements as well as requirements of State Office for Nuclear Safety or with requirements which could be valid in the near future (IAEA, EU etc. recommendations)
- to enable capability of the NPP to achieve operating licence after year 2000 up to year 2025
- to achieve core damage frequency less than value 10^{-5} for reactor/year
- to be in compliance with requirements of European Union (achievement of international safety standard for smooth transition of nuclear industry of the Czech Republic into the European Union)

the operation must be economical

- to operate all four units of Dukovany NPP up to the end of their planned lifetime (it means to year 2015)
- to create preconditions for lifetime extension of the NPP (achievement of operating licence up to year 2025)
- to utilise design reserve of units of NPP (power increase)
- to provide competitiveness both within the ČEZ company and on the market with electricity in the Czech Republic
- to enable further operation and maintenance of equipment together with an early supply of spare parts and provision of reliable service in order to reduce unplanned loss of production due to equipment failure

5.4. Main areas of modernization included MORAVA programme

(1) Containment sump protection
(2) Equipment modification at floor +14,7 m of intermediate building (resistance of equipment and piping systems)
(3) Instrumentation and Control system reconstruction
(4) Displacement of sectional collector of emergency feedwater collector
(5) Interim Spent Fuel Storage extension
(6) Utilisation of unit design reserve (power increase)
(7) Full scope control room simulator
(8) to increase pH in the secondary circuit (set of activities – replacement of main condenser - titanium tubes - and related activities enabling change of chemical mode)
(9) Reconstruction of electrical part of Diesel generator Stations
(10) Replacement of auxiliary feedwater pumps
(11) Elimination of hydrogen during accidents
(12) Completion of powered valves on the special drainage from RCPs room
(13) Upgrading of valve automatics on emergency feedwater pump injection lines
(14) Reconstruction of ESFAS signal “Main steam head rupture”
(15) Pressuriser relief valve – completion of protection against cold overpressure
(16) Signalling of basement flood under turbine hall
(17) TV camera control for check of reactor internals
(18) Fire protection (assembly of spraying fire extinguishing equipment, electrical fire detection signalling, etc.)
(19) Activity necessary for new emergency operating procedures
(20) Cabling rooms spray coating
(21) Reconstruction of subdistribution boards

(22) Change of connection of 110 kV power supply reserve
(23) Technical Support Centre
(24) Activity resulting from LBB (leak before break)
(25) Activity resulting from operating experience, audit and from assessment of reactor core melting probability (PSA)
(26) Reconstruction of the air measurement in the stack
(27) Venting systems of main and emergency control rooms
(28) Fast acting valves – VELAN DN 450
(29) Source term in radiation surveillance
(30) Diagnostic systems (the set of activities enabling more extensive equipment monitoring)
(31) Intangibles (the set of analyses and safety cases by the IAEA and the State Office of Nuclear Safety.

5.5. Instrumentation and Control system reconstruction

Main goal is not only replacement of old sensors and relay systems by new modern reliable systems, but also a change of approach to nuclear safety. The reactor protection and ESFAS system will have the same architecture – three independent divisions (current reactor protection has two divisions). Reactor protection system will be divided into a reactor trip system and a reactor limitation system. We will add some new signals and modify current signals on the base on analyses and European and world standards. Real change of I&C will start in 2002 and run until 2010, now there is going a competition of Suppliers. List of main changes:

- reactor protection
  - PRZR high level protection 6,3 m
  - PRZR low level signal 70 cm
  - reconstruction of “main steam header” protection
  - new protection high pressure in “main steam header” (heat sink failure)
  - new protection from low pressure in feed water collector
  - new protection in reactor low power
  - to cancel the reactor trip signal due to the turbine trip
  - to cancel the protection of reactor limitation (drop of groups of control rods consequently) to simplify the system of Reactor Trip

- ESFAS system

– to modify current signal “Main steam header rupture” (together with reactor protection)
– to modify signal “overpressure in containment” into two step (low overpressure and higher overpressure)
- new signal feed collector rupture
  to add function to reset ESFAS signal or time delay
  to add function to initiate signal manually

- ELS – Emergency Load Sequence (current two ELS programme replace by one common program)
- PAMS – post accident monitoring system — to add some reliable measurement that will monitor the unit after an accident for long-term conditions – this system will be tied with SPDS (safety parameters display system) for monitoring Critical Safety Functions and the information system in the control room during accidents.

COMMERCIAL OPERATION AND OUTAGE EXPERIENCE OF ABWR AT KASHIWAZAKI-KARIWA UNITS Nos. 6 & 7

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Abstract

Kashiwazaki-Kariwa Nuclear Power Station Units Nos. 6 & 7, the world's first ABWRs (Advanced Boiling Water Reactor), started commercial operation on November 7, 1996 and July 2, 1997, respectively, and continued their commercial operation with a high capacity factor, low occupational radiation exposure and radioactive waste. Units 6 & 7 were in their 3rd cycle operation until 25th April 1999 and 1st November 1999, respectively. Thermal efficiency was 35.4-35.8% (design thermal efficiency: 34.5%) during these period, demonstrating better performance than that of BWR-5 (design thermal efficiency: 33.4%). Nos.6 & 7 have experienced 2 annual outages. The first outage of unit No. 6 started on November 20, 1997 and was completed within 61 days (including 6 New Year holidays), and the second outage started on March 13, 1999 and was completed within 44 days. The first annual outage of unit No. 7 started on May 27, 1998, earlier than it would normally have been, to avoid an annual outage during the summer, and was completed within 55 days, and the second outage started on September 18th, 1999 and was completed within 45 days, All annual outages were carried out within a very short time period without any severe malfunctions, including newly designed ABWR systems and equipment. As the first outage in Japan, 55 days is a very short period, despite the fact that the Nos.6 & 7 are the first ABWRs in the world and the largest capacity units in Japan. The total occupational radiation exposure of No. 6 was 300 man-mSv (1st outage) and 331man-mSv(2nd outage). That of Unit 7 was 153 man-mSv (1st outage) Those of unit No. 6 were at the same level as those of unit No. 3, which is the latest design 1100MW(e) BWR-5. That of unit No. 7 was the lowest ever at Kashiwazaki-Kariwa nuclear power station. The drums of radioactive waste discharged during the annual outage numbered 54(1st outage) for No. 6 and 62 (1st outage) for No. 7, which was less than the design target of 100 drums and that of unit No. 3. Overhaul and inspection of new designs components, such as reactor internal pumps (RIPs), advanced type control rod drive mechanism (FMCRD) and large-capacity and high-efficiency turbine system, were conducted as planned, verifying their integrity after the commercial operation period. Operability during annual outage through ABWR-type main control room panels, which have a very new man-machine interface with touch-screen CRTs and flat displays, was also confirmed. In this paper, the commercial operation and annual outage experience of Kashiwazaki-Kariwa Nuclear Power Station units Nos. 6 & 7 are reported.

1. INTRODUCTION

Kashiwazaki-Kariwa Nuclear Power Station units Nos. 6 & 7, the world's first ABWR (Advanced Boiling Water Reactor) s, started commercial operation on November 7, 1996 and July 2, 1997 respectively, and continued their commercial operation with a high capacity factor, low occupational radiation exposure and radioactive waste. Units 6 & 7 were in their 3rd cycle operation until 25th April 1999 and 1st November 1999, respectively. Thermal efficiency was 35.4-35.8% (design thermal efficiency: 34.5%) during this period, demonstrating better performance than that of BWR-5 (design thermal efficiency: 33.4%).

In this paper, the commercial operation and annual outage experience of Kashiwazaki-Kariwa Nuclear Power Station units Nos. 6 & 7 are reported.

2. TECHNICAL FEATURES OF ABWR

The main technical features of the ABWR from an operation and maintenance point of view are as follows.

2.1. Reactor internal pump (RIP)

In place of the traditional external recirculation system with two large capacity pumps, a reactor internal-type pump system with ten pumps, directly installed on the bottom of the reactor pressure vessel, was introduced. This system has the technical features listed below.

(a) Reduction of radiation sources and in-service inspection work through elimination of external recirculation piping;
(b) Reduced leakage potential of reactor water through the adoption of a wet-type motor without shaft seals;
(c) Reduced station service power thanks to the reduction of the power requirement of the recirculation system.

2.2. Advanced-type control-rod drive mechanism (FMCRD)

In place of the traditional control-rod drive system, which drives control rods with hydraulic pressure both during normal and scram operations, an advanced-type control-rod drive mechanism with two driving methods — hydraulic pressure for scram and motors for normal operation — was introduced. This mechanism has the technical features listed below.

(a) Easier reactivity control through fine-motion control by using motors
(b) Shortened start-up time with automatic gang operation
(c) Simplified and optimized hydraulic scram accumulator system through two CRDs per accumulator and elimination of the scram discharge system
(d) Reduced occupational radiation exposure and shortened maintenance-outage period through the adoption of split-type housing, which makes the main body maintenance-free and gathers the parts that require maintenance to the spool piece at the bottom of main body.

2.3. Reinforced-concrete containment vessel (RCCV)

In place of the traditional self-standing steel containment vessel, a reinforced-concrete containment vessel, which allows a larger degree of freedom in selection of its shape, was introduced. Through design optimization, a cylindrical-shape containment vessel was decided upon, and a compact arrangement with sufficient workspace inside the containment vessel was achieved.

2.4. Integrated digital instrumentation and control system

The ABWR instrumentation and control system uses state-of-the-art technology, including digital technology, optical multiplexing signal transmission and power-electronics technology, and has the technical features listed below:

(a) Operator-friendly man-machine interface;
(b) Compact one-man operator console;
(c) Wide display panel showing a summary of plant status;
(d) Reduced operator work load by expanded automation;
(e) Easier recognition of plant anomalies by classifying alarms into priority ranking;
(f) Enhanced controllability, reliability and maintainability through application of a digital system.

2.5. Large-capacity and high-efficiency turbine system

Through the introduction of large turbine system featuring low-pressure turbines with 52-inch last-stage buckets, moisture separator/heaters, and high-pressure and low-pressure heater drain pump forward systems, large electricity output(1,356 MWe) and improvement in thermal efficiency of more than 1% are accomplished.

3. OPERATION EXPERIENCES

The first-cycle and second-cycle operation records of Kashiwazaki-Kariwa nuclear power station units Nos. 6 & 7 are shown in Table I. First cycle operation was very smooth, and we were able to have a very high capacity factor. We experienced 4 unscheduled shutdowns at Nos. 6 & 7 during the second and third cycle operation. But none of them were due to either safety-related issues or characteristic ABWR design features. A summary of these issues is described in Table II.

The thermal efficiencies and auxiliary power ratios of both units were 35.4-35.8% (design thermal efficiency: 34.5%) and 3.3% (design auxiliary power ratio: 3.3%) during that period, demonstrating better performances than those of BWR-5 (thermal efficiency of 33.4% and auxiliary power ratio of 4.0%; both are design values). The power required for the recirculation pumps during the first-cycle operation is shown in Table III, in comparison with unit No. 3 (BWR-5). This table shows that the adoption of the internal pump system contributes to the reduction of the auxiliary power ratio.

4. SUMMARY OF ANNUAL OUTAGES

The major inspection items of first and second annual outages are listed in Table IV. Many of these are required by the regulatory body and also inspected by them to confirm the integrity of these systems and equipment.

5. SCHEDULE OF ANNUAL OUTAGE

As mentioned, the first annual outages of units Nos. 6 & 7 of Kashiwazaki-Kariwa nuclear power station were completed within 55 days net, which is a very short first annual outage period.

TABLE I. COMMERCIAL OPERATION RECORDS OF KASHIWAZAKI-KARIWA UNITS NOS. 6&7

	K-6		K-7	
	1st cycle	2nd cycle	1st cycle	2nd cycle
Electricity power output	1356 MW (ABWR)			
Date of commercial operation startup	7 Nov. 96	18 Jan. 98	2 Jul. 97	20 Jul. 98
Date of off grid	20 Nov. 97	13 Mar. 99	27 May 98	18 Sep. 99
Electricity generated (MW(h))	12,295,400	13,382,472	10,698,920	12,115,576
Capacity factor (%)	99,9	98,5	99,9	87,7
Electricity generation period (h)	9,072	10,020	8,736	10,188
Auxiliary power ratio (5) (after startup)	3.3 (Design:3.3)	3.3 (Design:3.3)	3.3 (Design:3.3)	3.3 (Design:3.3)
Thermal efficiency of gross power generation (%)	35.8 (Design:34.5)	35.6 (Design:34.5)	35.8 (Design:34.5)	35.4 (Design:34.5)

TABLE II. UNSCHEDULED PLANT SHUTDOWN EVENT OUTLINE AT K-6 & 7

Unit	Date of Event Occurrence	Shotdown Period	Event Outline	Root Cause & Countermeasure
K-6	1998-08-29	98h58m	During rated power operation, the reactor was tripped by activation of the 500kV bus protection system. At the same as occurrence of this event, a transmission line accident occurred due to lightning.	Faulty wiring in the bus protection system during construction stage was the root cause of the actuation of the bus protection system. The improper conection was corrected and also QC activity for wiring work was improved.
	1999/5/25	201h12m	During rated power operation, the reactor was tripped by a generator exciter trip.Before the reactor trip,one of 5 power conversion modules in the exciter system was out of order. The power conversion module system has redundancy, and thus rated power operation could be continued. But the reactor was tripped just after the control power of conversion module with multifunction out of service for maintenance.	A control power shoutdown signal caused by power conversion module isolation and an out of service sigral from the same module occurred at the same time of the exciter trip. The computer misunderstand that those 2 signals were 2 power conversion modules trip signals cause by a program error. The computer program was corrected, and the computer system's integrity was confirmed by a mock-up test at the factory and an actual computer at the site.
K-7	1999/3/31	916h50m	Off-gas radiation level and reactor water I-131 level were increased during rated power operation.The I-131 level was below tech. spec. value.(I-131 level=1.2×102Bq/g, Tech. Spec. = 4.6×103Bq/g) But plant was manually shutdown for the sake of caution.	Fuel leak from one bundle was detected by in-core sipping.Leak bundle was replaced to new bundle. It is evaluated fuel leak occurred by random failure.
	1999/7/28	258h50m	During rated power operation,one reactor internal pump was tripped.Pump trip was caused by power supply terminal piece break.Full power operation could be continued by 9 RIPs but plant was manually shutdown to replace the terminal piece in the containment vessel for the sake of caution.	The broken terminal piece was caused by high cycle fatigue because of resonance between pump vibration and terminal cable.Cable support condition was improved to prevent resonance, and terminal piece structual strength was also improved.Terminal piece integrity was confirmed through a factory mock-up test and actual RIP operation test.

TABLE III. REQUIRED POWER FOR RECIRCULATION PUMPS DURING THE FIRST-CYCLE OPERATION

	Generator output (MWe)	Core flow rate (Ton/H)	Pump speed (%)	Required power for pump (MWe)	Ratio to generator output (%)
K-3	1100	44657	89.1	8.77	0.80
K-6	1356	53110	82.7	5.00	0.37
K-7	1356	52780	85.4	5.74	0.42

TABLE IV. MAJOR INSPECTION ITEM OF 1ST AND 2ND ANNUAL OUTAGE

Component			Total Number	Maintenance Cycle	K-6		K-7	
					1st Outage	2nd Outage	1st Outage	2nd Outage
Reactor Sys. Component	RIP		10	5 outage	2	2	2	2
	FM CRD	Main Body	205	25%/10 outage	3	5	11	3
		Spool Piece	205	10 outage	21	21	21	21
	Replacement of LPRM		52	-	7	7	7	7
	Safety Relief Valve		18	Every outage	18	18	18	18
	Main Steam Isolation Valve		8	Disassemble; 4 outage	2	2	2	2
				Leak Test; Every outage	8	8	8	8
	New Fuel Loading		872	-	180	240	156	192
Turbine Sys. Component	Main Turbine		4casing	2 outage	4	2	4	2
	Moisture Separator Reheater		2	2 outage	2	1	2	1
	Main Valve	Main Stop Valve	4	Every outage	4	4	4	4
		Control Valve	4	Every outage	4	4	4	4
		Combination Intercept Valve	6	Every outage	6	6	6	6

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A summarized first annual outage schedule of the turbine system and the reactor system is shown in Figs.1 and 2.

The inspection schedule of the turbine system, which contains many inspection items, is critical for the overall schedule of the first annual outage. The inspection procedure for the turbine system consisted of disassembly, repairs, inspection, re-assembly, oil flushing, pre-start test and restart. For unit No. 6, almost all of the turbine disassembly, repair and re-assembly work was done in two shifts lasting about 20 hours per day. The total time required for the turbine system inspection for unit No. 6 was 1,056 hours. On the other hand, the total

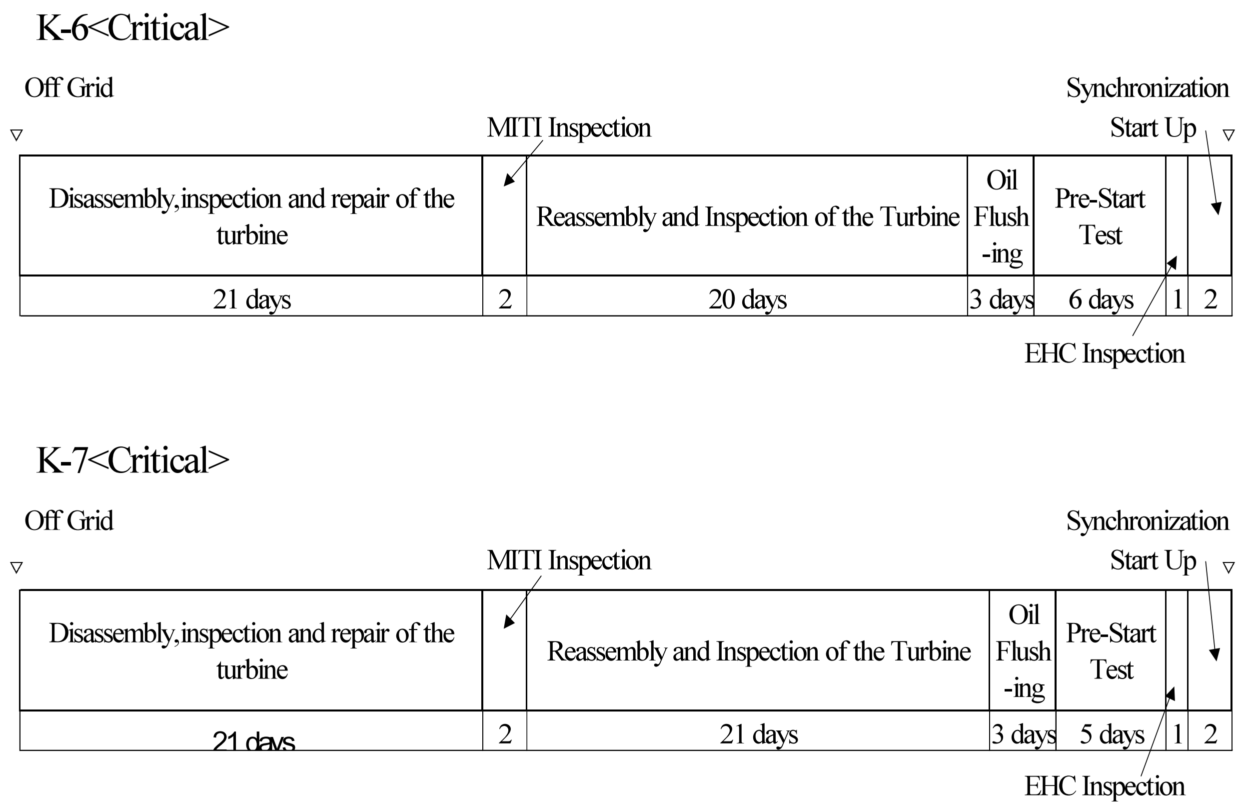

FIG. 1. Actual progress schedule for turbine system. 1st outage.

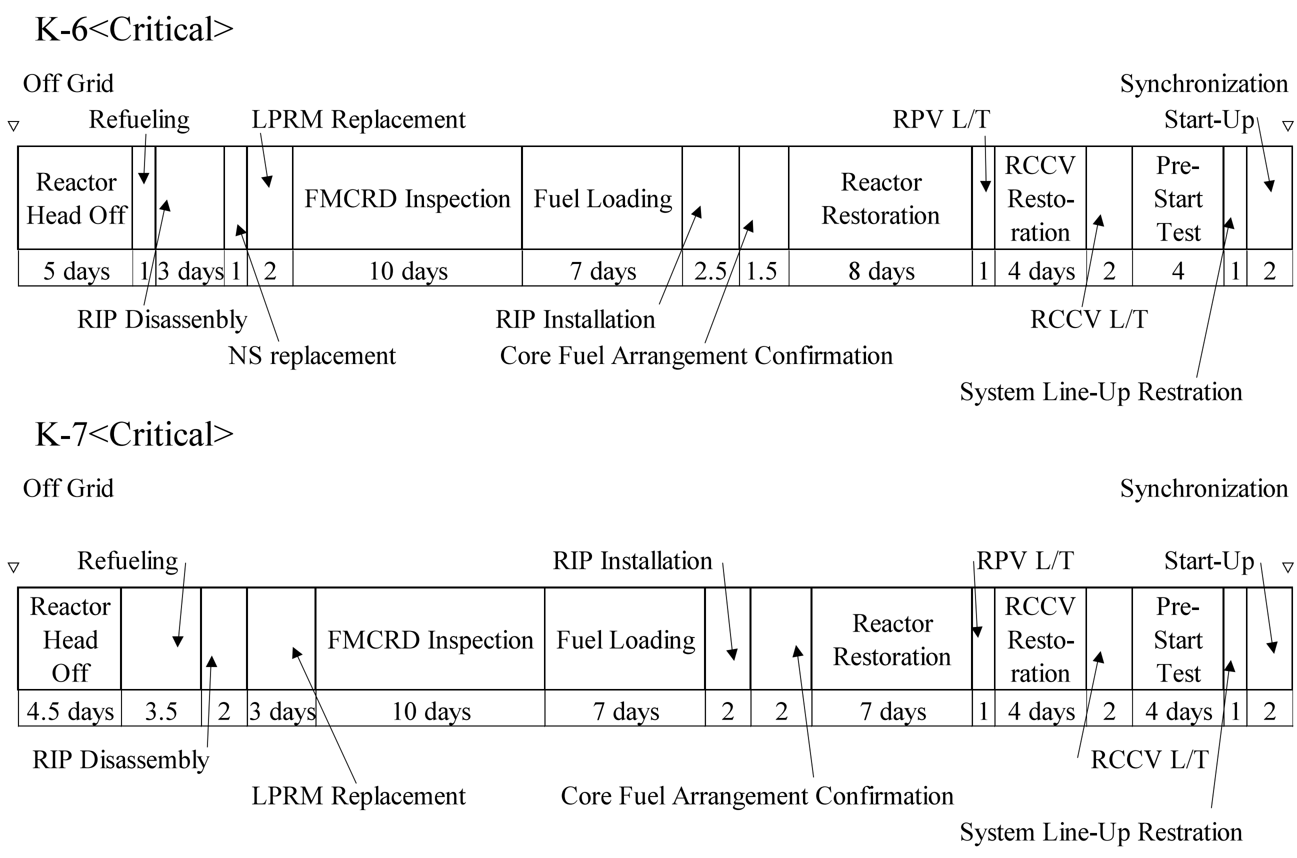

FIG. 2. Actual progress schedule for reactor system (1st outage).

time required for the turbine system inspection for unit No. 7 was reduced to 826 hours through sufficient preparation based on the experience with unit No. 6. The critical procedure for the reactor system consisted of RCCV/RPV head off, refueling, RIP removal, replacement of neutron monitors, FMCRD inspection, fuel loading, RIP re-installation, core fuel arrangement confirmation, reactor restoration, RCCV restoration, pre-start test and restart. The total time required for the reactor system was 734.5 hours and 891.5 hours for units No. 6 & 7, respectively. The difference between units No. 6 & 7 was caused by the difference in the number of inspected FMCRD main body and refueling fuels.

On the other hand, in the second annual outage, the inspection schedule of the reactor system becomes critical for the overall schedule, because there much fewer turbine system inspection items than in the 1st outage. (In the 1st outage, 4 turbine casing were overhauled, but in the 2nd outage, 2 casings were overhauled).

From the experience of the first annual outage, we estimated that an outage period of 45 days or less was possible with the present work shift. Nos. 6 & 7's outage was planned to be within 44 days. Inspection volume and contents of reactor system and equipment were at the same level as the 1st outage. Major difference between K-6 and K-7 was the neutron source removal at K-7. Regularly, the neutron source is removed at 1st outage, but K-7's first cycle operation period was 11 months to satisfy summer peak demand in 1998. Detailed evaluation concluded that the neutron source should remain in the core for the second cycle operation start up.

Both second outages were carried out smoothly. K-6's outage was carried out within 44 days. And at K-7, where during startup a turbine bearing thermocouple indication error was detected, the outage was 10 hours more than planned due to the repair work. The actual second annual outage schedule of units No. 6 is shown in Fig. 3.

Further reduction of the outage period can be achieved through reconsidering the inspection frequency of RIP and/or FMCRD, reconsidering the work shift, etc.

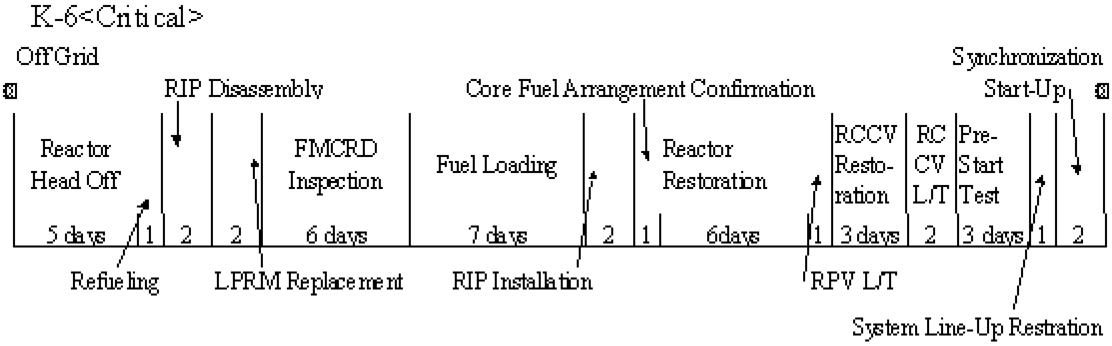

FIG. 3. Actual progress schedule for k-6 reactor system (2nd outage).

6. OCCUPATIONAL RADIATION EXPOSURE AND DISCHARGED RADIOACTIVE WASTE

The occupational radiation exposure of the first annual outage, classified by major work, is shown in Fig. 4. The total occupational radiation exposure of K-6 was 300 man–mSv (1st outage) and 331 man–mSv (2nd outage) and for K-7 153 man–mSv (1st outage) for units Nos. 6 & 7, respectively. That of unit No. 6 for 1st outage was at the same level as unit No. 3, which is a latest BWR-5. That of unit No. 7 was the lowest ever at Kashiwazaki-Kariwa nuclear power station. The reduction of the RIP-related occupational radiation exposure was due to the reduced dose rate at the lower drywell, where the RIP motor was removed and installed. The reduction of other work-related occupational radiation exposure stemmed from the improvement of water chemistry management and work management.

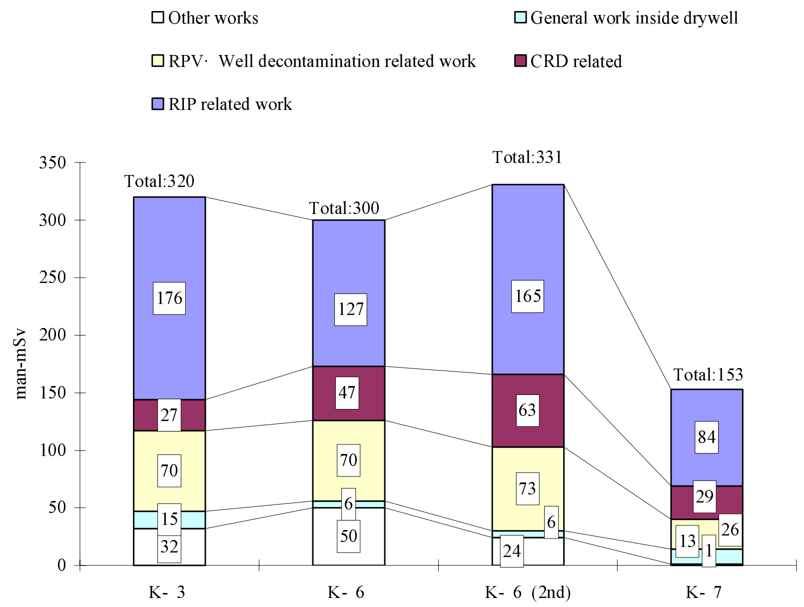

FIG. 4. Occupational radiation exposure during annual inspection.

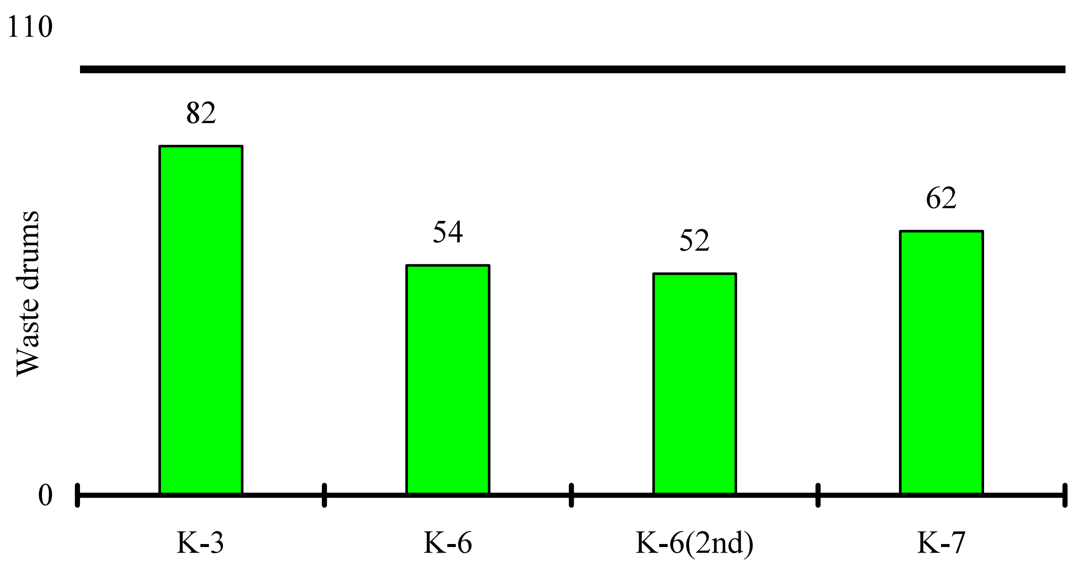

FIG. 5. Radioactive waste generated during annual inspection. (Design target=100).

The drums of radioactive waste discharged during the first and second annual outages are shown in Fig. 5. There were 54, 62 and 52 drums for units No. 6's first outage No. 7's first outage and No. 6's second outage, respectively, which was less than the design target of 100 drums and that of unit No. 3.

7. INSPECTION EXPERIENCE WITH NEWLY DESIGNED COMPONENTS

7.1. RIP

The removal and re-installation of the RIPs were performed from both the upper and lower sides of the reactor, as shown in Fig.6. From the upper side of the reactor, that is, the R/B operating floor, the RIP impeller shaft was removed and re-installed by means of a crane for a RIP, which was installed on the fuel-handling machine. From the lower side of the reactor, that is, the lower drywell, the RIP motor was removed and re-installed by means of a RIP handling machine.

The procedure for RIP removal is summarized as follows (See Fig.7):

(a) Pressurize the secondary seal to isolate the motor casing from the reactor water. After that, drain the water from the inside of the motor casing and remove the auxiliary cover.

(b) Remove the coupling stud nut, and we can disconnect the motor from the impeller shaft.

(c) Remove the motor.

(d) Attach the lower stop flange on the motor casing and supply clean water into the motor casing. After that, release the secondary seal and remove the impeller shaft.

(e) Install the upper plug.

Drain the water from the inside of the motor casing and remove the lower stop flange. After that, remove the secondary seal.

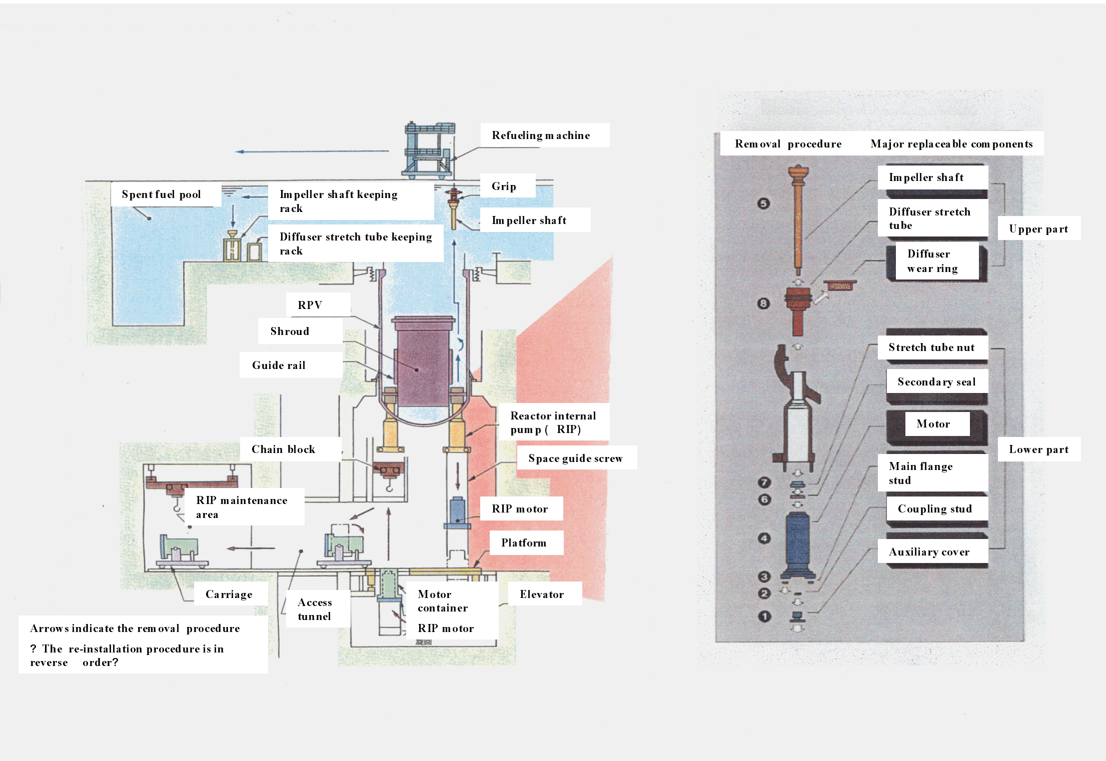

FIG. 6. RIP removal and re-installation.

The re-installation of the RIP was performed by reversing the procedure.

After moving the impeller shaft to the storage rack in the spent fuel pool, the impeller shaft was inspected by a remote inspection device. The removed motor was inspected in the RIP maintenance room outside of the lower drywell.

As shown in Fig.8, the required work time for the removal and re-installation of the RIP in the first annual outage was 27 hours for both units Nos. 6 & 7, which was as planned and less than planned, respectively.

The tests and inspections listed below were performed to ensure the proper operation of the RIP through 1st and 2nd outage.

(a) Visual inspection of impeller shaft
(b) In-service inspection of auxiliary cover, etc.
(c) Visual inspection, insulation-resistance measurement and dimension measurement of motor
(d) In-service inspection of RIP nozzle
(e) RIP test run (vibration characteristics, flow rate characteristics)

7.2. FMCRD

The removal and installation of FMCRD was performed by means of an FMCRD handling machine at the lower drywell, as shown in Fig.9. The FMCRDs were removed in the sequence of motor, motor bracket, spool piece and main body. The installation was done in reverse sequence. As the inspection frequency for motor and spool piece is 10 years, 21 motors and spool pieces out of 205 were inspected in the first annual outage. The FMCRD main body is designed to be maintenance-free, but 3 and 11 were inspected in units Nos.6 and 7 outages respectively to confirm their adequacy. As the inspection of the FMCRDs is critical for the reactor system inspection schedule, the removed FMCRDs were replaced with spare parts.

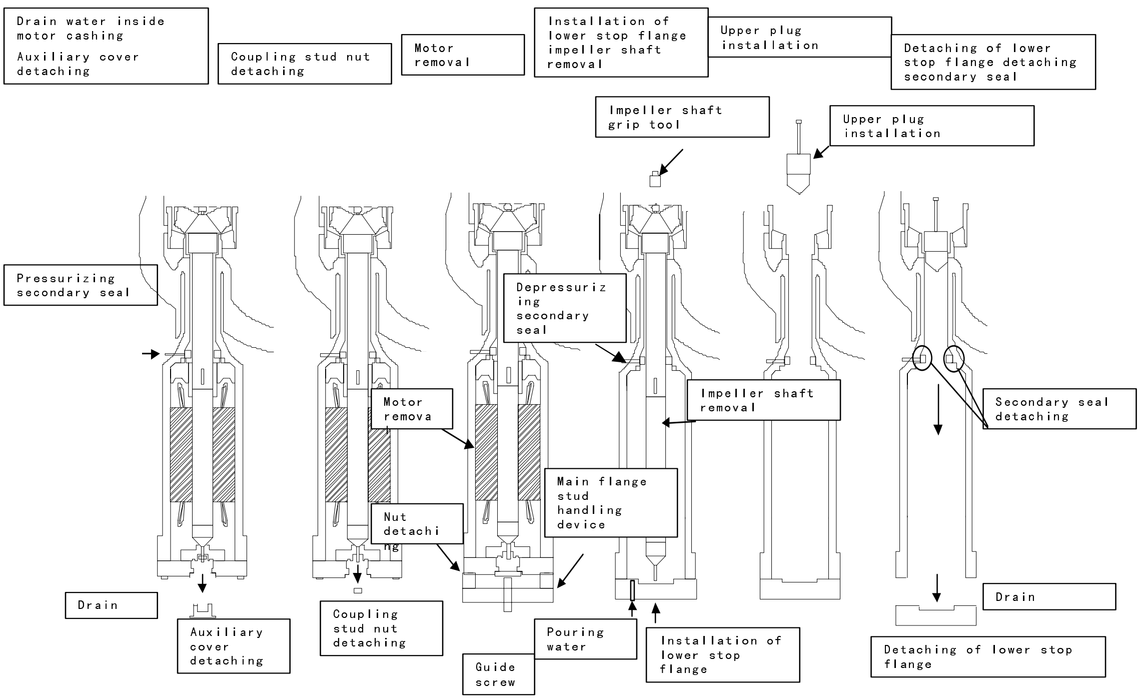

FIG. 7. RIP removal outline procedure.

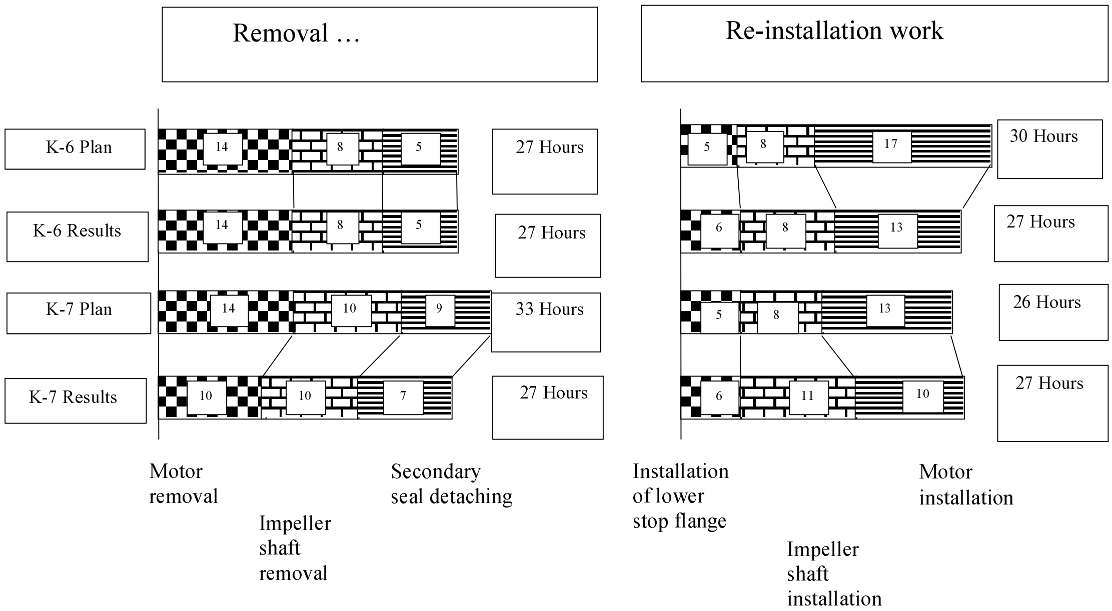

FIG. 8. Required work time for the removal and re-installation of RIP. The required work time for FMCRD removal and installation in the first annual outage is shown in Fig. 10. The planned and actual work time per FMCRD for units Nos. 6 & 7 is shown in the figure, classified into two kinds of work. One is full disassembly, and the other is partial disassembly (motor and spool piece). The actual work time was less than planned, but there were some differences between plant manufacturers.

The tests and inspections listed below were performed to ensure the proper operation of the FMCRDs through 1^{st} and 2^{nd} outage.

(a) Visual inspection and curvature measurement of main body
(b) Visual inspection and operational test of spool piece
(c) Visual inspection of motor
(d) In service inspection of housing bolt
(e) Friction test and scram test

7.3. ABWR-type main control room panel

Maintainability during an annual outage by use of an ABWR-type main control room panel was one of the check points, because the man-machine interface is considerably changed from the traditional main control room panel, such as a compact console with a touch-screen, large display panel, improvement of surveillance with CRT/FD and digitalization of all control systems, including the safety system.

Application of a software isolation system and a jumper lift device did not induce any problems, such as overlapping of operations or mis-operation. Plant monitoring capability during a process computer shutdown was able to be secured by the large display panel and FDs without CRTs. Isolation tags of FD with a touch-screen were stored in boxes set up beside the panel, and this was evaluated as the best possible method with the existing equipment.

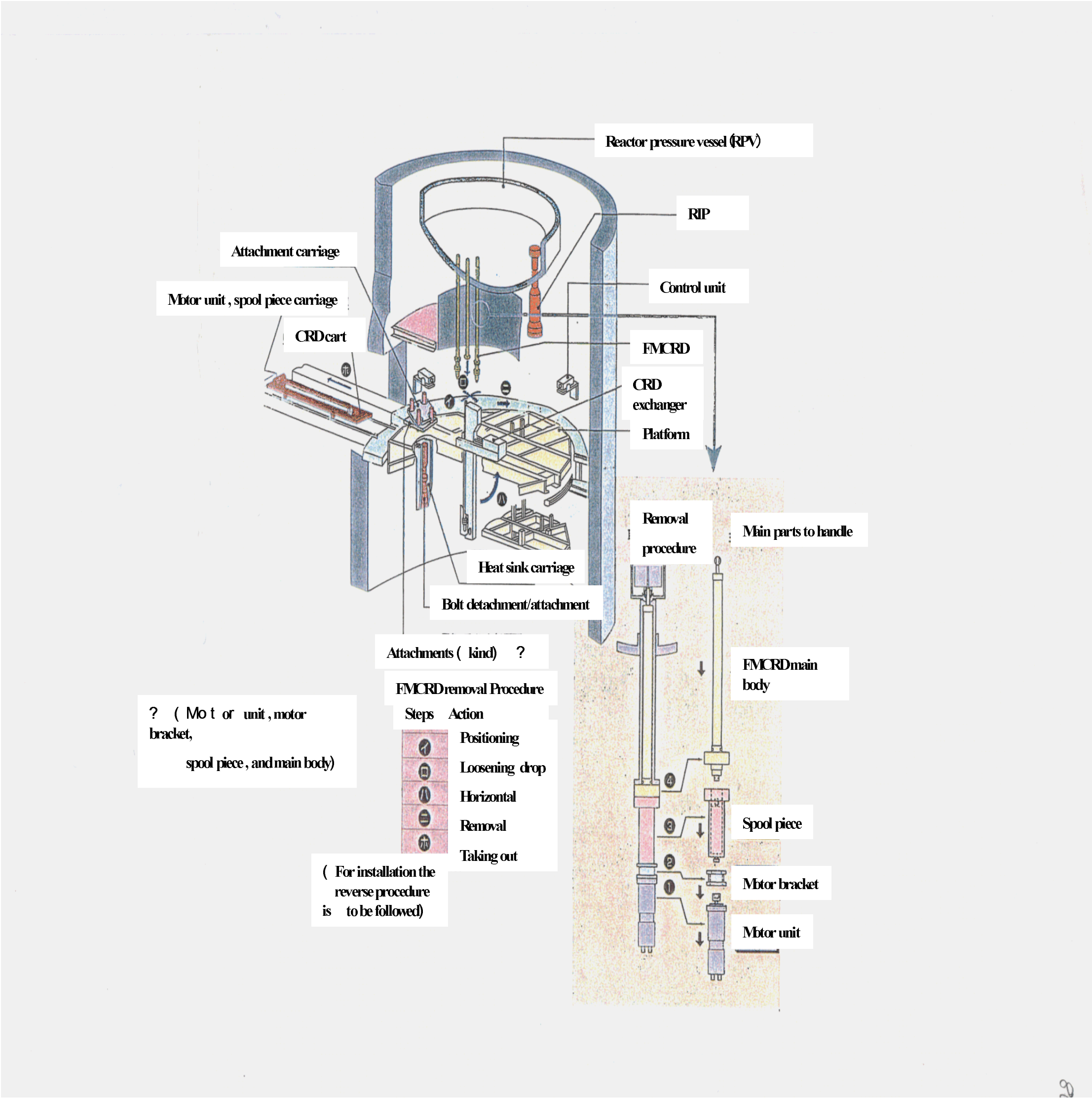

FIG. 9. FMCRD removal and installation.

The expansion of automatic operation, such as CRD operation, made the start-up time shorter than that of a traditional plant. Especially, the effect was great during critical operation and pressurization operation. (See Fig.11)

7.4. Large-capacity and high efficiency turbine system

In the compactly arranged turbine building, there were many inspection items during the first annual outage, such as disassembly of all turbine casings, all main valves and the generator. Therefore, lay-down planning for the turbine building-operating floor, including the common lay-down area on the second floor of the rad-waste building, was prepared in detail. The situation of the turbine-operating floor during the first annual outage is shown in Fig. 12.

In the second outage, there were fewer inspection items than in the 1st outage, and therefore the lay-down situation was less severe than that of 1st outage.

The newly designed turbine system components for an ABWR, such as large turbines featuring low-pressure turbines of 52-inch last-stage buckets, moisture separator heater and heater drain pump-forward system components, were inspected to ensure proper operation through 1st and 2nd outages.

			Removal	Installation	
K6	Spool piece Motor unit (18 bodies)	Plan	145 About 145 minutes	160 About 160 minutes	About 305 minutes
		Results	70 About 70 minutes	100 About 100 minutes	About 170 minutes
	Spool piece Motor unit Main body (3 bodies)	Plan	225 About 225 minutes	250 About 250 minutes	About 475 minutes
		Results	220 About 220 minutes	250 About 250 minutes	About 470 minutes
K7	Spool piece Motor unit (10 bodies)	Plan	100 About 100 minutes	145 About 145 minutes	About 245 minutes
		Results	90 About 90 minutes	120 About 120 minute	About 210 minutes
	Spool piece Motor unit Main body (11 bodies)	Plan	190 About 190 minutes	265 About 265 minutes	About 455 minutes
		Results	135 About 135 minutes	180 About 180 minutes	About 315 minutes

FIG. 10. Required work time for FMCRD removal and installation.

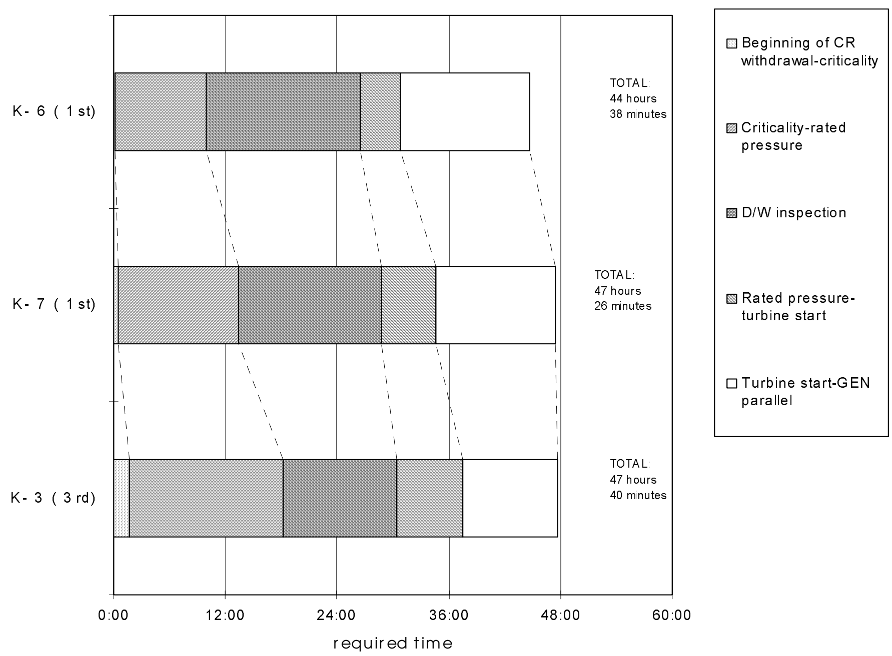

FIG.11. Comparison of start-up time.

FIG.12. Situation of the turbine operating floor during the first annual outage.

8. REACTOR-MAINTENANCE TRAINING FACILITY

We have an actual-size reactor pressure vessel, reactor internals, internal pump, FMCRD and handling machines in the reactor-maintenance training facility of Kashiwazaki-Kariwa nuclear power station. (See Fig.13)

Before the 1^{st} and 2^{nd} annual outages of units Nos. 6 & 7, training for the removal and installation of RIPs and FMCRDs was carried out. The procedures were checked and improved.

9. CONCLUSIONS

The commercial operation of and the annual outages experiences with an ABWR at Kashiwazaki-Kariwa nuclear power station units Nos. 6 & 7, the world's first ABWRs, were very satisfactory. During this period, we have accumulated valuable know-how and data for the operation and maintenance of ABWRs. We would like to further develop ABWR operation and maintenance.

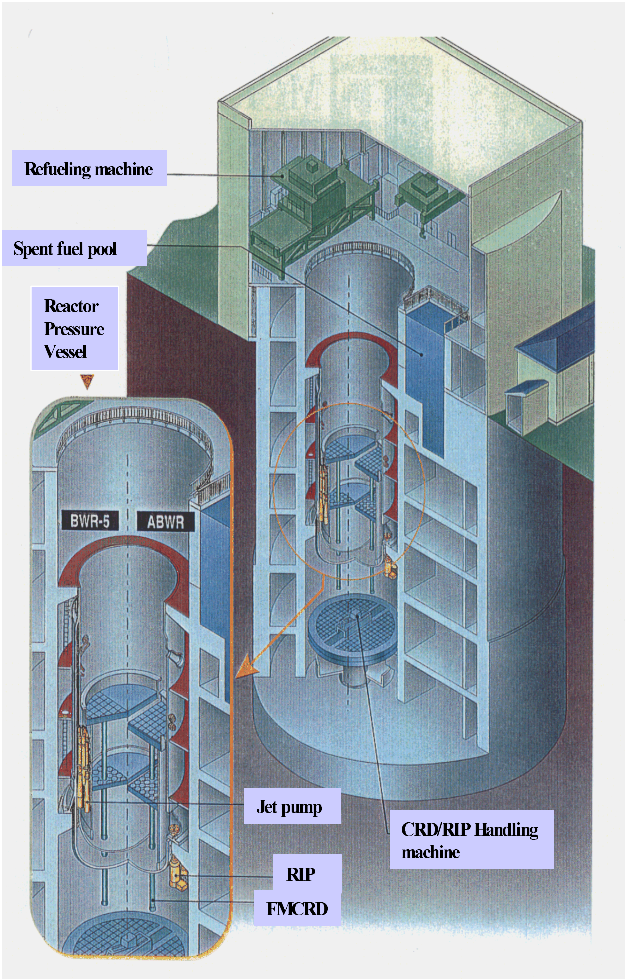

FIG. 13. Reactor maintenance training facility.

REFERENCE

[1] Fukuda, T. et al., “Operation and First Outage Experience of ABWR at Kashiwazaki-Kariwa Unites Nos.6 & 7”, 7th ICONE-7495, Tokyo, Japan Apr, 19-23, 1999

EQUIPMENT AND TECHNIQUES FOR COMPONENT INSPECTION, MAINTENANCE, REPAIR AND REPLACEMENT

(Session II-a)

Chairperson

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FULL SYSTEM DECONTAMINATION FOR DOSE REDUCTION AT THE PREVENTIVE MAINTENANCE WORK OF THE REACTOR CORE INTERNALS

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Abstract

At the Fukushima Dai-ichi Nuclear Power Station unit 3 and unit 2 of Tokyo Electric Power Company (TEPCO), the replacement of the core shroud and internals have been conducted respectively in the FY 1997 outage and in the FY 1998 outage. The replacement of the welded core internals in operating BWR plants is the first time in the world as complete countermeasure to improve SCC resistance. At present both units are operating smoothly. The developed technology concept is to restore those internals in air inside the reactor pressure vessel. To reduce the radiation dose rate inside the RPV, not only a shielding method was applied to cut the radiation from the irradiated structures but also a chemical decontamination method was applied to dissolve the radioactive crud deposit on the surface by using chemical agents. The CORD UV process was applied for this Full System Decontamination including operating the reactor recirculation pumps. The critical pass time required was approximately 7 days for each unit. In both units the radioactivity of 10 TBq (280 Ci) and the Fe, Ni, Cr crud of 60-70 kg as metal in total was dissolved and removed by 5 m^3 (175ft^3) ion exchange resins as only waste generated. The obtained decontamination factor (DF) at the RPV bottom reached 40-100. As result, the dose rate decreased to approximately 0.1 mSv/h under water. Before and after the installation of the in-vessel shielding, a mechanical cleaning was extensively applied inside the RPV to remove the residual crud as well as the cutting particles. As result, the RPV bottom dose rate decreased further to 0.03 mSv/h under water and 0.2 mSv/h in air. A better working environment for human access than expected was established inside the RPV, resulting the 70, 140 man*Sv saving respectively at unit 3 (1F-3) and unit 2 (1F-2).

1. INTRODUCTION

At the Fukushima Dai-ichi Nuclear Power Station of Tokyo Electric Power Company (TEPCO), the replacement work of the core shroud and internals with the SCC resistance material were conducted as a preventive maintenance countermeasure to the first generation BWR plants. At present, the replacement work has been completed at unit 3 (1F-3, 784 MWe) and unit 2 (1F-2, 784 MWe) respectively in the FY 1997 outage and in the FY 1998 outage. The same projects are now under going at unit 5 (1F-5, 784 MWe) and unit 1 (1F-1, 460 MWe) respectively in the FY 1999 and 2000 outage. As the internals, being replaced, are welded, a replacement technology concept that restores by welding in air has been developed for the first time in the world to secure the reliability of the replacement process. However, due to the more than 20 years operation, the radiation level inside the Reactor Pressure Vessel (RPV) is very high. In order to access and work there in air, the radiation dose rate had to be reduced. Radiation from the irradiated core structures is shielded by the in-vessel shielding. On the surface of the RPV and the core internals, the radioactive crud (metal oxide containing radioactivity such as Co-60) builds up. For this reason, a chemical decontamination method was applied to dissolve and remove the crud by using chemical agents.

The Full System Decontamination (FSD) was performed by operating the existing Reactor Recirculation Pump (RRP). In Japan component chemical decontamination is usually applied to the impeller of RRP. However, a system chemical decontamination for reuse has recently been applied to save the dose rate. Therefore a Full System Decontamination for human access into the RPV represents a break-through in the decontamination technology. After the FSD, some reactor core internals were cut under water and taken out in the DS pit. The in-vessel shielding equipment was successively installed surrounding the effective core region. Before and after the installation of the in-vessel shielding, a mechanical cleaning was also applied inside the RPV to remove the residual crud as well as the cutting particles. The result of the FSD and the mechanical cleaning is summarized in this paper.

2. FULL SYSTEM DECONTAMINATION

2.1. Chemical decontamination

2.1.1. Chemical decontamination method

The crud deposits on the surface of the structural materials generally consist of the outer layer and the inner layer. The outer layer is loose and is formed by the deposition of the radioactive crud in the reactor water. The inner layer is a tight grown-on oxide of the base material, in which the radio-activity penetrated. Since the reactor water in the BWR plant contains 200 ppb dissolved oxygen under the normal water chemistry condition (NWC) it is relatively oxidative, the outer layer consists of α-Fe_2O_3 and Fe_3O_4, the amount of which depends on the input from feed water. The inner layer consists of Fe, Ni, Cr spinel type oxides. The morphologies differ dependent upon the type of materials such as stainless steel and carbon steel, operation temperature, and water quality, etc. The iron oxides can be dissolved by the acid dissolution reaction and the reduction reaction by a reducing agent (a decontamination agent). On the other hand, the trivalent chromium oxide can be dissolved by the oxidation reaction as the hexavalent ion by an oxidation agent.

As well as an excellent decontamination factor and no detrimental effect to the structural material integrity, the minimization of waste generation are the important factors for selection of a decontamination process, especially for such large scale decontamination application. Today's decontamination processes show a tendency toward multi-cycle and very low chemicals concentrations. From these points of view, the CORD UV process (Chemical Oxidation Reduction Decontamination-Ultra Violet light) was chosen. This process is a multi-cycle multi-step decontamination process in the presence of oxygen at approximately 95 °C and atmospheric condition.

The oxalic acid of the CORD process is decomposed in situ to water and CO_2 by ultra violet light. The permanganic acid ($HMnO_4$) has less impact on waste generation since a relatively low concentration is enough in the oxidation step, and it can be also decomposed and removed as Mn^{2+} ion by ion exchange resin. This process provides a high decontamination factor and generates a minimal volume of chelate free waste by decomposing the decontamination agents.

The CORD process and its applications have been described in several papers [1-5]. The operational sequence of the FSD, typical for CORD UV, is shown in Fig. 1. A three cycle decontamination was applied at both 1F-3 and 1F-2. In the first cycle for BWR, the majority of the iron oxides was dissolved by the dilute oxalic acid with the preoxidation step skipped. All dissolved deposits including the activity were trapped on cation exchange resins. When the crud dissolution ceased, the following decomposition step was initiated in which, by ultra violet light, the oxalic acid was decomposed into carbon dioxide gas and water.

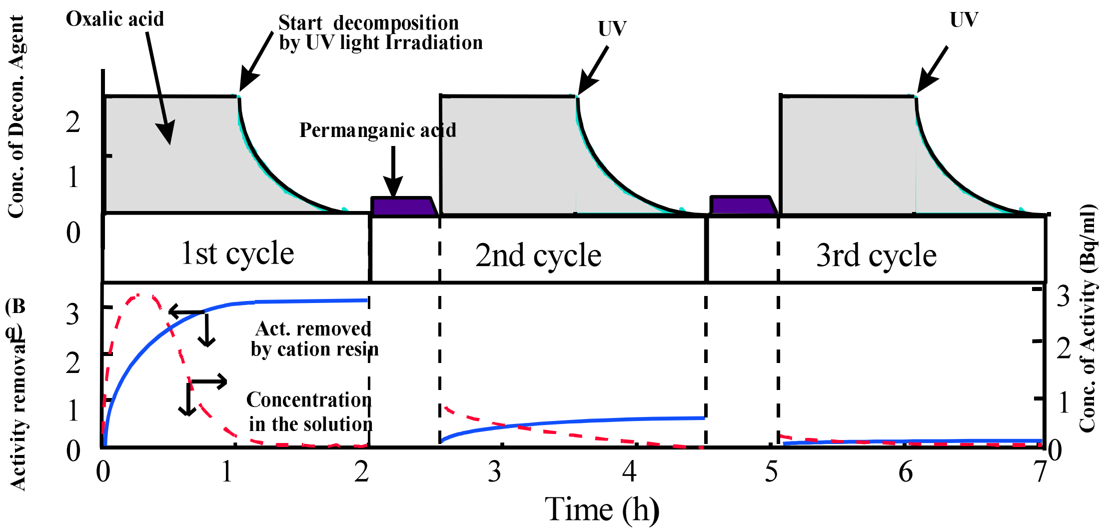

FIG. 1. Decontamination profile of the CORD/UV.

The second cycle started with an oxidation step by use of permanganic acid in order to oxidize the trivalent chromium oxides in the crud to the easily dissolvable hexavalent chromium ion. When the chromium oxide dissolution was no more observed, the permanganic acid was quickly reduced to manganese ions (Mn^{2+}) by oxalic acid injection in the system. In the subsequent decontamination step the oxalic acid was added up to the specified concentration. Manganese ions were removed together with the corrosion products by cation exchange resins after the resin columns started operation in this step. In the final step at the third cycle the oxalic acid in the system is decomposed by wet oxidation using ultraviolet light and the released ions and residual impurities were removed by mixed resin column.

2.1.2. Material compatibility

All relevant materials, being exposed to the decontamination solutions during the full system decontamination, were investigated. The materials, which had been tested, are shown in Table I.

Specific conditions like weld and gaps were considered. The material compatibility tests cover the material behavior during the decontamination as well as the post decontamination operation. The investigations covered all relevant aspects of material compatibility like general corrosion loss, selective corrosion like pitting or intergranular attack and Intergranular Stress Corrosion cracking (IGSCC), as shown in Table II. Some tests and their results are described below.

In the crevice corrosion test at the decontamination temperature (95℃), the prefilmed coupons were investigated after three decontamination cycles from the viewpoints of intrusion into the crevice and a localized corrosion. Besides this inspection, the decontaminated creviced-coupons were immersed in high temperature water (288℃). Then the surface of the coupons were inspected by Scanning Electron Microscope (SEM) and Electron Spectroscopy for Chemical Analysis (ESCA). The results of these corrosion tests indicated that the decontamination solution could not intrude deeply into the narrow gap, and a detrimental effect was not observed on the creviced surfaces. The ESCA analysis showed no difference of carbon and manganese elements density on the surface between the with-decon and the without-decon coupons. This proved that no residual decontamination chemicals remained on the surface.

TABLE I. MATERIAL TO BE TESTED

General Corrosion Test and Crevice Corrosion
– Type 304 stainless steel and its weld metal, CF8M, 630SS (17-4PH), – Alloy600, Alloy182, Alloy82, Alloy X-750, – Colmonoy, Stellite, – Low alloy steel and Carbon steel.
Stress Corrosion Cracking (SCC)
– Type 304 ss (0.08 % C) weld joint, 630SS (17-4PH), CF8M, – Alloy600/Alloy82 weld joint – Alloy600/Alloy182 weld joint, Alloy X-750

TABLE II. CORROSION TESTING

To investigate Corrosion Behavior in 95℃ CORD Solution Environment
– General Corrosion Test – Crevice Corrosion Test
To Investigate Corrosion and SCC Behavior in 288℃ Water after CORD Application
– Crevice Corrosion Test, – Slow Strain Rate Test (SSRT), – Crevice Vent Beam Test (CBB Test) – Crack Growth Rate Test by using CT and WOL specimens (CGR Test)

A slow strain rate test (SSRT) was conducted in 288 °C water containing less than 20 ppm dissolved oxygen in order to confirm IGSCC susceptibility of Type 304 stainless steel. A cylindrical tensile specimen with diameter of 6.0 mm and gage length of 20 mm was machined from the Type 304 stainless steel weld joint. The strain rate was $4 \times 10^{-7}\,s^{-1}$. The comparative tendency toward IGSCC was expressed by the ratio of IGSCC area to that of whole fracture surface. No substantial difference of the comparative tendencies between the coupons with and without decontamination was observed. The result of SSRT indicated that there was no harmful effect on IGSCC susceptibility of Type 304 stainless steel weld joint.

The influence of the CORD solution on the integrity of the structural materials was evaluated by investigating the SCC growth behavior in 288 °C water. The two Alloy182 CT specimens with 5% side grooves on each side surface were heat-treated at 61 °C for 10 hours which was simulated PWHT (Post-Welded Heat Treatment) condition. Both specimens were fatigue-precracked in air. One specimen was subjected to the three cycles of CORD process. These specimens were installed in an autoclave with an electrical hydraulic machine in a circulating loop. The crack growth test was carried out under actively loaded condition in 288 °C water containing less than 20 ppm O_2. The specimens were loaded at an initial K value of 30Mpa m under constant load condition accompanied with periodical unloading of R=0.7, f=0.01Hz every 10,000 seconds. The crack length was monitored by means of reversing d.c. potential drop method (PDM).

Based on the actual crack lengths observed on the specimen fracture surfaces, the crack lengths measured by PDM were corrected and crack growth rates were calculated from the slopes of crack length versus time curve by the least squares method. The crack grew monotonically with time during the test period for both specimens. There is no significant difference in crack growth rate of both specimens. Therefore, it is concluded that no adverse effect of decontamination by the CORD process was observed on the SCC growth behavior of Alloy182 exposed to a BWR reactor water environment.

As a result of the above-mentioned investigations, it has been verified that the CORD process has no detrimental effects on the integrity of the materials coming in contact with the decontamination solutions, as summarized in Table III.

2.2. Application of decontamination

According to the developed sequence of the core internal replacement project, the FSD was first performed before starting work inside the RPV. Then, the existing reactor

TABLE III. SUMMARY OF MATERIAL COMPATIBILITY TEST RESULT

Material	Results
Type 304 ss, CF8M 630SS (17-4PH) Alloy600, Alloy182, Alloy82, Alloy X-750 Colmonoy Stellite Low alloy steel , Carbon steel and C-Steel casting	No detrimental effects were observed

core internals were cut and removed from the RPV top, and the in-vessel shielding was installed. After establishing the necessary environment in air, the workers entered into the RPV to restore the reactor core internals.

2.2.1. Chemical decontamination

2.2.1.1. Scope of decontamination and the decontamination system

The scope of the FSD involved the RPV including the core shroud, the jet pumps, the core plate, the top guide, the core spray sparger, the feed water sparger as well as the two reactor recirculation loops, as shown in Figure 2. Prior to decontamination, the reactor internals such as the dryer, the separator, the fuel assemblies, the control rods and the control rod guide tubes were taken out from the reactor vessel according to the routine procedure. The systems such as the residual heat removal system (RHR), the reactor water clean-up system (RWCU), the core spray line (CS), the feed water line (FW), which are directly connected to the RPV, were isolated by closing the isolation valves and/or the mechanical plugs. The total area to be decontaminated was 1150 m^2, and the volume 360 m^3.

A forced circulation of the decontamination solution in the RPV is necessary in order to sufficiently dissolve the crud. Therefore the RRS pumps were operated for the forced circulation of the reactor water. A part of decontamination solution was extracted from the CRD housings / ICM housings in the RPV bottom, and was injected to the RPV wall through the temporary spray ring, which was installed between the RPV and the RPV head. Fig. 3 shows the outline of decontamination loop.

The spray ring served to contact the upper wall of the RPV with decontamination solution. By this the solution volume was reduced, resulting in a volume reduction of waste. The decontamination equipment, including the spray ring, consisted of pumps, ultraviolet (UV) skid, ion exchange resin skid, electric heater, cooler, etc. The ion exchange resin columns were designed to be covered by the lead shielding to reduce the radiation dose rate less than 2 mSv/h on the surface and less than 0.1 mSv/h at 1 m distance from the surface.

2.2.1.2. Actual decontamination schedule

The system to be decontaminated was filled with demineralized water and heated to around 95 °C. The heat sources to raise and maintain the temperature of 95°C were two 600 kW external heaters and the joule energy of the RRS pumps. The decontamination chemicals were injected according to the specified concentration after heating to 90-95 °C. Then according to the CORD procedural sequence, the additional chemicals were injected into the system without intermediate rinse or change of water.

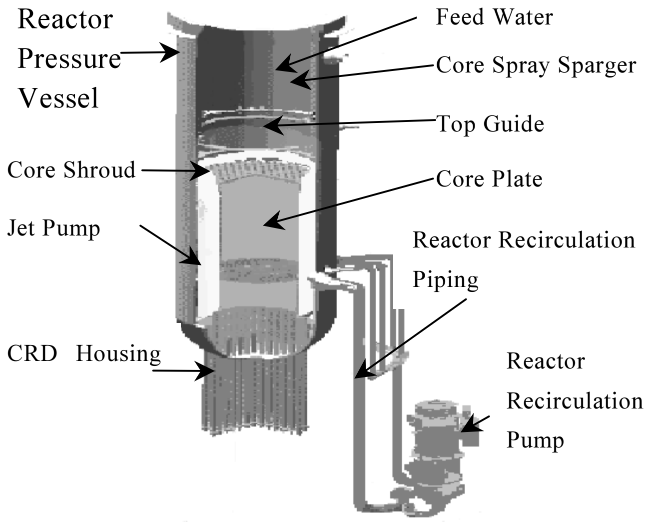

FIG.2. Scope of FULL system decontamination at 1F-3.

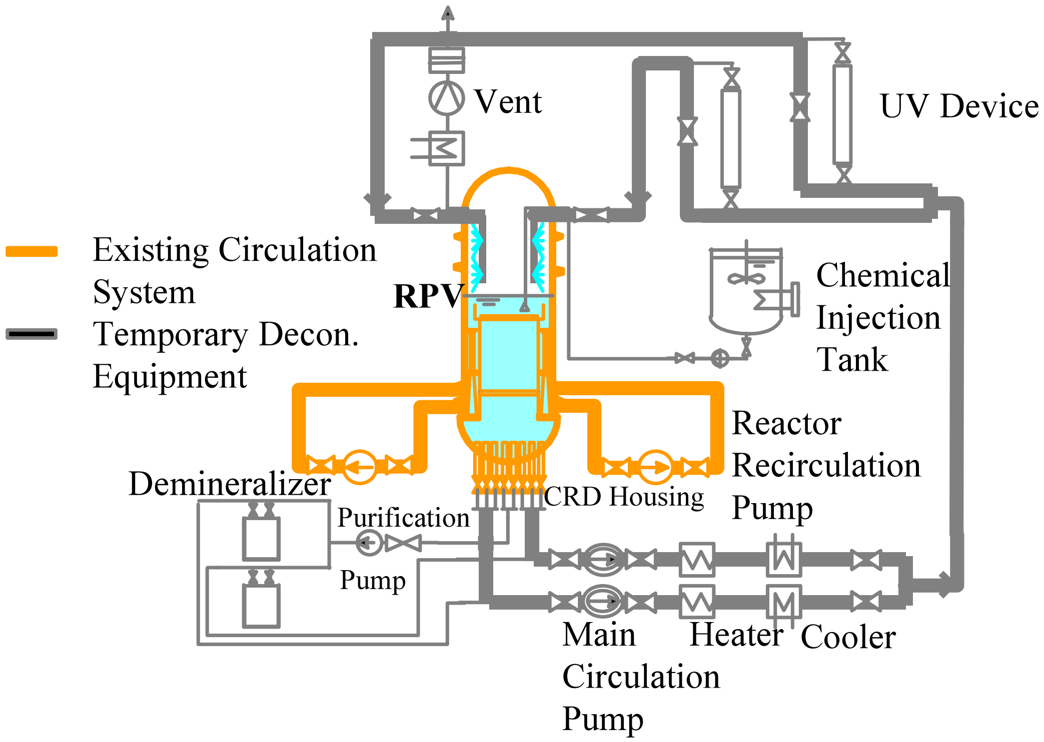

FIG.3. 1F-3 Full system decontamination flow.

The FSD was performed with three decontamination cycles, according to the preceding investigations. After the decontamination, all the isolated pipes and the small instrument piping were back-flushed into the RPV by make-up water. Furthermore the water was purified by mixed bed resin columns, and the absence of residual chemicals was confirmed. Three decontamination cycles and the final purification were performed in 168 hours at unit 3 and in 179 h at unit 2 nearly as planned. It was proved that even extensive decontamination can be completed in a week.

2.2.1.3. Dose rate for chemical decontamination

The total radiation exposure for FSD was 0.2 man-Sv for each unit, one third (1/3) of which was due to the decontamination operation and two third (2/3) was due to the preparation and restoration work. The dose rate was kept low during the decontamination, since the dissolved radioactivity was removed by the resin beds and the decontamination solution was regenerated, the dose rate was maintained low.

2.2.1.4. Waste generation

The volume of necessary ion exchange resins was estimated by preliminary evaluation of the crud inventory. Four resin columns were located on the first floor of the reactor building. One or two columns were used during the decontamination and the other columns were stand-by. Cation exchange resin columns were in operation during the decontamination cycles in order to remove activity and metal ions dissolved in the decontamination solution. Finally the mixed resin columns removed residual small amounts of impurities to meet the requirements for water polishing. Used resin was discharged as slurry to a spent resin tank after each decontamination cycle, then new resin was filled in.

The nine batches resulted in 5.4 m^3 of resin waste generated at unit 3, and the 7 batches (4.2 m^3) at unit 2. This resin volume was in accordance with the calculated numbers based on the crud inventory and consumed permanganic acid. This is an extremely small volume in comparison with the decontamination solution volume of 360 m^3.

2.2.2. Mechanical cleaning

2.2.1.1. Mechanical cleaning procedure

The reactor internals being replaced were cut out from RPV after the FSD. The residual insoluble crud and some radioactive cutting particles remaining had to be removed to improve the working conditions in the RPV. Normal cleaning methods such as brushing, suction and water jet were extensively applied for the removal of these active solids from all the inner surfaces of the RPV, especially from the horizontal parts like bottom and baffle plate where the insoluble solids had settled. After the CORD decontamination the residual crud is loose and easy to be cleaned. To clean effectively every part of inner area, some special devices were constructed to fit to the complicated shape, as shown in Figs. 4 and 5.

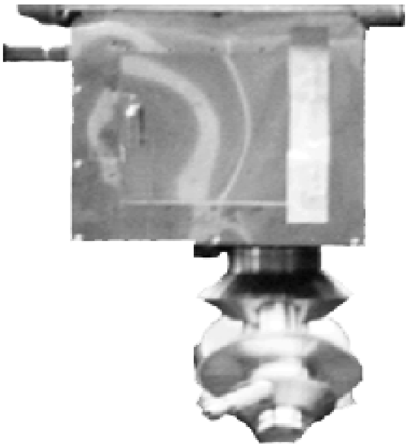

FIG.4. Water jet cleaning device. Nozzle rotation method. Air- motor driven.

FIG.5. High -pressure jet cleaning device nozzle rotation method: Water – driven.

Such mechanical cleanings were carried out twice or more to all the inner parts of the RPV. The main cleaning procedures are shown bellow and illustrated in FIG.6.

(a) Suction cleaning under water (reactor well, RPV wall, bottom & baffle plate)
(b) Brush cleaning (reactor well, RPV wall & bottom)
(c) Water drain & cleaning (reactor well)
(d) Reactor water drain
(e) Water jet lancing in the air (RPV wall, bottom & baffle plate)
(f) In-vessel shielding set-up under water
(g) High-pressure water jet lancing in air (RPV bottom)
(h) Final rinsing (RPV bottom & baffle plate)

2.3. Decontamination result

2.3.1. Chemical decontamination results

2.3.1.1. Activity and metal removal

Radioactivity and metal concentrations in the decontamination solution were measured every hour at the inlet and the outlet of ion exchange columns. The evaluation of measurements led to the total release rates from the contaminated surface. The total removed activity was approximately 10 TBq at unit 3, as shown in Fig. 7. The ratios of activity removal in the decontamination cycles were 90% in the 1st cycle, 8% in the 2nd cycle and 2% in the 3rd cycle. The dominant nuclide of the removed activity was ^{60}Co, followed by ^{54}Mn, ^{58}Co, ^{51}Cr, ^{59}Fe and ^{65}Zn. The removed oxide was 72 kg as metal at unit 3, 71% in cycle 1, 19% in cycle 2 and 10% in cycle 3, as shown in Fig. 8. The elemental distribution was 91% of Fe, 7% of Ni and 2% of Cr. The other metals were less than 1%.

From metal composition the removed crud almost was the iron oxide, which was easily dissolved by decontamination chemicals. After the first cycle, more than 90% of radioactivity was dissolved from the contaminated surface. And it was confirmed that almost the radioactivity was dissolved and removed by the three decontamination cycles. Therefore, it was confirmed from the result, that three decontamination cycles were sufficient for the FSD.

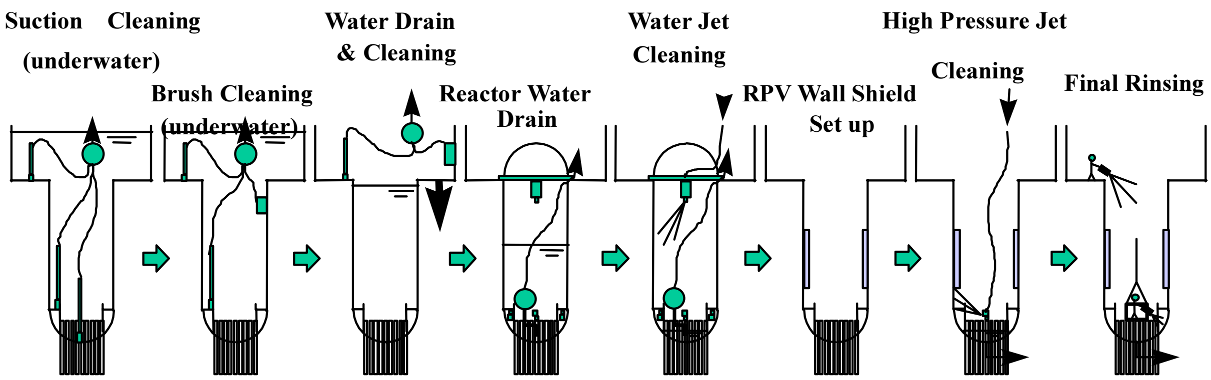

FIG.6. Sequence of the mechanical cleaning at 1F-3.

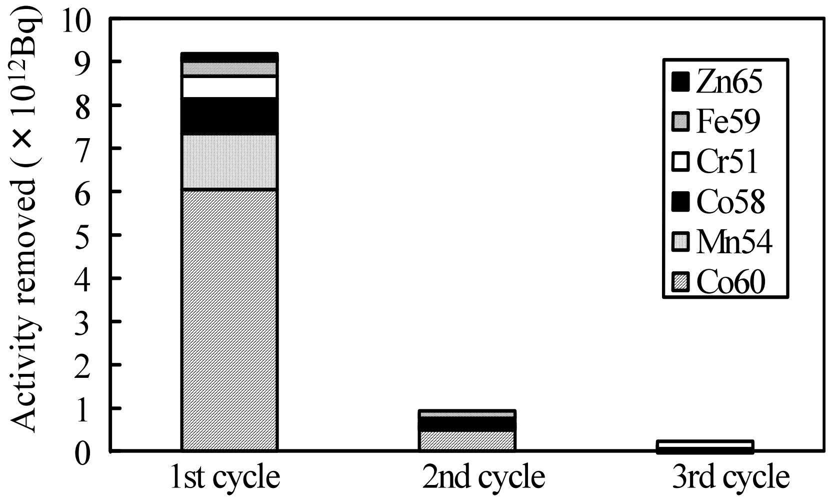

FIG. 7. 1F-3 full system decontamination release of activity.

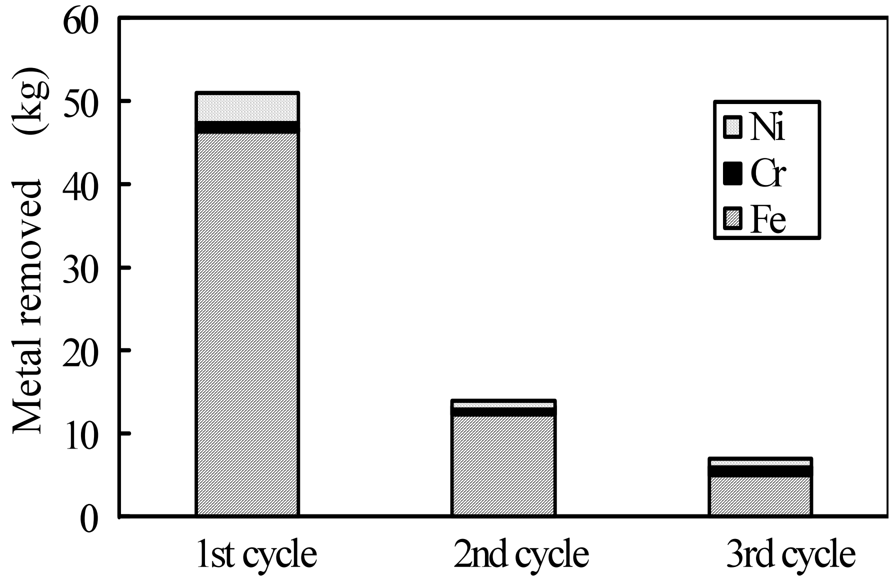

FIG. 8. 1F-3 full system decontamination release of metals.

In the case of the unit 2 (1F-2) FSD, 10 TBq of radioactivity and 63 kg of crud as metal was dissolved and removed during the three decontamination cycles. About 90 % of radioactivity also was removed in the first cycle, almost all metal oxide was dissolved in the three decontamination cycles. Unit 2 (1F-2) was operated under the Hydrogen Water Chemistry (HWC) condition for approximately one year a little longer than at unit 3 (1F-3) . However no negative effect by HWC was observed on the decontamination efficiency.

2.3.1.2. Dose rate reduction

The dose rate in the RPV was continuously monitored by three high temperature scintillation counters installed at the top of three CRD housings, arranged from the center to the RPV wall. The dose rate sensors indicated under water 2 to 8 mSv/h before the decontamination, and 0.1 mSv/h after decontamination. Figure 9 illustrates the locations of the sensors and the measurement results of unit 3 (1F-3) before and after the decontamination. The average decontamination factor (DF) at the vessel bottom was 43. This DF was higher than the target value of 20 and confirmed the completion of the FSD after three decontamination cycles.

Additionally the contact dose rate on the reactor recirculation system was measured at 20 locations by an ion chamber monitor (ICM) at the end of each cycle. For the recirculation piping the dosimeter indicated 0.9 - 2.8 mSv/h (average 1.6 mSv/h) before the decontamination, and 0.02 - 0.13 mSv/h (average 0.06 mSv/h) after the decontamination, as shown in Fig. 10. The average DF of 46 for the outer surface of the recirculation piping was more than the target of 20. The γ-scan by Ge detector with tungsten collimator and multi-channel pulse height analyzer on the adhered inner surface of recirculation piping led to a DF of 72, as summarized in Fig. 11. Consequently the remarkable dose rate reduction in the dry-well by a factor of 5 was also achieved as well as in the RPV.

At unit 2 (1F-2) by the three cycles decontamination, the decontamination factor reached DF 108 at the RPV bottom and DF 68 for the reactor recirculation system.

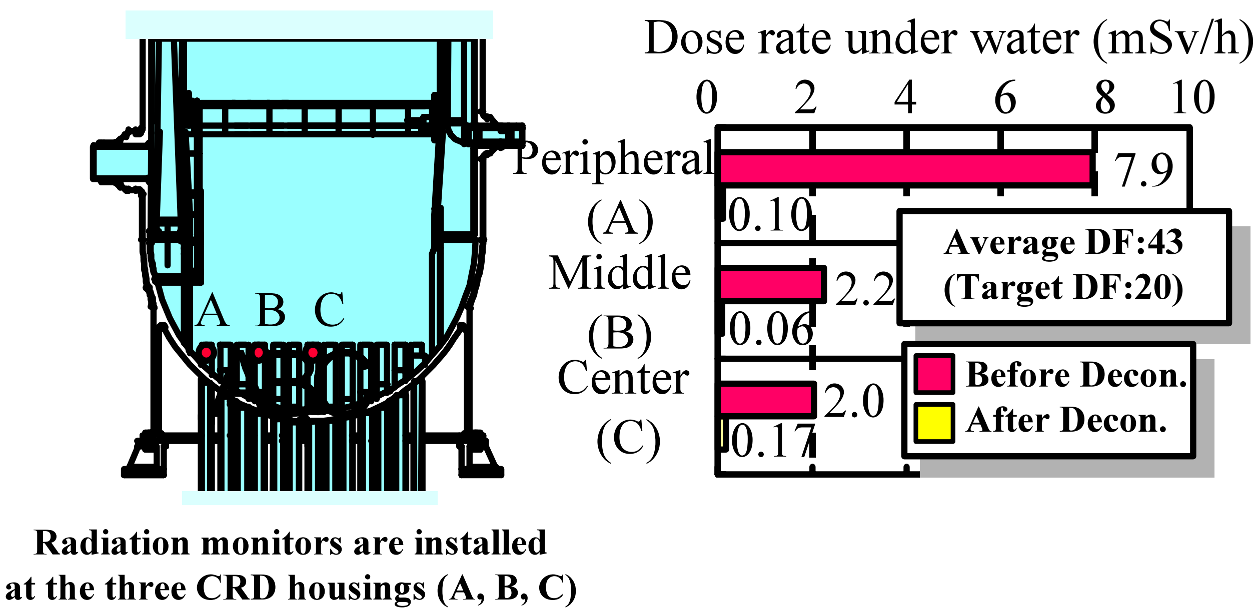

FIG. 9. 1F-3 full system decontamination. Dose rate at the RPV bottom.

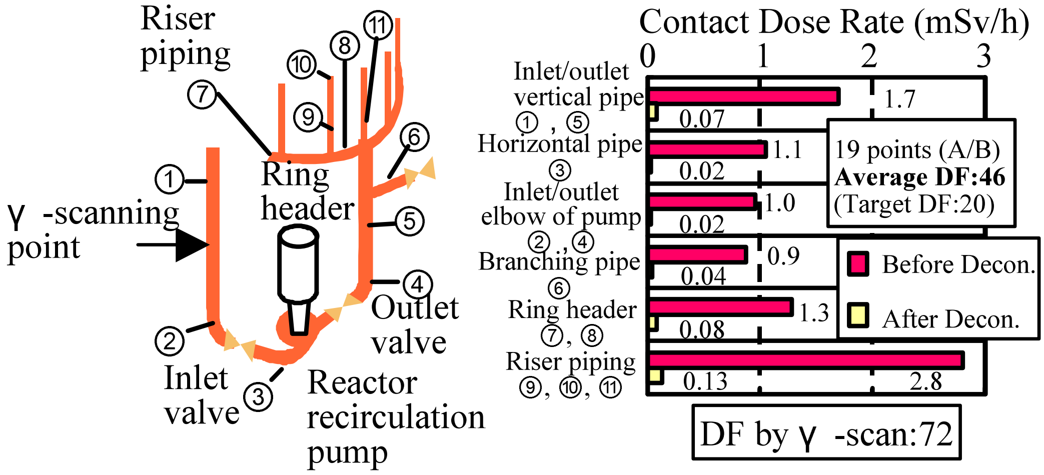

FIG. 10. 1F-3 full system decontamination. Contact dose rate of reactor recirculation piping.

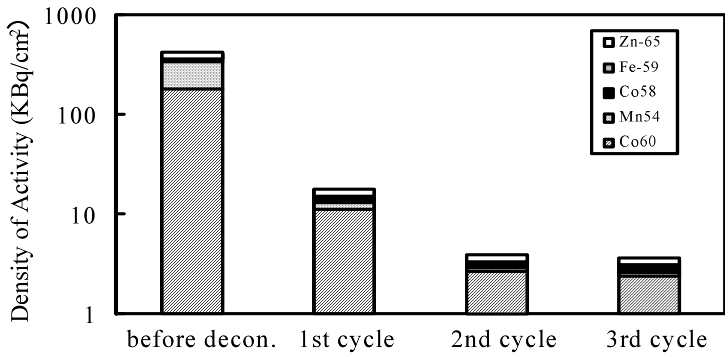

FIG.11. 1F-3 full system decontamination. γ - scanning the inlet reactor recirculation piping.

2.3.2. Cleaning effect

After all the mechanical cleaning, the dose rate of RPV bottom was measured by the underwater dose rate measuring devices. The result is shown in Fig. 12. The RPV bottom dose rate, 0.1 mSv/h, after the FSD decreased to 0.03 mSv/h under the water together with shielding, the resulting total DF was 160. After draining the rector water, the dose rate in the air was 0.2 mSv/h, low enough for human access inside the RPV.

2.3.3. Dose saving

The dose reduction was estimated for the entire shroud and core internals replacement project. In this estimation the effect by the in-vessel shielding and the mechanical cleaning was subtracted. The dose rate savings by the FSD were 70 and 140 man*Sv respectively at 1F-3 and 1F-2.

3. SUMMARY

- The Full System Decontamination by the CORD UV process resulted in the removal of 10 TBq of activity and 60-70 kg of metal oxides respectively at 1F-3 and 1F-2.
- The average decontamination factors are 43-106 at the RPV bottom and 46-68at the RRS.
- After the FSD the underwater dose rate of RPV bottom was 0.11 mSv/h.
- The waste generated was only 4.2-5.4 m^3 of ion exchange resins in each full system decontamination.
- A combination of mechanical cleaning methods was very effective to remove residues and other insoluble solids after the FSD,
- After mechanical cleaning the underwater dose rate at the RPV bottom was 0.03 mSv/h.
- The average dose rate in air at the RPV bottom was finally reduced to less than 0.2 mSv/h after mechanical cleaning and in-vessel shielding.
- As result a dry condition inside the RPV was realized, and the preventive maintenance work was successfully performed.
- The dose rate saved by the FSD in the core shroud replacement project was calculated to be 70 and 140 man *Sv respectively at 1F-3 and 1F-2.

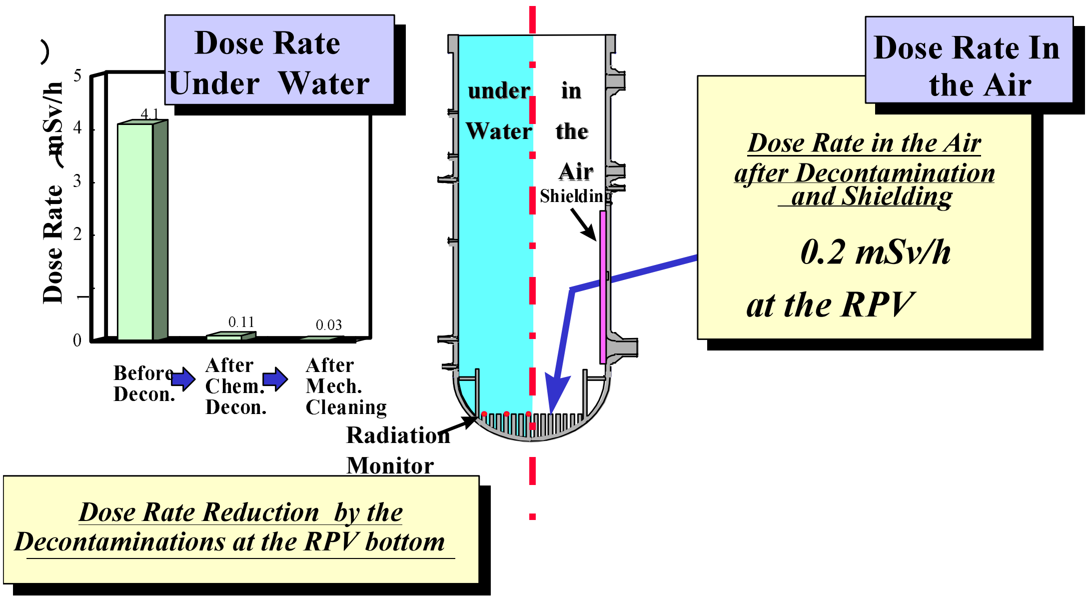

FIG. 12. 1F-3 full system decontamination and mechanical cleaning dose rate at the RPV bottom.

4. CLOSURE

The Full System Decontamination by the CORD UV process has attained a remarkable reduction of dose rate first at 1F-3. It took three years of planning, studies, and evaluation. In the 1F-2 FSD, the same result was obtained also by the CORD UV process as at 1F-3. Similar projects are now planned at several nuclear plants in Japan. Decontamination is an essential technology to facilitate successful replacement of the reactor internals. Authors hope this experience will be useful for the successful performance of the projects.

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APPLICATION OF WATER JET PENNING TECHNOLOGY TO BWR CORE SHROUD FOR IGSCC MITIGATION

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Abstract

Water Jet Peening (WJP) is one of the promising SCC mitigation technologies which make original surface tensile residual stress to compressive one. The Water Jet Peening Technology has the following advantages: a)no foreign material entering into the reactor because of using only water, b) applicability to narrow and complicated structure because it is effective in the wide range of parameters, c) simple in the system/equipment and short period of application in actual plant. WJP was first applied to BWR Core Shroud for preventive maintenance purpose during 1999 outage in Japan. Although the target welds of Shroud are surrounded by various kinds of other components and access space is very limited, most of the weld could be peened by optimizing the peening condition. Effect of residual stress improvement was verified by mock-up test prior to actual work. WJP application was completed within the planned schedule without trouble. Application experience to the Shroud and examples of development of application to other Reactor Internal components will be presented.

1. INTRODUCTION

Long-range plan of preventive maintenance is the important subject for economical plant life extension. For Reactor Internal components, application of mitigation technologies for material degradation is the key issue for the plant life management. Stress Corrosion Cracking (SCC) is one of the degradation mechanism which occurs on the Reactor Internal components in BWR.

SCC is known to occur if material factors, stress factors and environmental factors overlap to each other. SCC mitigation technologies have been developed to be exempted from these factors.

At several BWR plants, Core Shroud has experienced SCC at heat affected zone of the weldments. Thus, application of preventive maintenance technologies to the weld lines has been necessary.

For more than ten years, we have been developing Water Jet Peening (WJP) technology which makes surface residual stress from tensile to compressive. As described in the next chapter, this technology has several benefits. Using the benefits, the WJP technology has been applied to BWR Core Shroud at Japan Atomic Power Company Tokai-2 plant in 1999 outage. In this paper, the outline of the WJP technology and the actual application to the Core Shroud are described. Applications of the technology to other parts of internals are also introduced.

2. OUTLINE OF THE WJP

A schematic mechanism of residual stress improvement by WJP is shown in Fig. 1. The cavitation bubbles are collapsed after they are reduced to minimum diameter. It is said that the collapse pressure reaches to 1000 MPa and the propagating velocity of pressure wave exceeds the sound speed.

The water jet with extremely high velocity causes turbulent flow with a lot of cavitation bubbles which collapse at the impinged surface and produce high impact pressure on the material surface. Due to the high pressure the metal surface is extended in the horizontal direction, then the compressive residual stress is produced by the elastic constraint surrounding the extended. Since the cavitation bubbles are very small, the compressive residual stress is considered to be formed just in the sub-surface of the material.

The WJP technology has the following advantages:

a) no foreign material entering into the reactor because of using only water,
b) applicability to narrow and complicated structure because it is effective in the wide range of parameters,
c) simple in the system/equipment and short period of application in actual plant.

The residual stress on surface and subsurface of type 304 stainless steel were measured [1]. The distribution of residual stress perpendicular to the nozzle scanning direction and in the subsurface are shown in Fig. 2. The residual stress turned to compressive up to the depth of about 400 μm. The width in which WJP is effective ranges about 60-70 mm. It is found by exprimet that the residual stress improvement is also effective to Ni-base alloys [2].

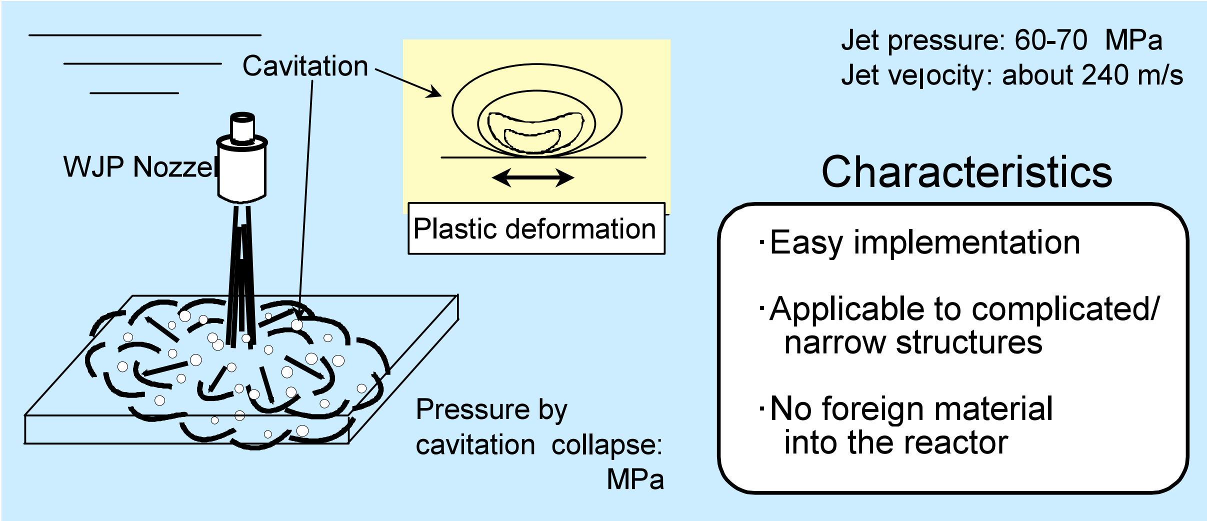

FIG. 1. Schematic illustration of the mechanism of WJP.

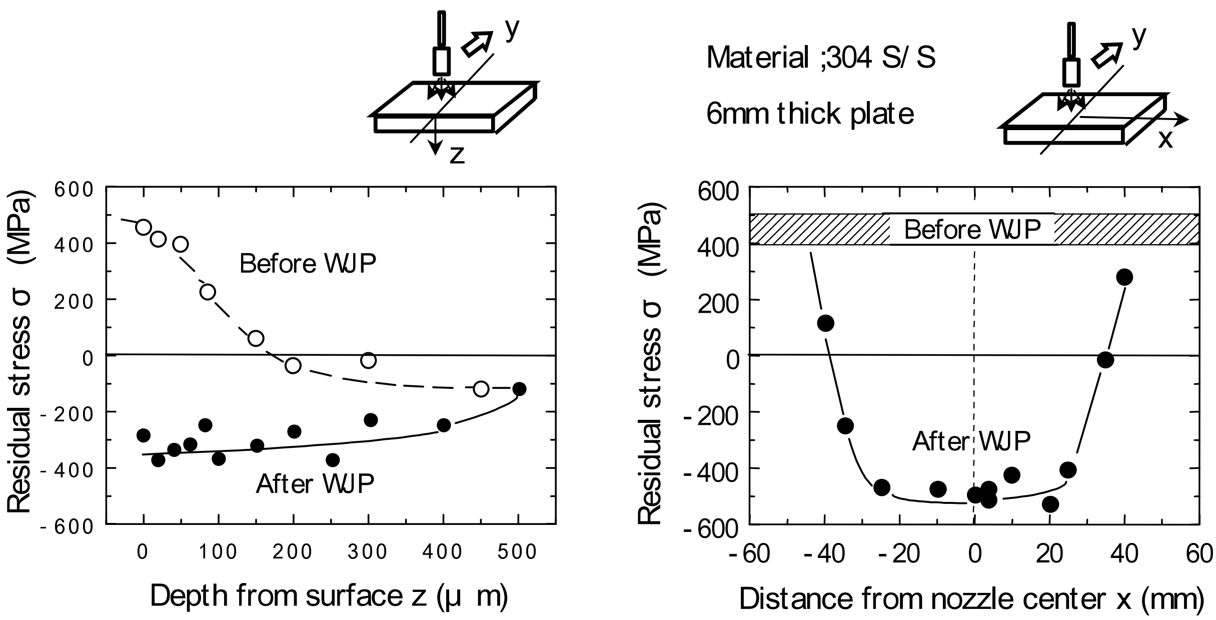

FIG. 2. Residual stress distribution before/after WJP(304S/S)[1].

The effect of WJP to improve SCC resistance in high purity BWR water environment is tested and evaluated by Creviced Bent Beam test [3]. The results are shown in Fig. 3. Intergranular crack was not observed on test pieces with WJP, whereas cracks were found on ones without WJP. The results demonstrate that WJP is useful in preventing SCC initiation. The effect was also confirmed on welded pieces and irradiated materials as well as the base metal [2].

Several tests were conducted to investigate effects of WJP on corrosion and surface roughness and to verify its durability [2, 4]. Table I summarises the results. It is confirmed that WJP has no harmful effect on materials and the WJP effect is expedted to continue during plant operation.

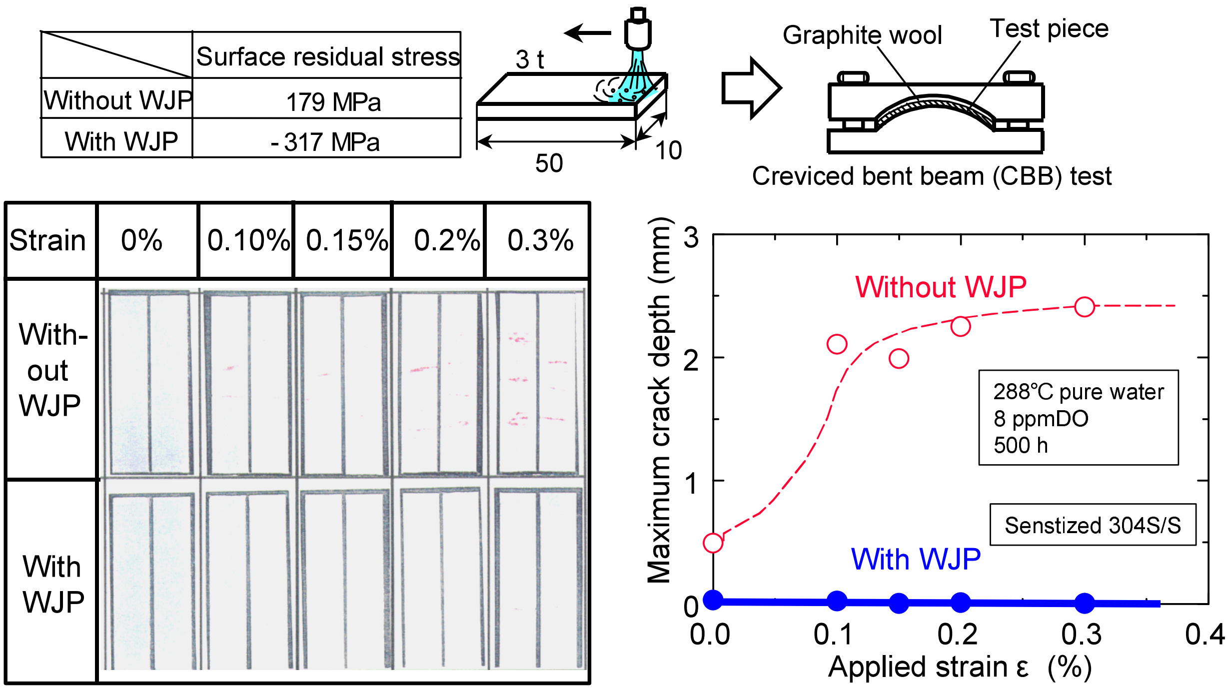

FIG. 3. WJP effects on SCC mitigation (Sensitized 304S/S)[3].

TABLE I. SUMMARY OF EFFECT OF WJP ON MATERIALS AND THE DURABILITY

Item examined	Results
Surface hardness	Increase in hardness on the WJP is moderate
Surface roughness	No effect on surface roughness
General corrosion	General corrosion of the WJP surface is negligible
Stress relaxation	Stress relaxation under BWR operating temperature is estimated to be small

3. EXPERIENCE OF WJP APPLICATION TO BWR CORE SHROUD

In Japan, the WJP technology was first applied to BWR Core Shroud at Japan Atomic Power Company Tokai-2 plant in 1999 outage. Although Core Shroud is made of 304L stainless steel and there is no experience of SCC, WJP was applied for the preventive maintenance purpose.

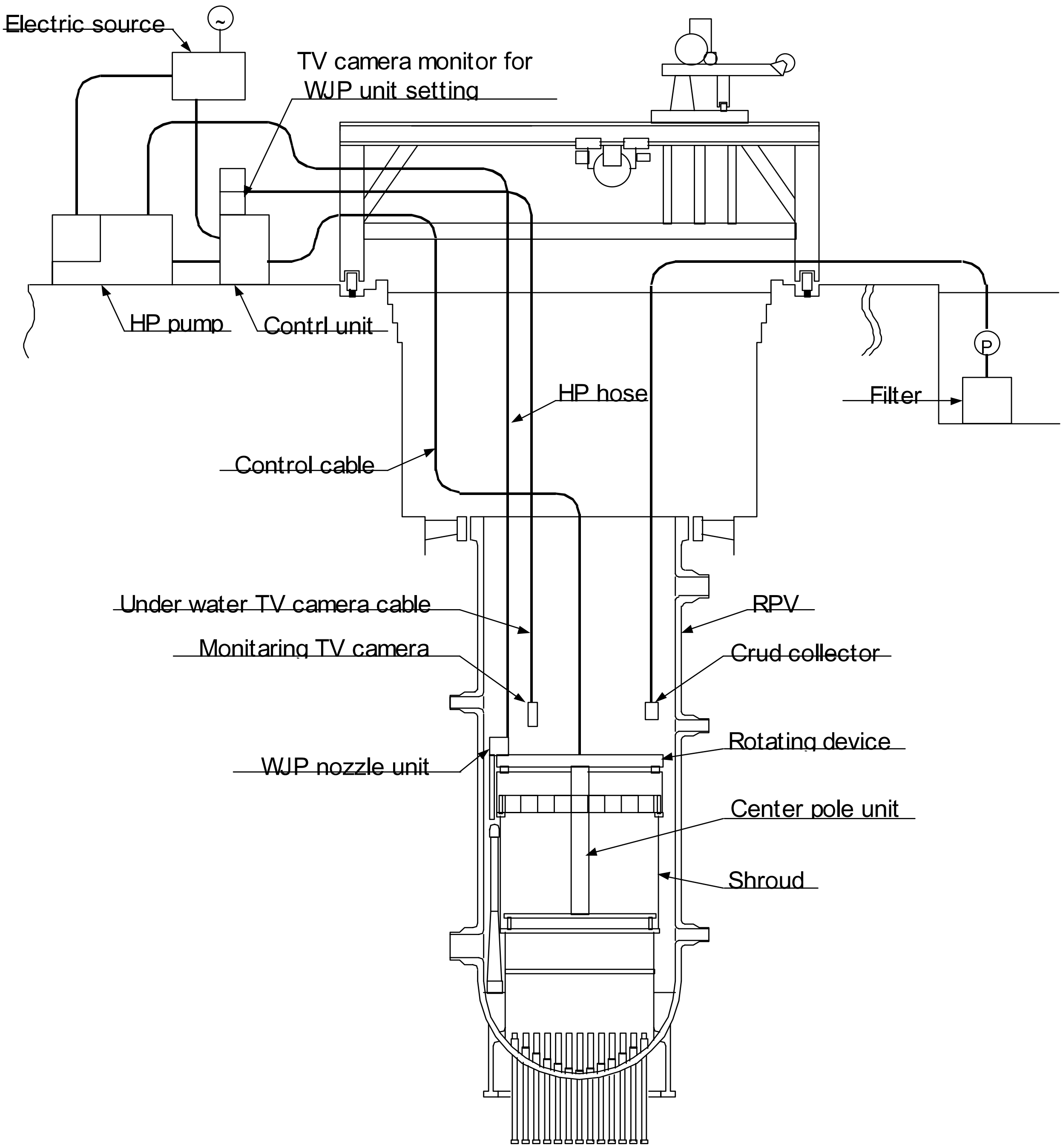

FIG. 4. Arrangement of WJP equipments for Core shroud.

Arrangement of WJP equipments for Core Shroud is shown in Fig. 4. HP pump, control unit, monitoring equipment etc. are installed on the operating floor and WJP nozzle units are installed on the remote handling device and lowered under the water.

Fig. 5 shows photo of the installation work of the remote handling device on the work platform.

Fig. 6 shows the nozzle handling devices. Center pole unit was installed on the Core Plate and Top Guide and arms were spread under and over the Top Guide . The arms can rotate circumferentially with nozzle unit on them. Typical figure and photo of the nozzle for WJP on the Shroud flange weld are also shown in Fig. 6.

Comfirmation of the WJP treatment was done by using under water TV camera after the application to each weld. As soft clud was removed on the metal surface after the WJP treatment, the peened surface became glossy and easily distinguish from the untreated surface.

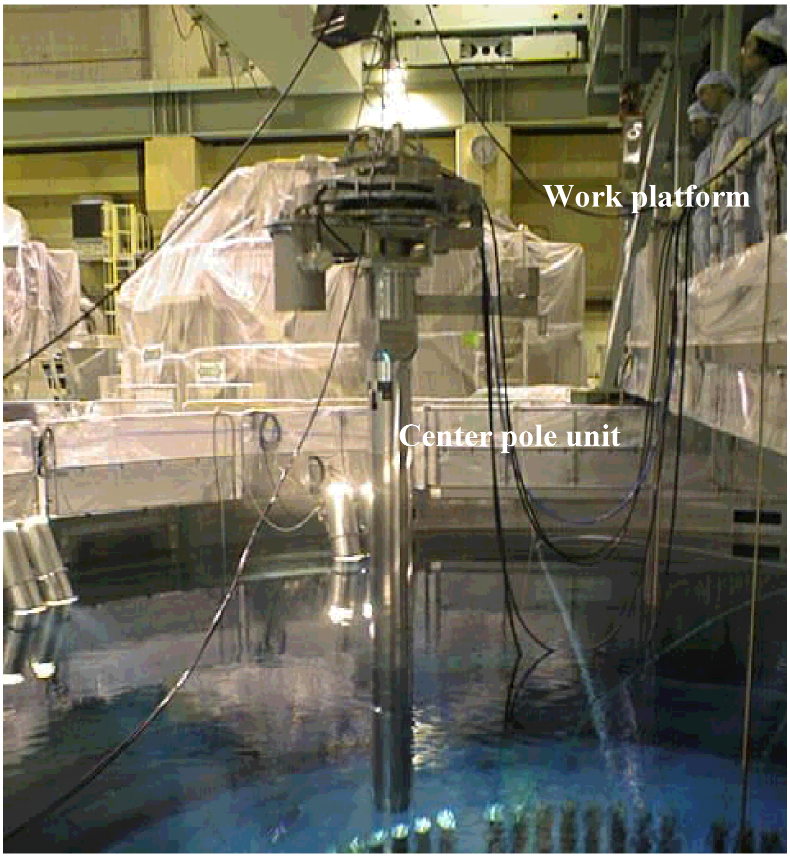

FIG. 5. Installation work of the center pole unit.

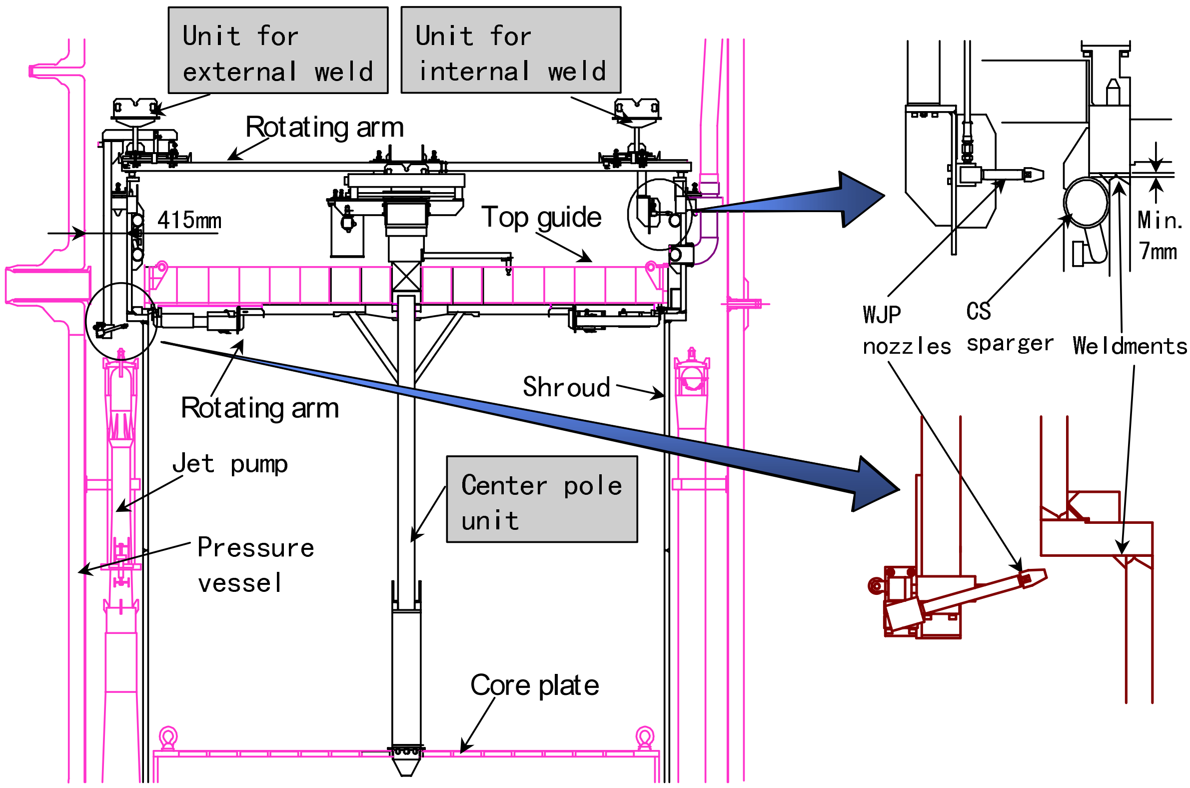

FIG. 6. Nozzle Handling Device.

4. EFFECTIVENESS OF WJP ON CORE SHROUD WELDS

The welds of the Core Shroud to be treated by WJP have various configurations and restrictions to access. By above reasons, optimization of the peening parameters and effectiveness of WJP were confirmed by mock-up test before actual application.

Fig. 7 shows the example of WJP on flange welds behind pipe (Core Spray Sparger). Cavitation jet flew into the small gap between Core Spray Sparger and flange. The residual stress was improved to become compressive over wide ranges beyond weld heat affected zone. WJP was perfomed twice in different injection angles to achieve the wide treated area.

Fig. 8 shows the WJP on rectangular fillet weld . By injecting on side plate in 45 degree, residual stresses on both side plate and bottom plate can be improved at the same time due to the turning flow of the cavitation jet. Fig. 9 shows the residual stress distribution of the rectangular fillet weld after WJP. It is shown that residual stress becomes compressive more than 40 mm from weld on both plates.

5. APPLICABILITY OF WJP TO THE OTHER INTERNALS

WJP can be used for many internals as well as the Shroud due to the wide range of applicability. Te present state of the develpoment is shown in Fig. 10. There are pipe shape interanal components such as ICM Housing, CRD Housing and Jet Pump Diffuser.

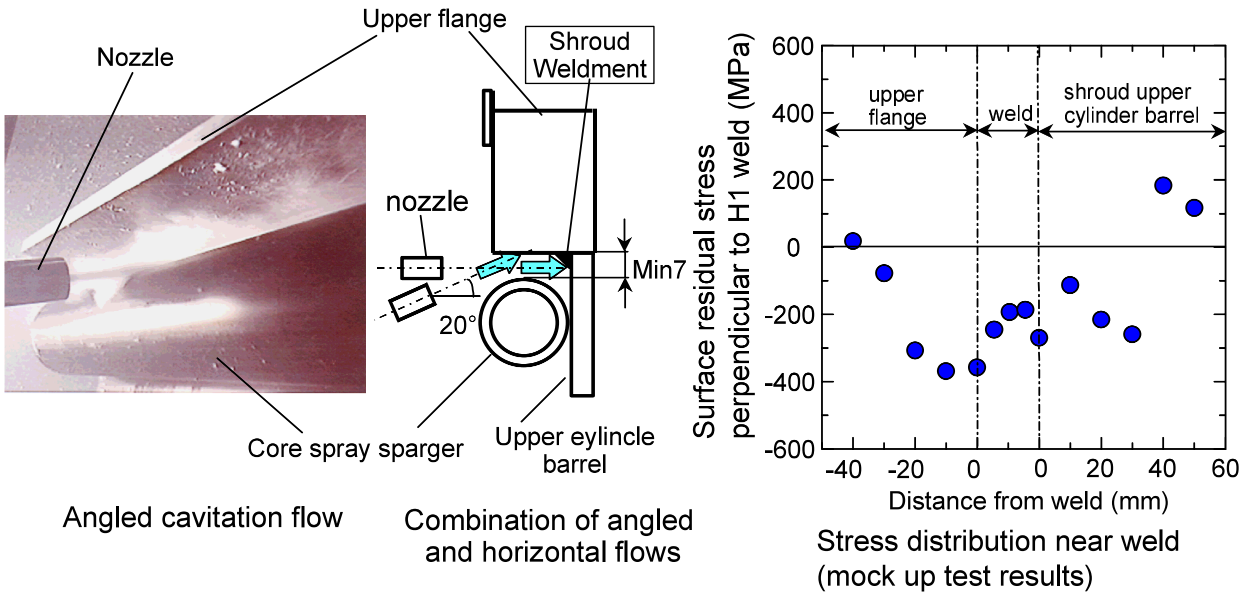

FIG. 7. WJP on narrow weld and residual stress distribution of the weld after WJP.

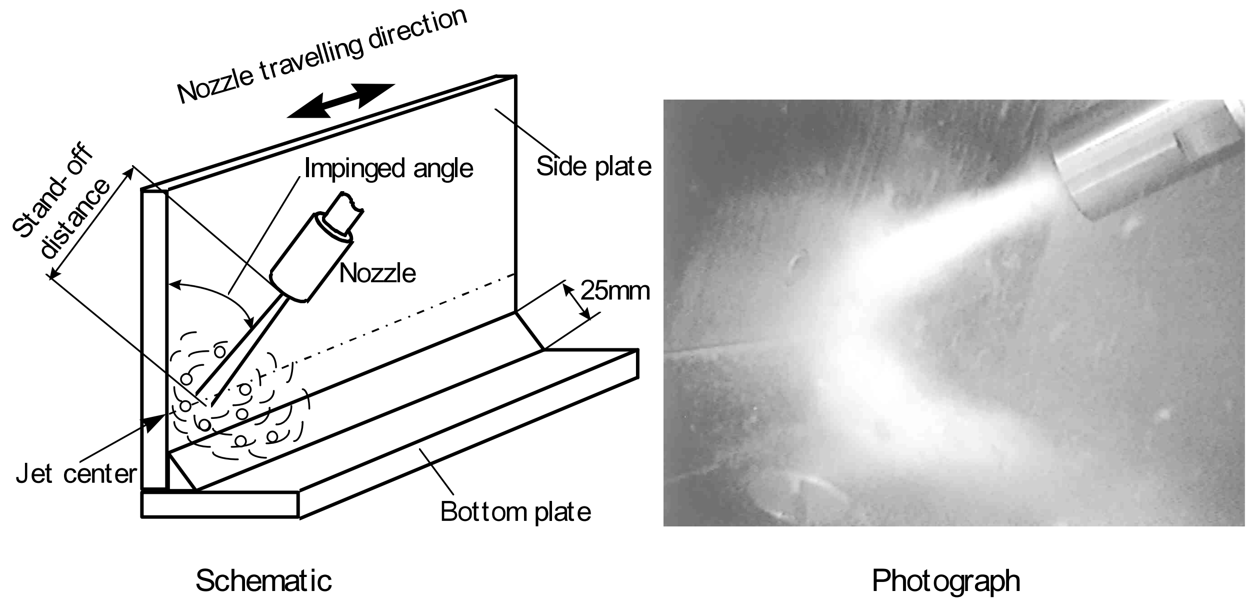

FIG. 8. WJP on rectangular fillet weld.

For the application of thses pipe shape components, tangential flow peening and nozzle with flow buffle have been developed. Fig. 11 shows schematic example of the nozzle with flow bufflle [5]. The flow buffle plate produces complex reflected flow and cause wide range of compression stress. Fig. 12 shows the results of tangential flow peening for external treatment of large diameter pipe [6]. Tangential flow peening can treat wider range of the pipe simultaneously.

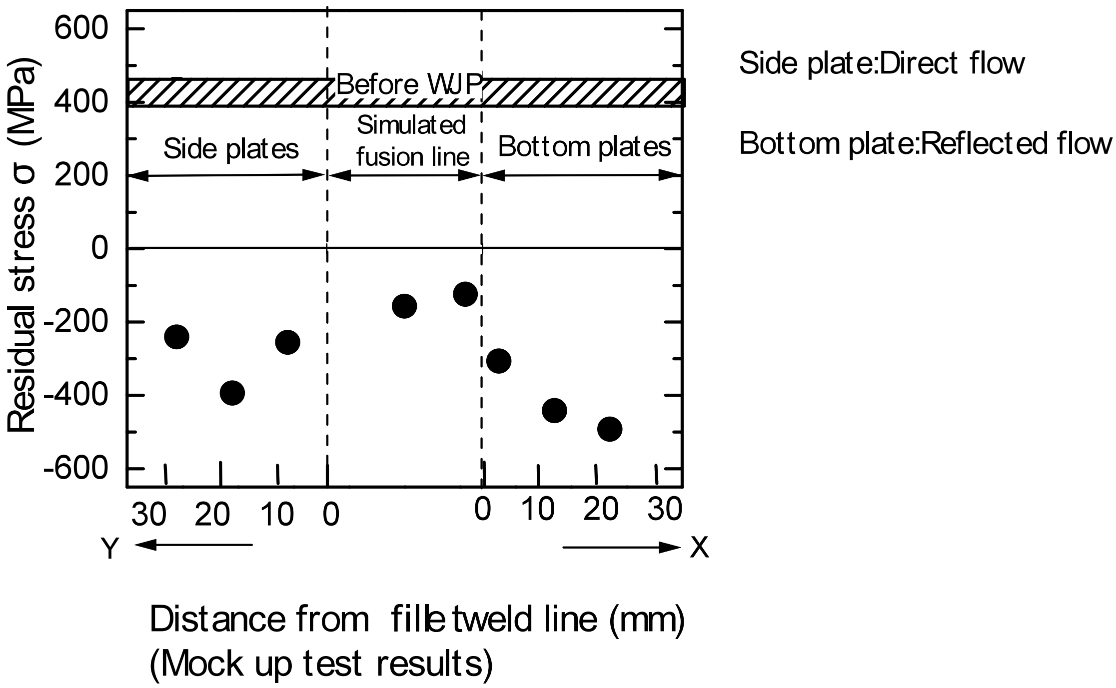

FIG. 9. Residual stress distribution of the rectangular fillet weld after WJP.

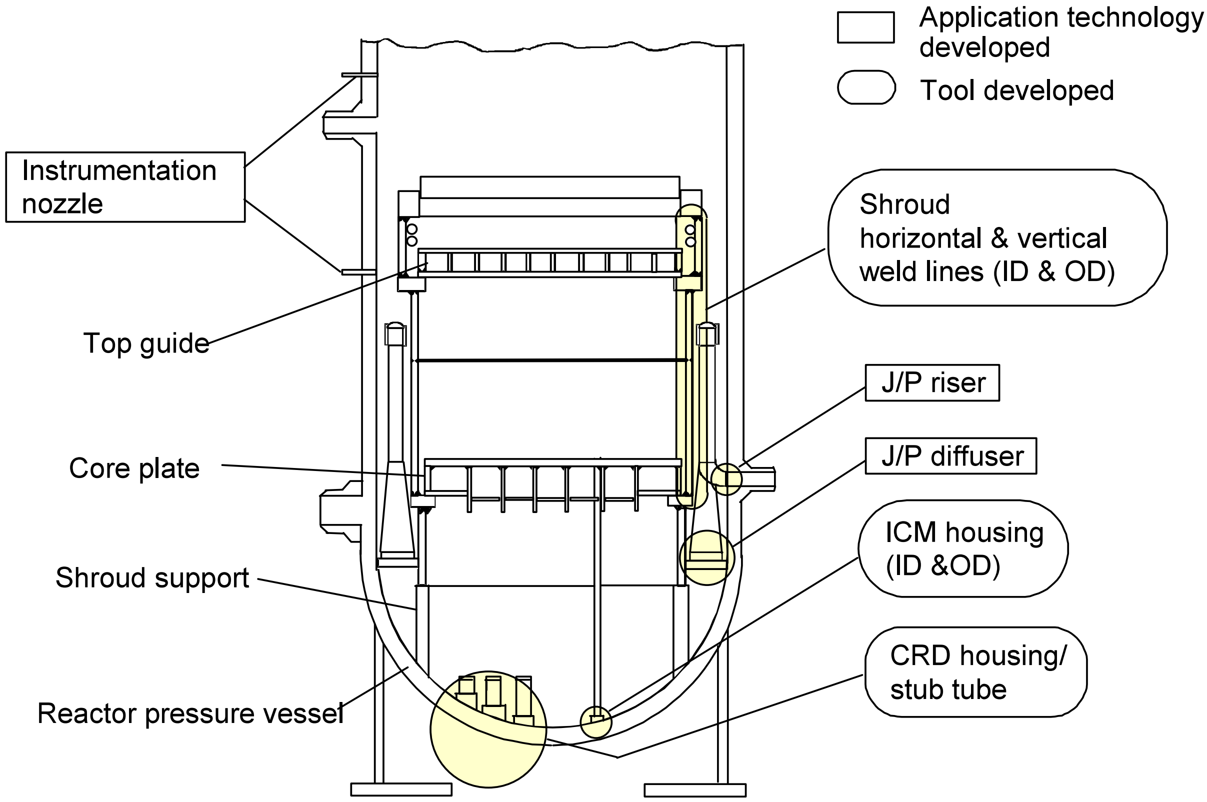

FIG. 10. Applicability of WJP to BWR internals.

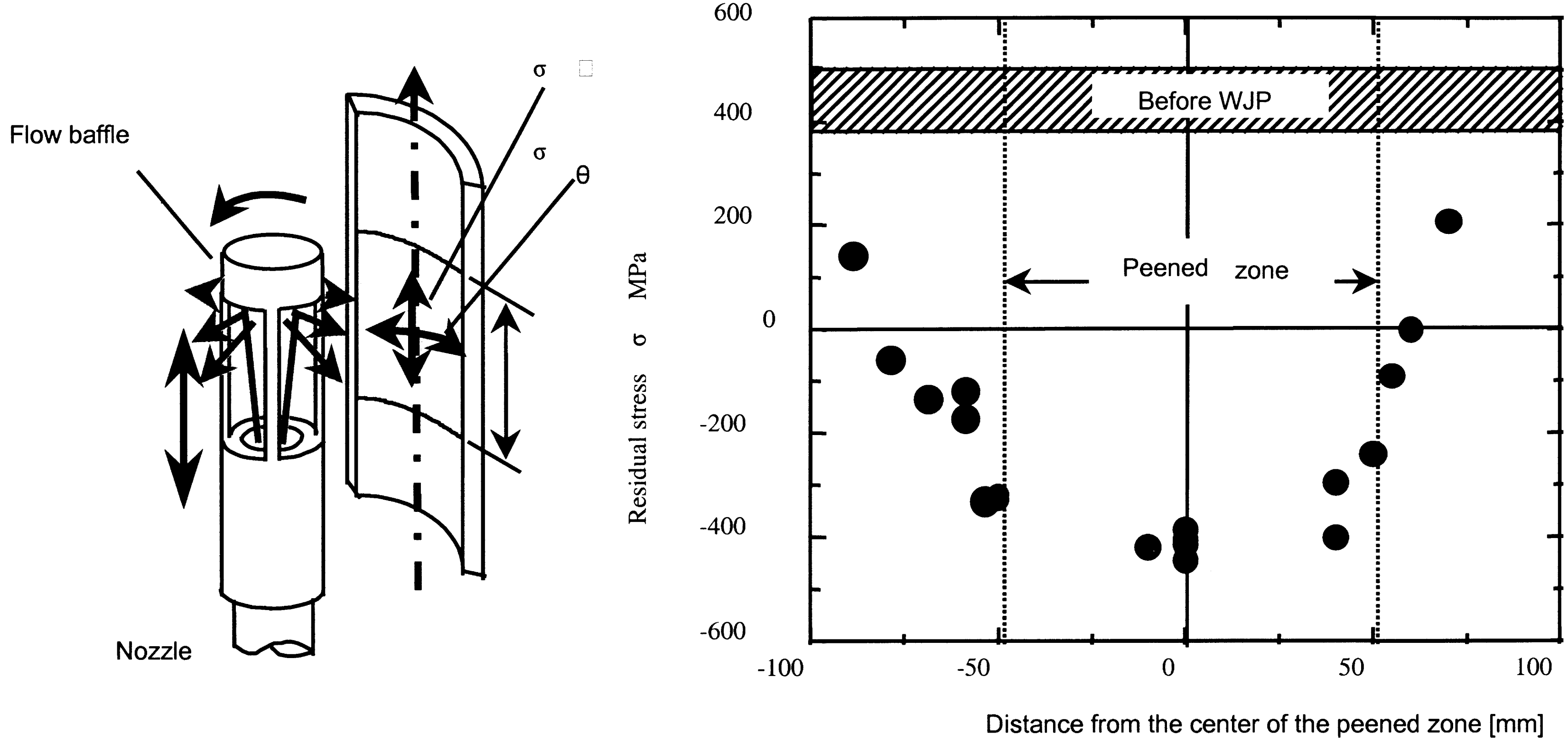

FIG. 11 Residual stress distribution of inside of small pipe after WJP using flow buffle nozzle[5].

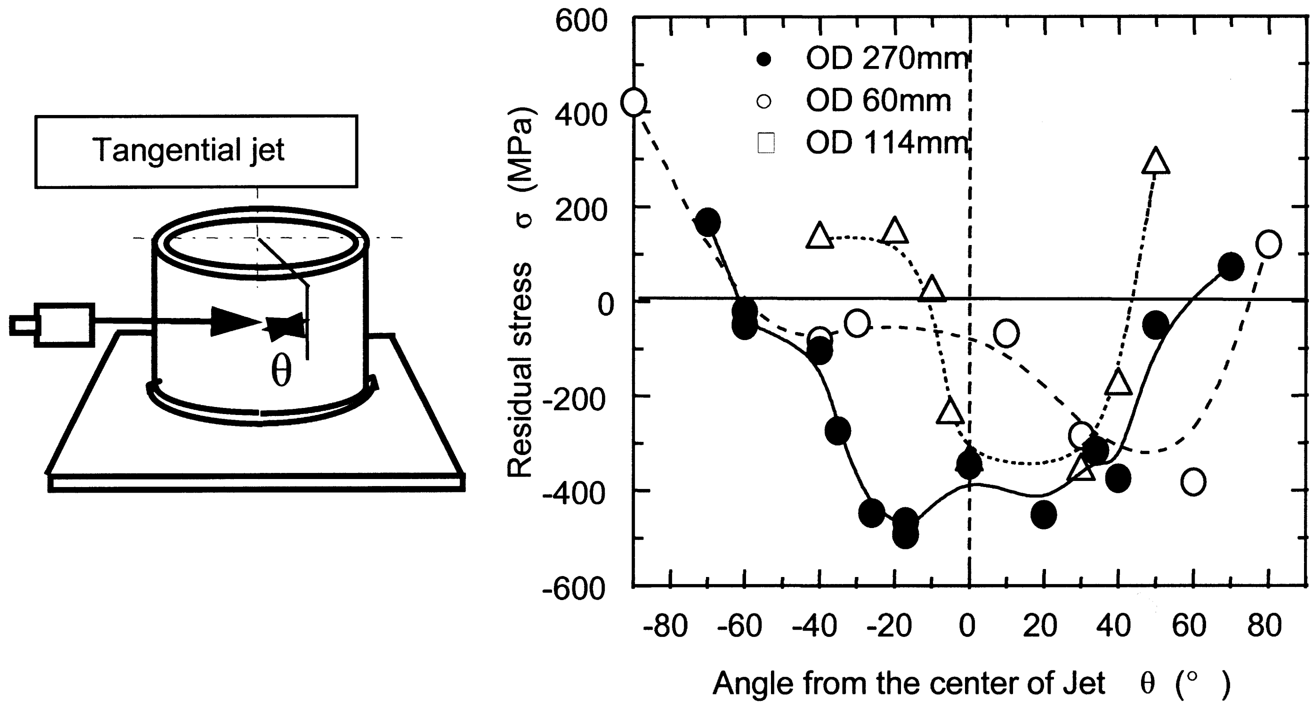

FIG. 12 Residual stress distribution of outside of large diameter pipe after WJP using tangential jet flow[6].

6. CONCLUSION

The WJP technology has several benefits for the application to water cooled reactors in service.

a) no foreign material entering into the reactor because of using only water,
b) applicability to narrow and complicated structure because it is effective in the wide range of parameters,
c) simple in the system/equipment and short period of application in actual plant.

The WJP technology was applied at Japan Atomic Power Company Tokai-2 plant in 1999 outage. Remote handling tools and devices have been developed for the application of each weld lines and the effects are verified using mock up tests in advance. The WJP treatments were performed without any trouble.

WJP treatment technology has been developed for inner and outer diameter pipe for application to the other BWR internals.

For the economical life extension of internals against SCC, the mitigation technology should be flexible and good cost performance. We believe the WJP technology is one of the leading candidates for the needs and will be widely used in the near future.

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REMOTE CONTROLLED IN-PIPE MANIPULATORS FOR DYE-PENETRANT INSPECTION AND GRINDING OF WELD ROOTS INSIDE OF PIPES

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Abstract

Technical plants which have to satisfy stringent safety criteria must be continuously kept in line with the state of art. This applies in particular to nuclear power plants.

The quality of piping in nuclear power plants has been improved quite considerably in recent years. By virtue of the very high quality requirements fulfilled in the manufacture of medium-carrying and pressure-retaining piping, one of the focal aspects of in-service inspections is the medium wetted inside of the piping.

A remote controlled pipe crawler has been developed to allow to perform dye penetrant testing of weld roots inside piping (ID $\geq$ 150 mm). The light crawler has been designed such that it can be inserted into the piping via valves (gate valves, check valves, ...) with their internals removed. Once in the piping, all crawler movements are remotely controlled (horizontal and vertical pipes incl. the elbows).

If indications are found these discontinuities are ground according to a qualified procedure using a special grinding head attached to the crawler with complete extraction of all grinding residues. The in-pipe grinding is a special qualified three (3) step performance that ensures no residual tensile stress (less than 50 N/mm^2) in the finish machined austenitic material surface.

The in-pipe inspection system, qualified according to both the specifications of the German Nuclear Safety Standards Commission (KTA) and the American Society of Mechanical Engineers (ASME), has already been used successfully in nuclear power plants on many occasions.

1. INTRODUCTION

Technical installations that are subject to special safety considerations have to be constantly checked and brought into line with the latest developments in the state of the art. This is especially important in the case of nuclear power plants. The longer they have been in service, the greater the maintenance and repair effort are required.

Piping systems form the links between the components of the reactor coolant system (RCS) and between the RCS and the auxiliary systems, and they are thus required to meet the stringent safety standards prescribed for nuclear systems. It is frequently necessary to inspect, test or perform machining operations on the inside of pipes.

A range of equipment is available for such purposes which makes it possible to perform visual inspection, ultrasonic (US) and eddy current (EC) testing and grinding, cleaning, milling and welding work inside of horizontal as well as vertical pipes.

A special remote controlled manipulator has recently been developed which enables to perform dye penetrant inspection inside of pipes in combination with local grinding. This system of manipulators is able to be inserted into the pipe via valves (gate valve, angle valve, check valve, etc.) with their internals removed. So, it is not necessary to cut the pipe to get access to the inner pipe surface.

The in-pipe manipulators cover the inner diameter from 80 mm to 1000 mm.

2. DESCRIPTION OF THE IN-PIPE MANIPULATOR TECHNOLOGY

2.1. General Requirements to be fulfilled by the manipulators

A relevant criterion for the design of an in-pipe manipulator is, apart from the task itself, the situation regarding accessibility in the section of piping on which work is to be performed. It is necessary to negotiate horizontal and vertical pipes including the elbows. Generaly it should be possible to introduce the manipulator into the piping via a valve with its internals removed.

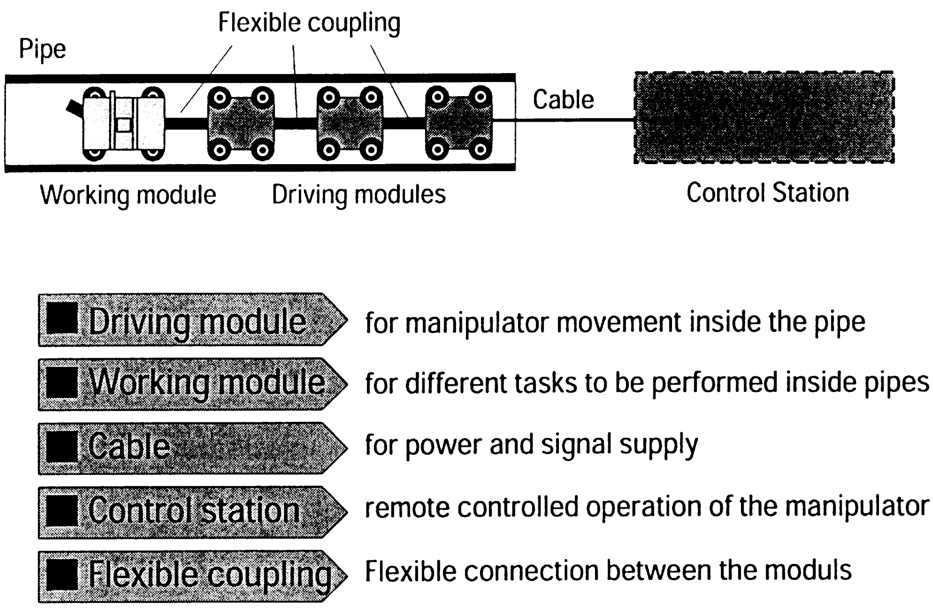
The design of the driving unit must be such that piping junctions can be negotiated. Furthermore account must be taken of the following pipe specific factors:

- Coverage of as large as possible a diameter range
- No obstruction as result of changes of inner diameter and ovalities
- It must be possible to negotiate edge misalignment and weld sag without any problem
- Adaptability of various working module to the driving module
- It must be possible to negotiate projecting parts such as sensors

It is technically difficult and not economically expedient to design the pipe manipulator for all eventualities. Before used for in-pipe operations, an examination of interferences must be performed, and if necessary, the manipulator must be adapted to specific requirements.

2.2. General Equipment Configuration

The general arrangement of the equipment is that the manipulator is inside of the pipe, connected with a cable, going to the outside located control station.



Sketch 1 shows the general arrangement of in-pipe manipulator

The manipulator consists of two (2) main parts, the driving module and the working module. These components are linked with flexible couplings that allow to pass elbows and enables the insertion of the manipulator via valves.

Responsible for movements inside of the pipe is the driving module. If a high pulling force is required the number of driving modules will be increased to ensure proper and reliable function.

There is a choice of six (6) different driving modules, the different design (A to F) has different advantages and disadvantages.

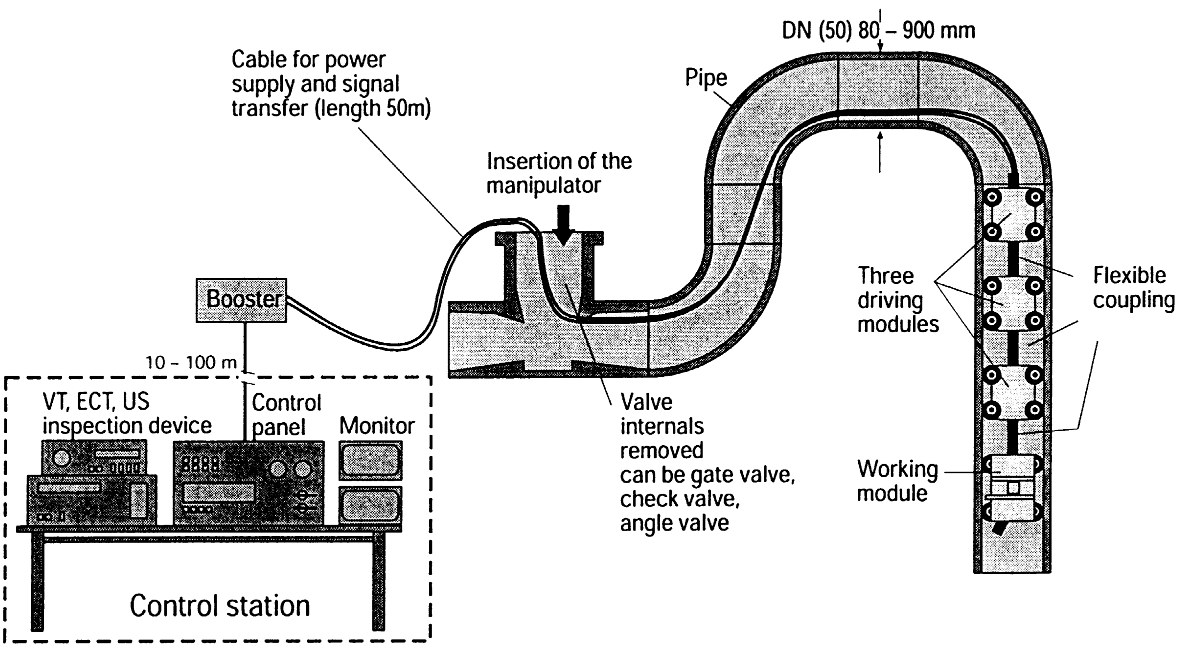
The basis for the decision what design shall be applied is always the requirement-specification for the application of the manipulator in the pipe. E.g. how to insert the manipulator into the pipe or wether there are obstacles in the pipe or how many elbows must be negotiated or what task(s) has to be performed, ...

The different driving modules are described as follows

A) module with spring loaded wheels
B) module with pneumatically loaded wheels
C) module with hydraulically loaded wheels
D) module with spider legs
E) module with stepping mechanism
F) module with guide elements and vacuum unit

To be able to perform the demanded in-pipe task a working module is necessary. The working module is attached to the driving module via a special flexible coupling and can be installed in front of the driving modules, between or in the back.

It is possible to use different working modules with only one set of driving modules together with one (1) set of control station. But it is also possible to have different tasks on only one (1) working module, e.g. milling, EC-testing and cleaning.



Sketch 2 shows the arrangement of an in-pipe manipulator during application on site

The different working modules are described as follows

G) module with high resolution camera for visual inspection
H) module for qualified grinding
I) module for milling
J) module for welding
K) module for in-pipe cleaning
L) module for eddy current testing (ECT)
M) module for ultra sonic inspection (US)
N) module for dye - penetrate inspection
O) module for delta ferrite measurement

3. EXAMPLE OF A SPECIAL DEVELOPED IN-PIPE MANIPULATOR

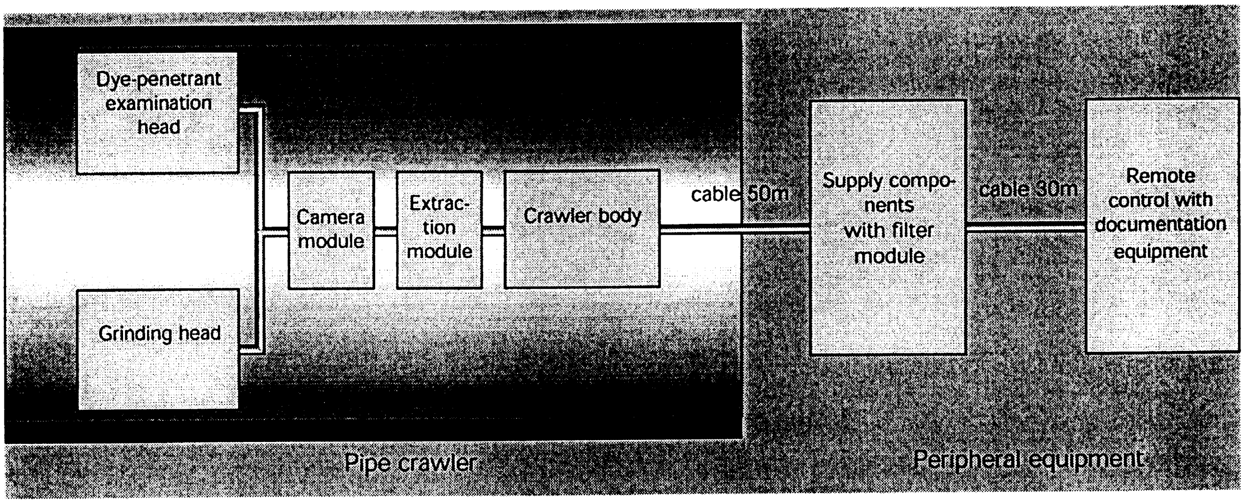
3.1. Remote controlled manipulator for surface crack-examination and local grinding at the weld roof area inside of pipes

A remote controlled in-pipe manipulator has been developed to allow surface crack (dye penetrant) examination of weld roofs inside of piping. The light weight manipulator has been designed such that it can be inserted into the piping via dismantled gate valves or other valves. Once in the piping, all manipulator movements are remote-controlled.

If indication of cracking are found these discontinuities are ground according to a qualified procedure using a special grinding head attached to the driving module. This grinding module is equipped with cleaning features with complete extraction of all grinding residues.

The in-pipe manipulator can be used in piping with an inside diameter of ≥ 150 mm.

This in-pipe surface inspection technology, qualified according to both the specifications of the German Nuclear Safety Standards Commission (KTA) and the American Society of Mechanical Engineers (ASME), has already been used successfully in nuclear power plants on many occasions.



Sketch 3 shows a diagram of the in-pipe manipulator for dye penetrate inspection and grinding

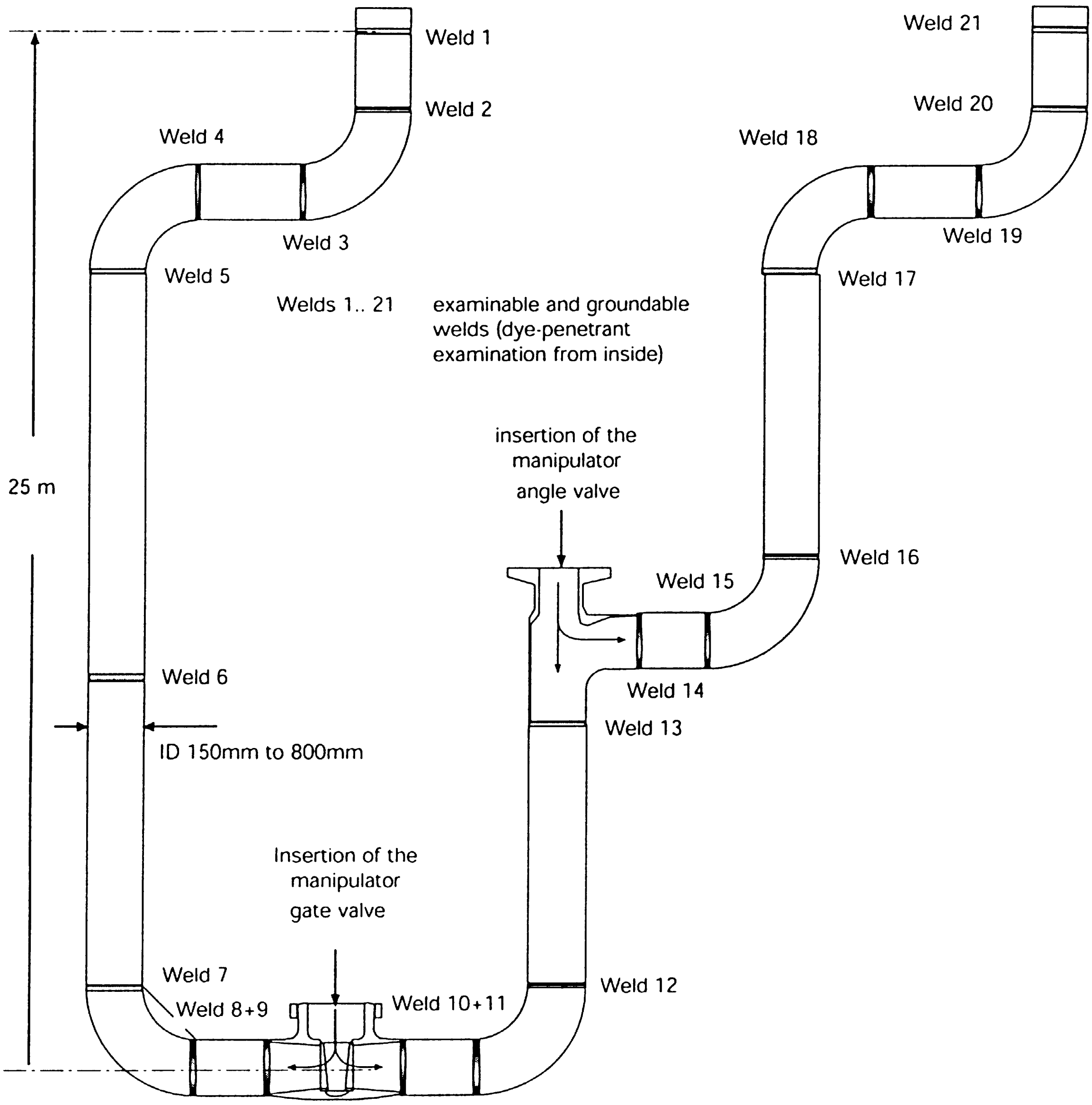
3.2 Main features of the manipulator system for dye penetrant inspection

- light weight, modular design, can easily be handled
- can be used in a large range of diameter, e.g. inside diameter of 450 mm to 650 mm or 150 mm to 250 mm
- it takes only around five (5) minutes to insert or retrieve via gate valve with trim removed (e.g. gate valve of recirculation pipe line of BWR)
- dye-penetrant examination capability in line with KTA and ASME requirements
- qualified grinding of austenitic materials in three passes up to 10 mm depth of material removal (first pass: qualified grinding with a special disc to remove material, second pass: polishing to avoid material hardening, third pass: super polishing to avoid tensile stress on the finished material surface)

features of qualified grinding

- no material hardening
- no residual tensile stress in the finish-machined material surface
- material temperature during grinding < 75 °C
- resistance to intergranular corrosion as per DIN 50914
- no cracks, no impurities (electron-microscopic examination)
- notch-free surface with surface roughness Ra < 1 µm

- grinding residues are completely extracted
- can be used in both, horizontal and vertical piping including elbows (bend radius 1,0 times pipe inner diameter)
- freely adjustable speed of travel from 0 to 5 m per minute
- can travel up to 80 m into piping
- visual inspection capabilities with high resolution zoom-camera



Sketch 4, example: pipe line showing examinable/grindable welds (the manipulator is inserted via the gate valve or via the angle valve).

4. CONCLUSION

Siemens has been active in the field of in-pipe manipulator technology for over 20 years and can boast numerous successful applications in nuclear power plants, so that Siemens pipe manipulator technology is at an extremely advanced stage of development.

Our continually expanding knowledge in this field, along with the expertise gained during project planning, equipment design and qualification, as well as during actual field applications in nuclear power plants, forms the basis of our scope of supplies and services.

EQUIPMENT AND TECHNIQUES FOR COMPONENT INSPECTION, MAINTENANCE, REPAIR AND REPLACEMENT

(Session II-b)

Chairperson

D. SHAH
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SIZING AND RECOGNITION OF CRACKS AND POROSITY IN WELD METALS USING ACOUSTICAL HOLOGRAPHIC INSPECTIONS

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Abstract

Non-destructive inspection of weld metals in carbon steel and stainless steel pipes become more important if high reliability of power plants are to be maintained. Flaws in the weld metals, i.e. porosity or cracks, may be produced by weld condition changes and then grow due to up by thermal stress repetitions. If non-destructive inspections of the flaw sizing and recognition in pipe welds are possible, reliability evaluations and residual life estimations of each pipe can be improved. Based on this background, the authors applied an acoustical holographic method to weld metal inspections. In the present work, flaw sizing and recognition capabilities of the acoustical holographic inspections were compared with those of conventional radiographic testing and the ultrasonic tip-echo method. Ultrasonic inspections with normal, 45 and 70 degree angled beams were made for flaws in weld metals of SGV410 steels and SUS316 stainless steel. The capabilities of the acoustical holographic method as determined by the above inspections are as follows. For porosity in 28 mm thick weld metals, the holographic method with a normal beam can detect and represent the flaws as spherical images, although sizing is overestimated, making it inferior to radiographic testing. For cracks in the weld metals, the sizing errors in the holographic method with 45 and 70 degree angled beams are confirmed to be superior to those of the tip-echo method.

1. INTRODUCTION

Non-destructive sizing and recognition of flaws in weld metals of pipes which have been put into service was very important for power plant safety considerations. Acoustical holography has been developed to improve lateral and range resolutions for flaw sizing and flaw recognition from ultrasonic images.

In the present work, flaw sizing and recognition capabilities of acoustical holographic inspections were compared with those of conventional radiographic testing and the ultrasonic tip-echo method.

2. THEORY OF ACOUSTICAL HOLOGRAPHY

2.1. Construction and reconstruction of a hologram

Figure 1 shows the concept of hologram construction though ultrasonic inspection and flaw image reconstruction from the hologram [1]. The acoustical hologram was constructed by spatial interference fringes between reference waves and object waves reflected by the flaw. The reconstruction waves pass through the pattern on the hologram and are diffracted to construct the flaw image. The position where the image is focused on is the same as the flaw position.

In the present work, the hologram construction procedure was changed to use digital signal processing which makes hologram fringes through a time coincidence measurement between the object reflected pulse and high frequency clock pulses instead of reference waves [2]. The new method has two advantages. First, it is to be able to use wide band transmission pulses which remarkably reduce the range resolution in comparison with that of narrow band transmission pulses. Secondly, it is able to make a minute hologram by making the frequency of the clock pulses higher than that of the ultrasonic waves. This can improve lateral resolution in the reconstructed images. The reconstruction flaw images can be obtained by computer reconstruction [3] with numerical calculation of wave interference.

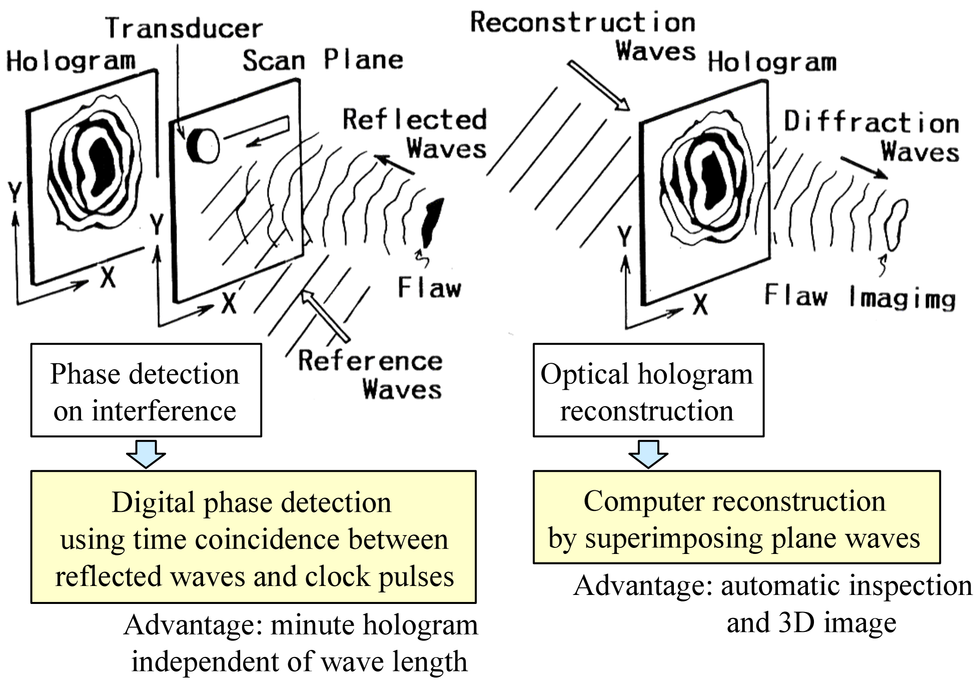

FIG. 1. Concept of acoustic Holography.

2.2. Configuration of the inspection system

The acoustical holographic inspection system consists of an automatic scanner, holographic signal processor and a mini-computer imaging unit, as shown in Figure 2. The signal processor can process signals of flaw echoes at a maximum repetition transmission cycle of 500 Hz. It records the inspection data at each scan position, separated by 0.125 mm, onto floppy disks. Inspection data corresponding to a holographic signal consist of the scanning positions, coincidence signal (plus or minus), echo heights and propagation time that are measured by 32 MHz clock pulses. A mini-computer imaging unit is used for automatic data processing including computer image reconstruction. The operator can obtain three-dimensional flaw images on the display of the unit at any convenient time. The scanner drives the ultrasonic probe with a rectangular scan path on the plane surface. Its speed is changeable from 15mm/s to 60mm/s. The probe holds several kinds of point focus type transducers which can be inclined for normal and angled beam testing. The resonant frequencies of the transducers are 5 MHz for inspections of SGV410 steel and 2.25 MHz for SUS316 stainless steel.

3. ULTRASONIC IMAGING EXPERIMENT

3.1. Test pieces and flaws

Carbon steel SGV410 and stainless steel SUS316 were chosen as the test piece materials because they are the main metals for piping components which are effected by thermal stress cycles, resulting in possible formation of fatigue cracks. The test pieces were flat plate shapes and have a weld region at the center, as shown in Figure 3. All test pieces were 28 mm thick.

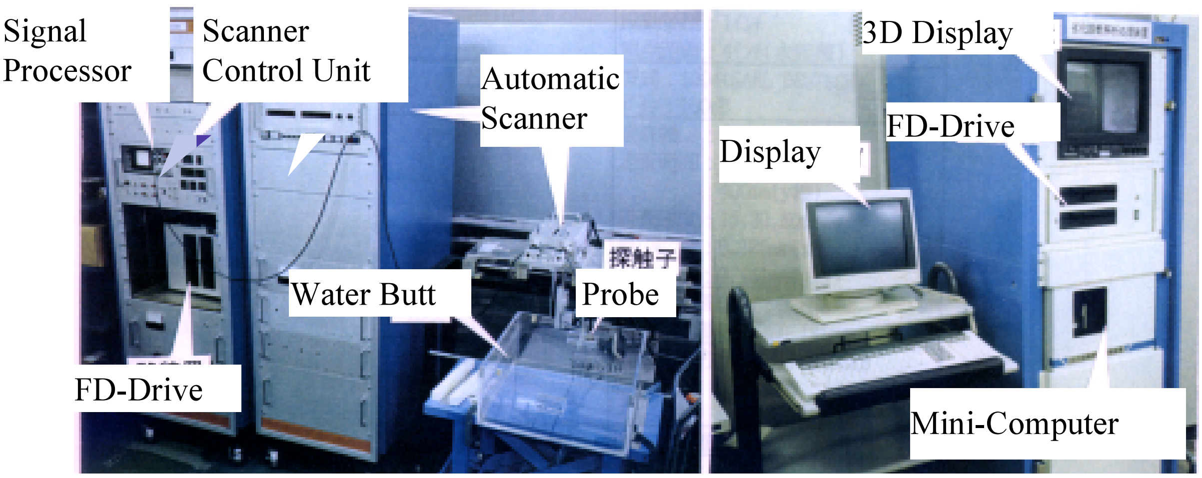

FIG. 2. System configuration for acoustical Holographic inspection.

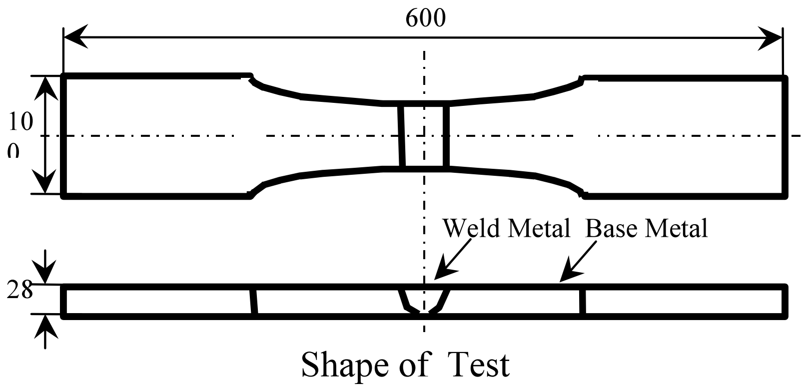

FIG. 3. Geometry of test piece (unit: mm).

3.2. Flaw imaging

Porosity in the weld metals was imaged by holographic inspection with a normal focused beam and an immersion method. It may note that the holographic inspection method can realize crack images like the actual shape and which are correctly positioned.

3.3. Flaw sizing

Sizes of porosity were measured by holographic inspections with a normal focused beam and conventional radiographic tests. The maximum length of the image was measured as the porosity diameter in both methods.

Sizing results of porosity obtained by acoustical holography indicate that porosity sizes obtained by holographic inspections are overestimated against the real sizes, thus tendency in the sizing in SUS316 stainless steel is greater than that in SGV410 steel. The tendency will be caused by increment of amplifier gain to receive the reflection waves which are strongly attenuated in the stainless steel welds.

Sizes of cracks were measured by each non-destructive method, i.e. acoustical holography with angled beams, ultrasonic tip-echo method and conventional radiography.

4. DISCUSSIONS

4.1. Comparison with results by radiographic testing

It is clear that the deviations from the real sizes in the measurements by radiography are smaller than by the holographic method, and the deviations for radiography do not change independently of the weld materials.

4.2. Comparison with results by tip-echo sizing method

The sizing error in SUS316 stainless steel is much greater than that in the SGV410 steel. The reason may be the same as for holographic sizing of porosities. Holographic sizing capability for the fatigue cracks in weld metals is proved superior to that of the tip-echo method.

5. CONCLUSION

Flaw sizing and recognition capabilities of acoustical holographic inspections were compared with those of conventional radiographic testing and the ultrasonic tip-echo method. The nondestructive inspection methods were done to detect and image flaws in weld metals of SGV410 steel and SUS316 stainless steel. The capabilities of the acoustical holographic method, as obtained from the experimental results, were as follows.

For porosity in 28mm thick weld metals of SGV410 steel and SUS316 stainless steel, the holographic method with an normal beam could detect and represent the flaw as a spherical image, although its sizing was inferior to that of radiographic testing.

For the cracks in the weld metals, the sizing capability in the holographic method with 45 and 70 degree angled beams was confirmed to be superior to that of the tip-echo method.

The research was carried out by the Japan Power Engineering and Inspection Corporation (JAPEIC), which was entrusted by the Ministry of International Trade and Industry of Japan (MITI).

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REMOTE CONTROLLED IN-PIPE MANIPULATORS FOR MILLING, WELDING AND EC-TESTING, FOR APPLICATION IN BWRS

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Abstract

Many pipes in power plants and industrial facilities have piping sections, which are not accessible from the outside or which are difficult to access. Accordingly, remote controlled pipe machining manipulators have been built which enable in-pipe inspection and repair.

Since the 1980s, defects have been found at the Inconel welds of the RPV nozzles of boiling water reactors throughout the world. These defects comprise cracks caused by stress corrosion cracking in areas of manual welds made using the weld filler metal Inconel 182. The cracks were found in Inconel-182 buttering at the ferritic nozzles as well as in the welded joints connecting to the fully-austenitic safe ends (Inconel 600 and stainless steel). These welds are not accessible from outside.

The ferritic nozzle is cladded with austenitic material on the inside. The adjacent buttering was applied manually using the weld filler metal Inconel 182. The safe end made of Inconel 600 was welded to the nozzle also using Inconel 182 as the filler metal.

The repair problems for inside were solved with remote-controlled in-pipe manipulators which enable in-pipe inspection and repair.

A complete systems of manipulators has been developed and qualified for application in nuclear power plants. The tasks that must be performed with this set of in-pipe manipulator are as follows

1st step — Insertion of the milling/ET manipulator into piping to the work location

2nd step — Detection of the transition line with the ferritic measurement probe

3rd step — Performance of a surface crack examination by eddy current (ET) method.

4th step — Milling of the groove and preparation for weld backlay and, in case of ET indications, elimination of such flaws also by milling.

5th step — Welding of backlay and/or repair weld using the GTA pulsed arc technique.

6th step — After welding it is necessary to prepare the surface for eddy current testing. A final milling inside the pipe is done with the milling manipulator to adjust the diameter with a smooth surface according to the design.

7th step — Final acceptance test by NDE, using ET method from inside and UT method from outside.

The manipulators are planned to be applied in different nuclear power plants in spring 2000.

1. INTRODUCTION

Additional measures must be taken to ensure operational reliability on piping which is subject to special safety criteria.

Owing to the very high manufacturing quality of piping which conveys a medium or retains pressure, particular attention is paid during in service inspections to the inside surface of the piping which is in contact with the medium. On the one hand, there are extensive requirements for verification of the integrity of safety-related power plant components, on the other there are requirements for regular inspection of piping from the inside. These inspections must be able to provide reproducible results.

Damage caused by wear, erosion, corrosion and crack formation may occur in the course of the piping's service file. Furthermore, undesired incrustations and deposits may form.

A visual inspection provides information on the optical condition of the inside surface of the piping or plant items. To detect material defects which cannot be located visually, suitable in-pipe inspection systems are necessary. If damage is then detected and is not accessible in any other way, it must be repaired from the inside. This repair work requires remote-control machines which can be operated with a high degree of precision and reliability. Pipe manipulators are required which can perform remote-controlled milling, welding and cleaning work in addition to US/EC-inspection and testing operations.

2. TECHNICAL STATUS OF IN-PIPE MANIPULATOR TECHNOLOGY

The power plant industry has been building and developing remote-controlled in-pipe manipulators which have been used for inspection, examination and repair of piping for over 20 years. These piping vehicles can move independently in the piping by means of special driving modules. The basic design of the machines comprises the two modules, the working module and driving module, which are connected by a flexible coupling. A cable bundle between in-pipe manipulator and control station is responsible for power and signal transfer.

2.1. Pipe manipulators for visual in-pipe inspection

Visual in-pipe inspections using a pipe manipulator give a general idea of the condition of the inside pipe surface or internals. This pipe inspection work concentrates on the detection of surface damage, which can be caused by wear, erosion, corrosion or crack formation. This damage occurs primarily in the area of weld roots, in pipe elbows and branch connections and on highly stressed items. In addition to this damage, deposits and incrustations can form at certain points, the extent of which can be established by means of visual in-pipe inspection.

Pipe inspection manipulators have been developed with which piping can be visually inspected by remote control. These pipe vehicles can be used for piping with an inside diameter ≥ 80 mm and can be operated at a speed of between 0 an 5 m/min. The standard length of the cable package necessary for power supply and for control and signal transfer is 70 m; longer cable bundles can be used without any problem.

After performance of pipe repair work it is often established that foreign matter has accumulated in the pipe. This foreign matter is located using the pipe inspection manipulators and recovered by remote control with special gripper or suction equipment. This recovery equipment is adapted to the manipulator head so that recovery manipulations can be performed within camera range. Different gripper sizes are available for recovery of different sizes of foreign matter.

Today it is standard practice-after repair work has been completed on safety-related piping for a final inspection to be performed-to ensure that all foreign matter has been removed and to document the cleanliness of the piping.

In-pipe inspections are nowadays performed using high resolution cameras with zoom. The camera system is fastened to the manipulator head and is fully mobile in all directions thanks to the pan-tilt head. An infinitely variable lighting source, is attached concentrically to the camera.

Piping inspections are always recorded on video together with the positioning data; the depth of penetration of the pipe manipulator and the angle position of the camera is displayed in the inspection recording (positioning data).

The video recording is therefore a complete document of the piping inspection and provides valuable information for subsequent inservice inspections. Furthermore, video images displayed on the monitor can be printed to provide comprehensive report documentation.

2.2. Manipulators for in-pipe machining with subsequent materials examination

Many pipes in power plants and industrial facilities have piping sections, which are not accessible from the outside or which are difficult to access. Accordingly, remote-controlled pipe machining manipulators have been built which enable in-pipe inspection and repair.

The task of the machining module is to machine the inside of piping (weld root) to permit non-destructive examination. After subsequent performance of an eddy current examination, any cracks detected are milled out. When the milling work and eddy current examinations in the pipe have been completed, the volume of material machined out must be replaced so that the original wall thickness is achieved. After welding work on the pipe interior has been performed, a combined ultrasonic and eddy current examination must be performed by remote control.
Prior to this, it must be confirmed using the TV camera that the welded surface requires no additional machining before examination.

The remote-controlled in-pipe machining modules can move independently within the piping by means of a driving module. The basic structure of the machines consists of two modules, the working module and driving module, which are connected by a flexible coupling. A control

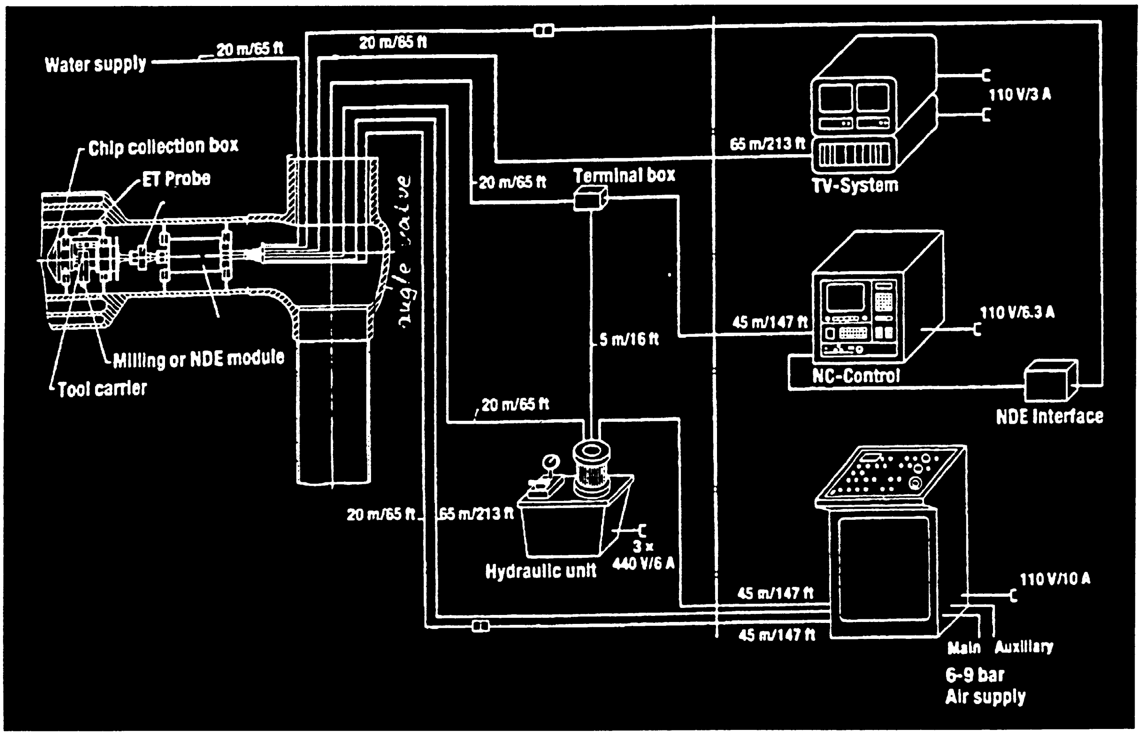


Figure 1
Equipment arrangement for in-pipe machining

system ensures reliable movement of the manipulator in the piping. Pipe elbows with a radius of curvature greater than or equal to 1.5 x D and gradients up to the vertical can be negotiated by the machining modules without any problems.

The entire milling consists of: tool carrier with examination and milling unit with suction features for chip removal, hydraulic equipment unit, camera unit, numerical control system and operator and control console for the stepping mechanism control system.

The powerful suction extractor is designed so that extraction can be performed with the milling machine at any location. An eddy current probe is integrated into the milling module which inspects the inside surface of pipes for surface cracks. This combination has the advantage that surface examination and machining can be performed with one inside manipulator. The frequency of introducing and moving the machine into the piping system is therefore reduced to a minimum.

To be able to perform the non-destructive examination of the repair weld after mechanized welding, the milling module is quickly replaced by the examination module. The driving module can therefore also be used for the ultrasonic and eddy current examinations.

With a defined interface, the position data of the examination system are available to the data acquisition and evaluation unit.

2.3. Manipulators for performance of in-pipe repair welding

The pulsed tungsten arc orbital welding method permits performance of all out-of-position welds.

The basic design of the in-pipe machine consists of two modules, the tool carrier and the driving module, which are connected by a flexible coupling. The in-pipe welding machines also satisfy the same criteria as the milling and examination machines; they can negotiate 1.5 x D pipe bends and gradients up to the vertical and perform welding in any location of the pipe. Movement and bracing of the machine in the pipe is performed in a similar manner to that used for the milling and examination machines.

The tool carrier is equipped with welding equipment for in-pipe welding tasks (repair welds and weld overlays). This consists of a GTAW torch, two welding cameras and an integrated welding wire feed.

The welding equipment consists of the following components: welding machine, welding power supply, TV unit and driving unit, control station, whereby cooling of the power source and the torch are performed by an additional equipment unit (cooling unit) in closed cooling loops.

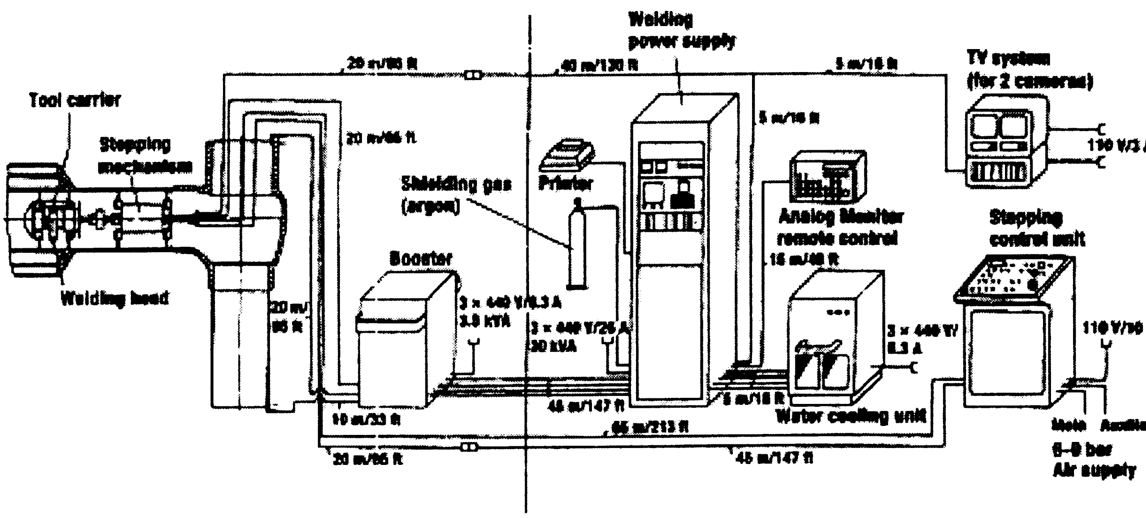


Figure 2: Equipment arrangement for in-pipe welding (sketch)

To achieve 100 % ignition reliability in the pipe, a pilot arc torch is used. This activates a small arc in the torch head by remote control which preionizes the shielding gas for the main arc. The welding head and the two cameras are water-cooled to protect against elevated temperature.

Guidance of the welding head parallel to the weld joint is ensured using a teach-in control system. Height adjustment of the welding torch in a radial direction is performed by arc control (AVC).

The operator puts in the welding parameters into the program unit of the power supply. The integral recorder monitors the important welding parameters. After centering and bracing the welding machine in the pipe, the milled out weld must be positioned with the aid of the TV positioning camera.

The inspection to determine whether the welding machine is placed parallel to the weld is performed at the weld edge and at the electrode tip above it.

The weld wire coil is installed on the front of the tool carrier. Filters which supply an adequate image both with an arc and without an arc have been installed. The welder monitors that the welding wire is being fed correctly into the molten weld pool via one of the two TV cameras attached to the welding head. With the second camera he monitors the finished weld.

Return of the welding head, lateral shift (overlapping) and positioning can be performed manually or automatically, as can restarting of the welding process. Each time the welding machine is removed the welding head must be repositioned to the weld or the bead last welded. Visual monitoring of the welding process is supplemented by position indicators and recording of current, voltage, welding speed and wire speed.

3. EXAMPLE OF A SPECIALY DEVELOPED IN-PIPE MANIPULATOR

3.1. Remote controlled in-pipe manipulators for milling, welding and EC-testing, for application in BWR's

Since the 1980s, defects have been found at the Inconel welds of the RPV nozzles of boiling water reactors throughout the world. These defects comprise cracks caused by stress corrosion cracking in areas of manual welds made using the weld filler metal Inconel 182. The cracks were found in Inconel-182 buttering at the ferritic nozzles as well as in the welded joints connecting to the fully-austenitic safe ends (Inconel 600 and stainless steel). These welds are not accessible from outside.

A remote controlled manipulator system has recently been developed which enables to repair the Inconel 182 welds of the RPV nozzles of boiling water reactors (BWR's). The detail design of the nozzle weld is shown in Fig. 1.

The system of manipulators consists of two working modules

- one working module for milling cleaning, eddy current testing and ferritic inspection
- one working module for welding

In this case, the insertion of the manipulator into the pipe is via an open pipe end, but it could also be via valves with there internals removed. Inside the pipes the manipulator can travel in horizontal as well as vertical pipes incl. the elbows.

Working module for milling

This module is equipped with the milling tool with suction features, a ferritic measurement probe and an eddy current probe .

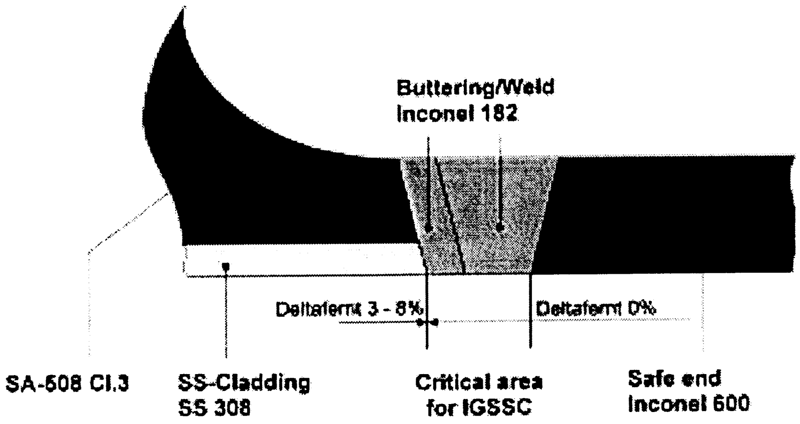


Fig. 1: Detail design of the Nozzle

The milling tool is located on a axial-, rotation- and radial slide system. With this slide system the milling tool is fully mobile in all directions, the tool is always observed via two (2) cameras. Next to the tool there is a very specialy designed suction nozzle for removal of the milling chips; this can be performed during or after the milling process.

The tool is driven by a powerful hydraulically turbine with adjustable speed.

To determine the transition line of ferritic and austenitic material the ferritic measurement examination with the eddy current probe is performed before and after repair.

To observe all the a.m. processes (milling, cleaning, ECT, ferritic measurement) two (2) TV cameras are installed at the head of the module.

The working parameters are inserted via the control station. Tool as well as probe movements can be performed with the aid of the program unit or manually.

Working module for welding

The welding module consists of the clamping unit and the welding head. The wire coil is mounted back to the tool carrier. All functional movements of the welding head are performed by electric DC motors and controlled by the power supply unit. The electrode is – similar to the milling tool – also located on an axial, rotation, and radial slide that ensures full mobility in all directions.

The power source permits precise reproduction of welding parameters. The welding parameters are stored in the control unit and the actual parameters are shown in the display.

The module for welding is clamped pneumatically inside the pipe. In the case of the feedwater nozzle the positioning of the manipulator take place behind the flow limiter (ID flow limiter 150 mm, ID pipe 300 mm).

Positioning of the electrode is accomplished by using the program unit at the control station.

The welding process is started according to the welding program.

The monitoring of the welding process via camera will be observed on the screen. One camera is for monitoring the bead and evaluating each bead right behind the puddle. The other camera is for monitoring the weld puddle and positioning the welding torch to the starting.

Task to be performed on site:

1st step Insertion of the milling/ET manipulator into piping to the work location.

2nd step Detection of the transition line with the ferritic measurement probe and either visually by the TV-system.

3rd step Performance of a surface crack examination by eddy current (ET) method.

4th step Milling of the groove and preparation for weld backlay and, in case of ET indications, elimination of such flaws also by milling.

5th step Welding of backlay and/or repair weld using the GTA pulsed arc technique.

6th step After welding it is necessary to prepare the surface for eddy current testing. A final milling inside the pipe is done with the milling manipulator to adjust the diameter with a smooth surface according to the design.

7th step Final acceptance test by NDE, using ET method from inside and UT method from outside.

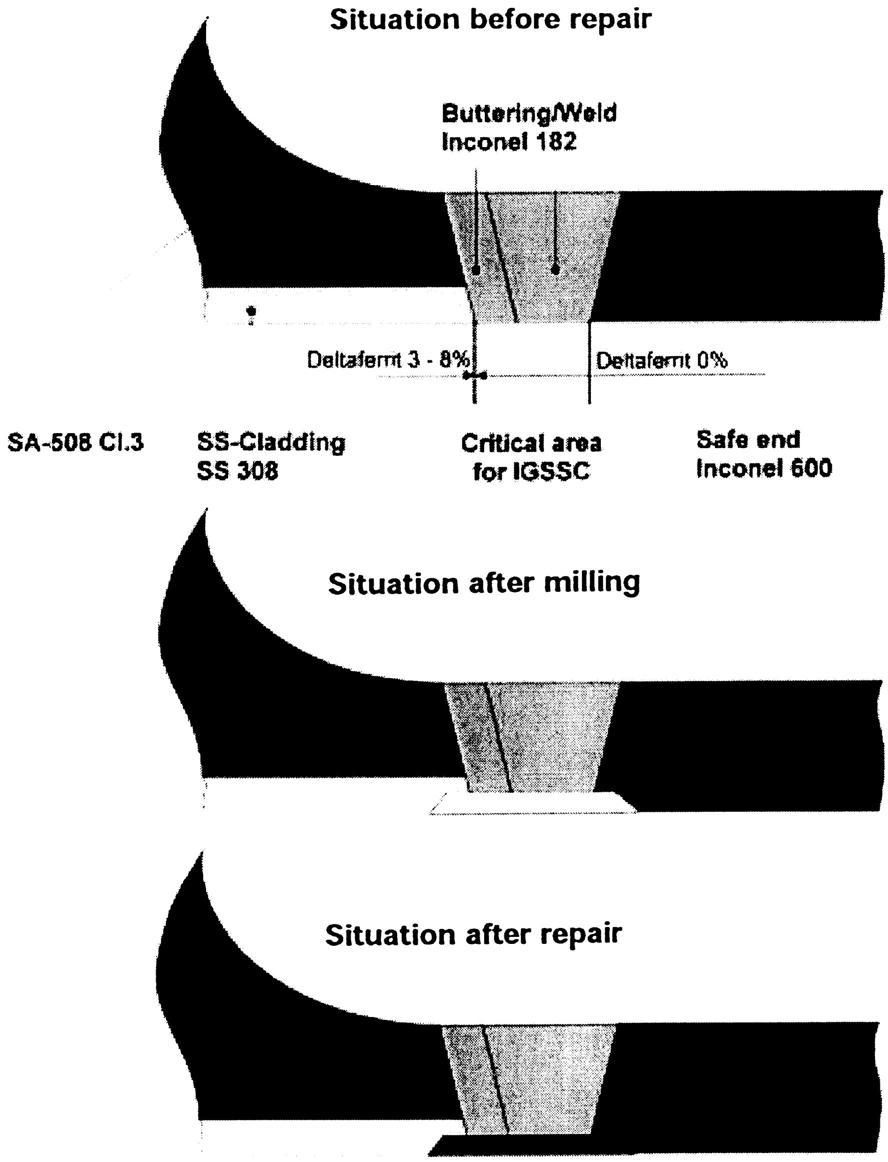
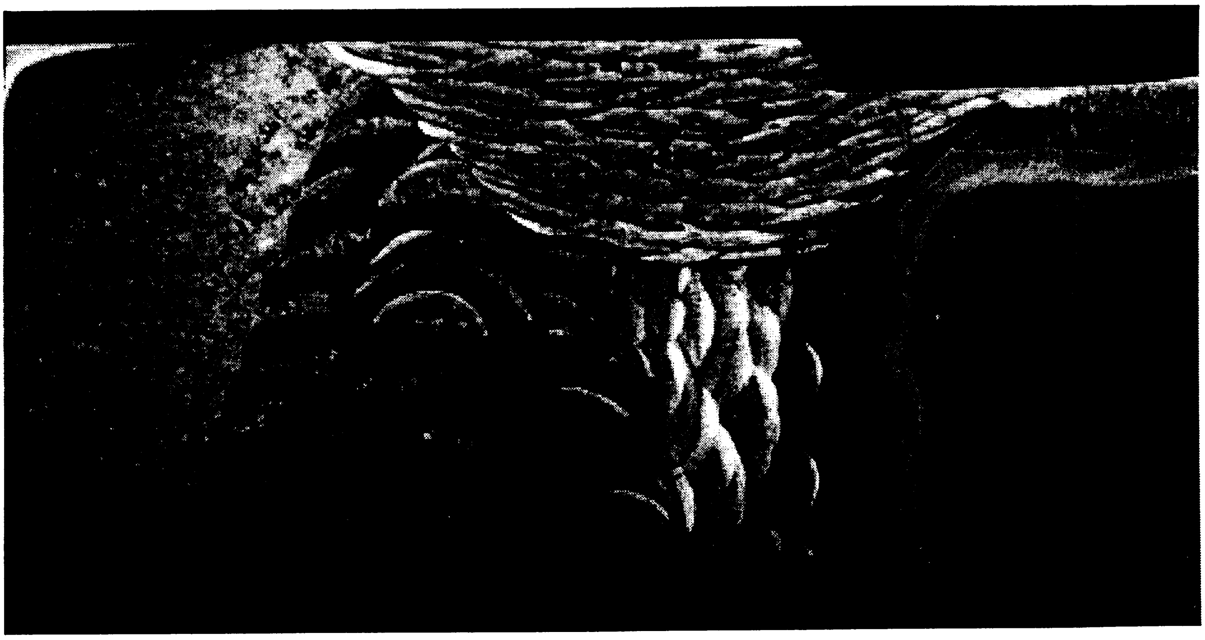


Fig. 4: Nozzle repair from inside

The following two pictures are showing the repaired nozzle weld. In the critical area for IGSSC all of the crack sensitive material of Inconel 182 is removed and replaced by Inconel 82.

picture 1: Standard repair of the nozzle weld, the Inconel 182 has been machined out up to a depth of 3 mm.



picture 2: Local repair of the nozzle weld, the depth can be up to 20 mm.

Conclusion

The knowledge acquired through on-going development work, our experience accumulated in engi-neering, equipment design and testing, and field assignments in nuclear power plants form the basis for the services and products we offer. Our manipulators are at the leading edge of development

RELIABILITY TESTS FOR REACTOR INTERNALS REPLACEMENT TECHNOLOGY

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Abstract

Structural damage due to aging degradation of LWR reactor internals has been reported in several nuclear plants. NUPEC has started a project to test the reliability of the technology for replacing reactor internals, which was directed at preventive maintenance before damage and repair after damage for the aging degradation. The project has been funded by the Ministry of International Trade and Industry (MITI) of Japan since 1995, and it follows the policy of a report that the MITI has formally issued in April 1996 summarizing the countermeasures to be considered for aging nuclear plants and equipment. This paper gives an outline of the whole test plans and the test results for the BWR reactor internals replacement methods; core shroud, ICM housing, and CRD Housing & stub tube. The test results have shown that the methods were reliable and the structural integrity was appropriate based on the evaluation.

1. INTRODUCTION

LWRs will provide a major part of the nuclear power generation in future in Japan. It is important to have measures against aging degradation in order to improve the reliability and the safety for LWRs, since structural damage of reactor internals results in a plant shutdown and it takes a long time to resolve the problem. The MITI report states that countermeasures should be considered because it can be anticipated that components similar to those that have been damaged in aging nuclear plants in other countries may become damaged in Japanese nuclear plants. Repair of reactor internals is complicated due to difficult access and working in high radiation areas. Therefore a demonstration test of the replacement method, which depends on very precise remote control techniques, is required using full-scale mockup. This must precede the application to actual plants. To conduct the demonstration test at the government level will go a long way towards getting public acceptance in the actual plant work. The other advantage of a test is to obtain much data that can be referred during the examination by the MITI for a construction permit under the regulations. The executive committee of the NUPEC project includes some members of the technical sub-committee in MITI concerned with construction permits. The examination by the executive committee will be a means of prior consultation for judging the reliability and structural integrity of the plant after the replacement has been done.

NUPEC has planned this project for technologies for rejuvenating reactor internals for the eight year period from 1995 to 2002 based on the background described above and as scheduled in Table I. The project consists of six tests for the methods to replace an in-core monitor (ICM) housing, a core shroud, a control rod drive (CRD) housing & stub tube, a jet pump riser brace (for BWR), a core barrel, and a bottom mounted instrumentation (BMI) adapter (for PWR), as illustrated in Figure 1. Each reliability test includes full scale mockup test for the replacement method. Replacement procedure reliability and structural integrity of new replaced structure are planned to be evaluated for each replacement method.

TABLE I. SCHEDULE FOR RELIABILITY TESTS OF REACTOR INTERNALS REPLACEMENT TECHNOLOGY

JFY	1995	1996	1997	1998	1999	2000	2001	2002
Plan and Design								
Manufacturing and Installation								
Tests								
ICM housing								
Core Shroud								
CRD Hsg. /Stub								
J/P riser brace								
Core Barrel								
BMI Adaptor								
Evaluation								

Following paragraphs summarize the test results obtained from three BWR reactor internals replacement methods (Core shroud replacement method, In-core monitor housing replacement method, and CRD housing & stub tube replacement method), that have already finished the mockup test and evaluation.

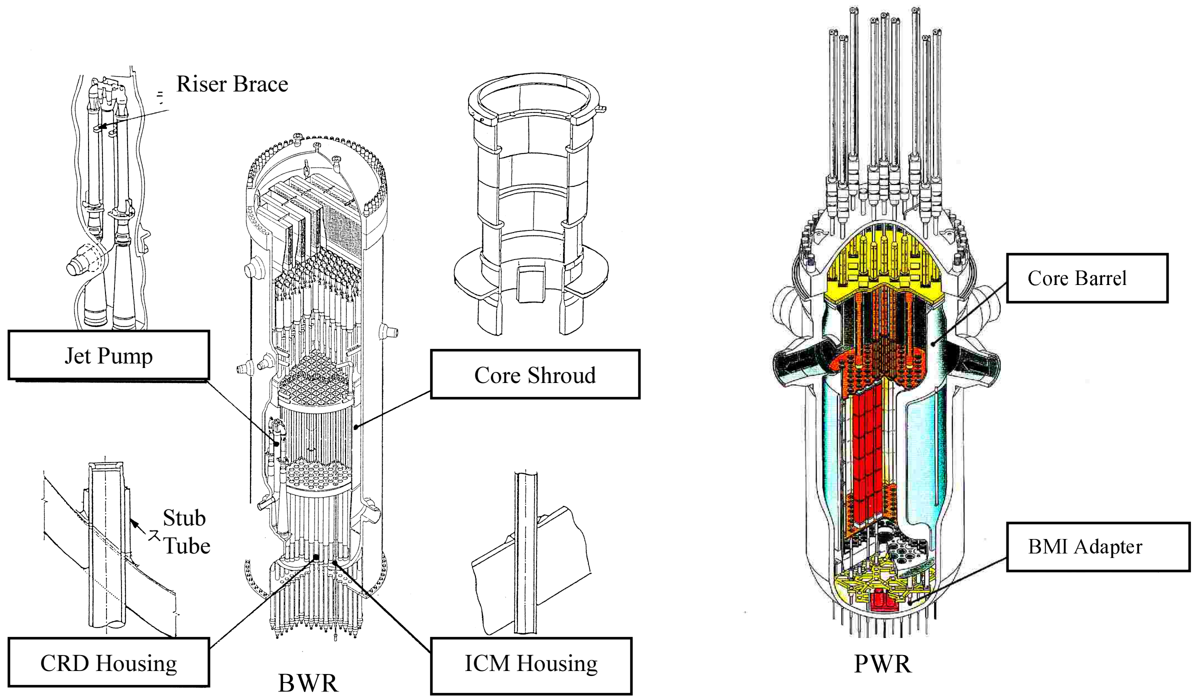

FIG 1. Replacement of reactor internals.

2. BWR CORE SHROUD REPLACEMENT METHOD

2.1. Test plan

Fundamental techniques of the shroud replacement method have been developed in a joint study program between utilities and manufacturers. The main objective of the test was to demonstrate and verify a series of steps of fitting up and welding a new core shroud using a full-scale mockup before actual application in a plant. The test consisted of the following steps:

(1) Shroud fitting-up
(2) Shroud welding
(3) Core plate installation
(4) Top guide installation.

The replacement method aims to avoid the potential stress corrosion cracking (SCC) damage of a core shroud made of SUS304 by replacing it with the new shroud made of SCC resistant materials. Figure 2 shows the shroud structures before and after replacement. Figure 3 shows the sequence of the replacement method. The test results were evaluated by considering the items of (1) alignment of the shroud and (2) structural integrity, and the reliability of the method was examined accordingly.

2.2. Test results

2.2.1. Reliability of the replacement procedure

It was confirmed that all operations were carried out successfully and the requirements were satisfied at each step.

2.2.2. Shroud fitting-up

The full-scale model of the core shroud with a height of 7 meters and a diameter of 4.5 meters was installed on the shroud support. In the test, the maximum root gap between them provided 0.15 mm which was sufficiently smaller than the specified value of 1.1 mm. The welding conditions for the test also allowed for the case in which the gap exceed 1.1 mm.

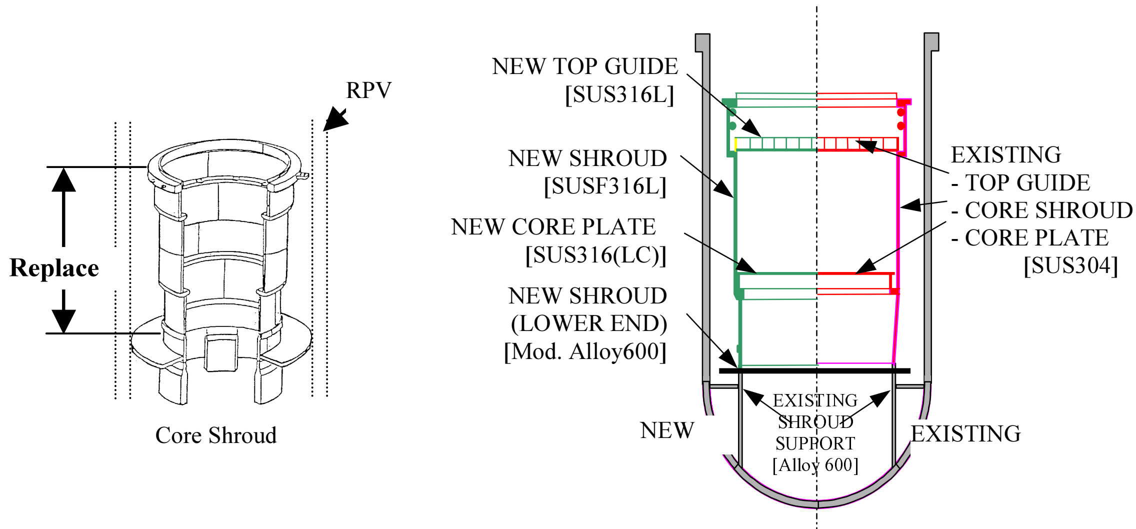

FIG. 2. New core shroud structure.

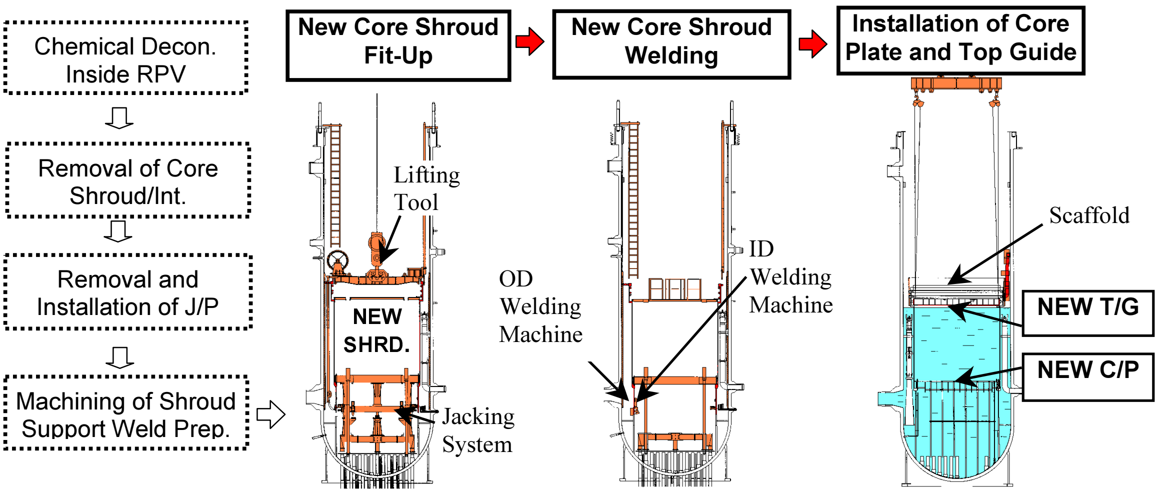

FIG 3. Core shroud replacement sequence.

2.2.3. Shroud welding

Welding between the shroud and the shroud support was conducted by automatic TIG welding technique with narrow gap weld preparation. Full automatic outside welding machine was used to make two passes on the outside, and semi automatic inside welding machine made 26 passes of narrow gap welding as shown in Figure 4. Visual tests (VT) after welding (by a video camera on the outside and direct observation on the inside) found no defects.

2.2.4. Alignment of the shroud

Deviation of the concentricity between the shroud and vessel was measured with laser alignment tools on three occasions; before, during and after the welding. The measured results met the criterion of equal to or less than 6.0 mm. The verticality of the shroud was also measured with similar laser alignment tools. The result was about 0.1mm over the 4 meter distance between the upper and lower templates installed on top guide and core plate location.

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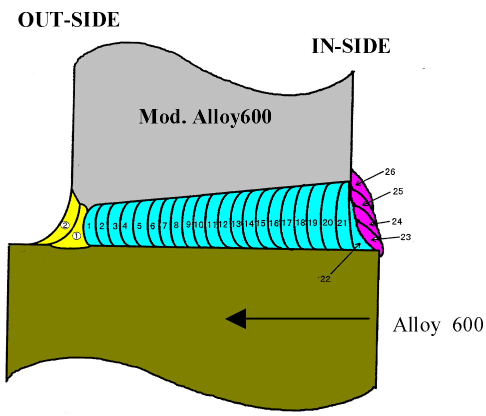

FIG. 4. Welding sequence.

These results showed that the deviation of the alignment of the shroud to the vessel had been prevented by two operating procedures designed to minimize distortion. One of these was to use simultaneously two sets of welding machines operating on opposite sides and the other was to change the starting point of each welding pass based on data from monitoring the alignment.

Deviation of the concentricity of the core plate and that of the top guide were measured in the same way. For both components the criteria were met. Deviation data at mockup test was summarized in Table II.

2.2.5. Structural integrity tests

The penetrant testing (PT) was performed on the inside weld after the first welding pass, the middle pass and the last pass instead of volumetric testing. No defects were found by PT on any pass. Two samples were cut out from the weld. No defect was observed by metallographic examination of the etched specimens (see Figure 5). The integrity of the welded joint was confirmed by these test results.

TABLE II. ALIGNMENT OF NEW SHROUD

INSTALLED COMPONENTS	Permissible deviation (mmDIA)	Measured deviation [tool-No 1 /tool -No2] (mmDIA)
CORESHROUD CENTER	6.0	1.6/3.6
CORE PLATE CENTER	3.8	3.1/2.9
TOP GUIDE CENTER	2.0	0.6/0.4

NOTE: 2 Alignment tools were used to measure the concentricity deviation

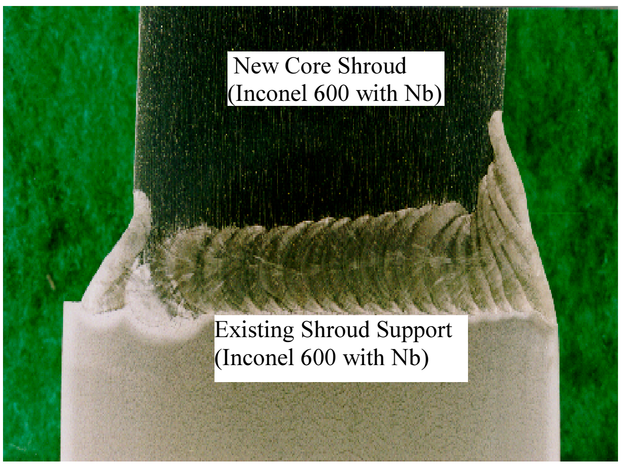

FIG. 5. Cross section view of shroud welding.

2.3. Structural integrity evaluation

Structural integrity of the replaced shroud was evaluated by the mockup test results and other study results including joint study program of utilities and manufacturers. To qualify installation welding, not only mockup test results but also tensile test, bend test, and hardness test data were evaluated. Investigations of material susceptibility, water chemistry, and stresses in the material against SCC were performed using the creviced bend beam (CBB) test. The low susceptibility of the Inconel 600 used for the replaced structures has been evaluated using the data referred to in the joint study program of utilities. These evaluation results are summarized in Table III.

2.4. Summary

The mockup test of the core shroud replacement method was performed in 1997. The test demonstrated the reliability of the prescribed construction procedures for shroud fit-up, shroud welding, core plate installation and top guide installation for a BWR-4 (800MW(e)).

The method was applied in the preventive maintenance of actual plants; the core shroud replacement of TEPCO's Fukushima Daiichi Unit #3 and Unit #2.

TABLE III. SUMMARY OF STRUCTURAL INTEGRITY EVALUATION

EVALUATED PORTION	ITEM	EVALUATION METHOD	RESULTS
Shroud/ Shroud Support Installation Welding	Quality of Welding	(1) NDT(VT,PT) of welding during mockup test. (2) Metallographic observation of the welding cross-section (3) Hardness measurements of the welding cross-section. (4) Mechanical property test - Tensile test - Bend test	(1) No defect was found by NDT and Metallographic examination (2) Mechanical properties of the welding portion were found to be good enough.
	Corrosion Resistance	(1) Creviced bend beam (CBB) test	(1) No Stress Corrosion Cracking was found in the new installation welding on this test.
	Residual Stress caused by Welding	(1) Measurements of welding residual stress	(1) Residual stress was found to be low enough
	Influence of Neutron Irradiation to the shroud support	(1) Evaluation of dissolved Helium concentration around the installation welding	(1) Analysis has shown that Helium generation by Neutron irradiation around welding portion was low.

3. BWR ICM HOUSING REPLACEMENT METHOD

3.1. Test plan

The method involves a repair technique (replacement) to avoid potential damage of the ICM housing due to stress corrosion cracking (SCC) of SUS304 housing welding and an evaluation of the structural integrity after the replacement. A comparison of the ICM housing structures before and after the replacement is shown in Figure 6. New ICM housing and lCM guide tube is made of SUS316L, which is a SCC resistant material.

The replacement method consisted of the four conceptual steps using full remote technique of machining, welding, and testing, to ensure the work at highly activated environment inside the Reactor Vessel. These four conceptual steps are described below and are illustrated in Figure 7. The full scale mock-up test was conducted through (2) to (4) steps described below that should influence structural integrity of new replaced structure.

(1) The existing ICM guide tube, housing and weld buildup are removed.
(2) A new weld buildup is formed and machined.
(3) A new housing made of SCC resistant material is inserted and welded to the weld buildup.
(4) A new guide tube is connected to the new housing by a coupling made of shape memory alloy (SMA).

3.2. Test results

3.2.1. Reliability of the replacement procedure

It was confirmed that all operation were carried out successfully and requirements were reviewed at each step.

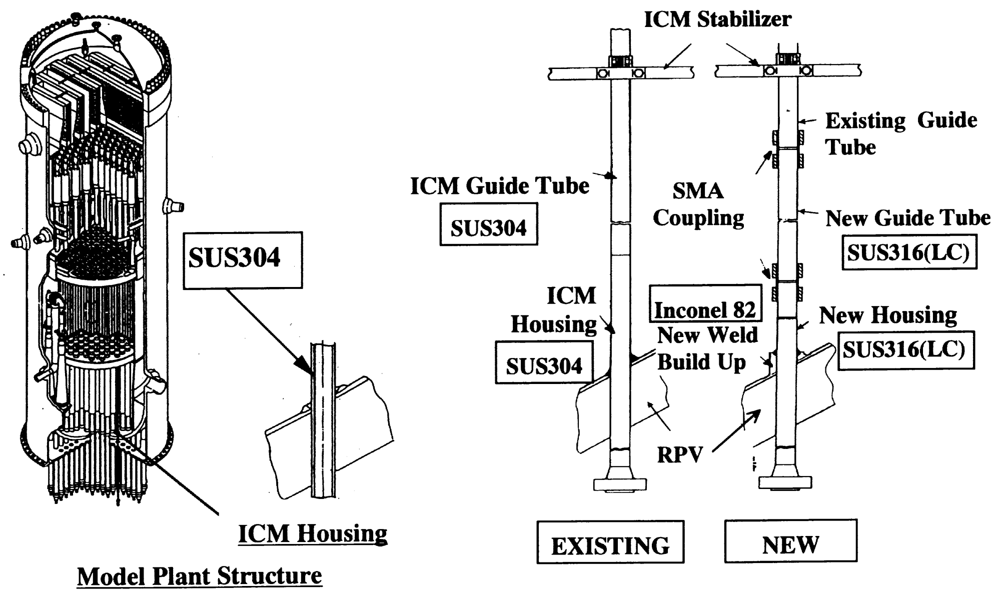

FIG. 6. New ICM housing structure.

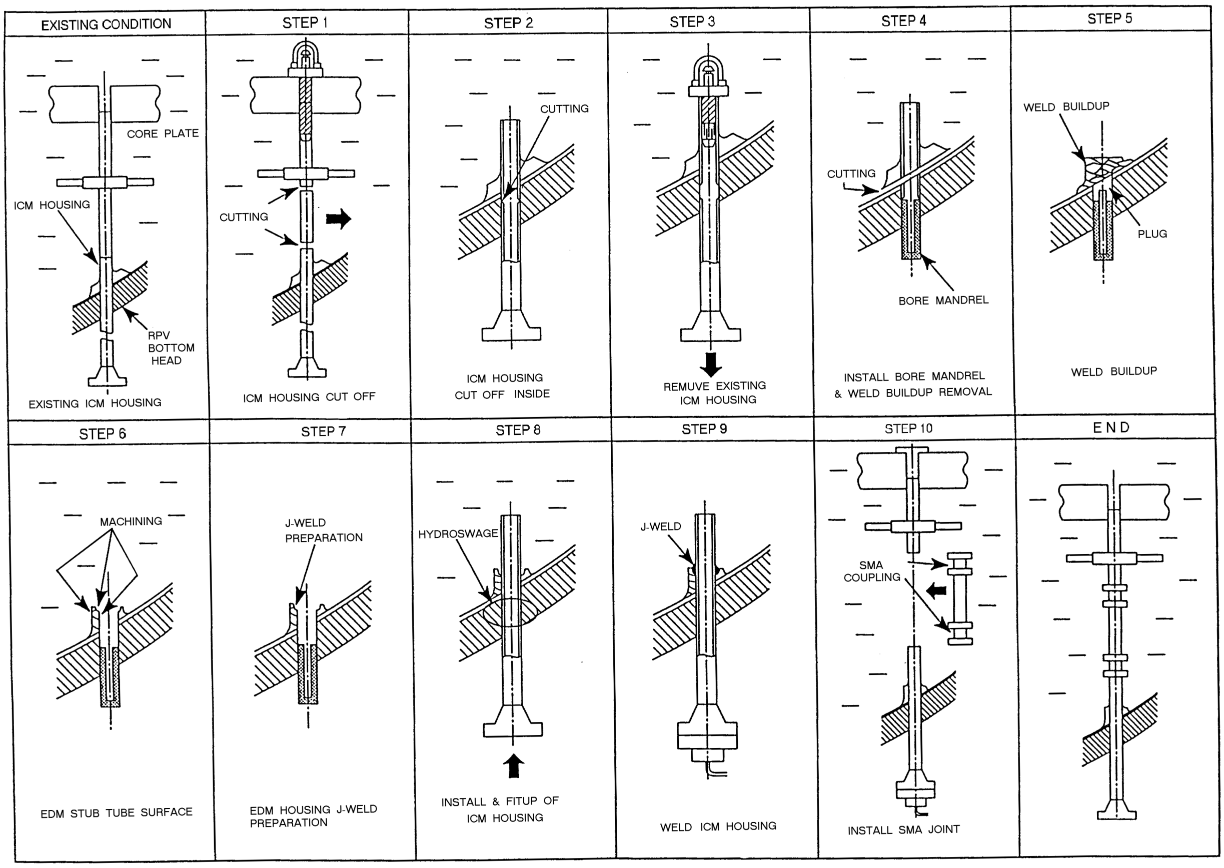

FIG. 7. ICM housing replacement sequence.

3.2.2. Welding buildup and machining

The new weld buildup was formed on the bottom surface of the reactor pressure vessel (RPV) using a remote controlled welding device and machined with electric discharge machining tools. No flaw was detected in the weld buildup by a penetrant testing (PT), and an ultrasonic testing (UT).

3.2.3. Housing connection welding

After installation in the RPV bottom, the new housing was connected to the weld buildup by a J-weld using a remote controlled welding machine. PT and UT inspections discovered no defect in the weld during the test or by subsequent visual checks.

3.2.4. SMA coupling joint

The new guide tube with an SMA coupling at each end was installed between the new lCM housing and the existing guide tube. The coupling was heated up and connected at the both ends. The structural adequacy of the connection was evaluated by an eigenvalue analysis compared the frequencies before and after the replacement and by a flow induced vibration analysis. Figure 8 shows the new ICM housing after SMA coupling installation.

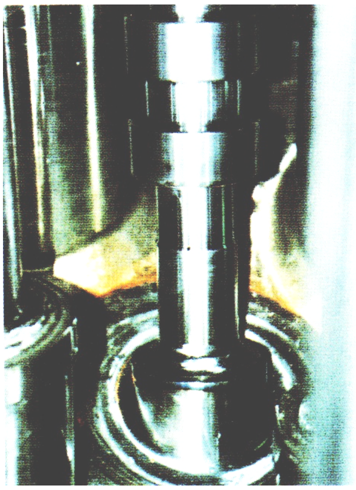

FIG. 8. New ICM housing with SMA coupling joint.

3.2.5. Structural integrity tests

PT and UT inspections of weld build-up and housing J-welding were performed. Metallographic examination of a section cut through the welding material and SMA couplings were performed after these nondestructive tests (Figure 9). No defect was found by these tests. Hardness of the welding material cross section was measured and the obtained data has shown good results.

3.3. Structural integrity evaluation

Structural integrity of replaced ICM housing is evaluated by the mockup test results and the results of joint study program of utilities and manufacturers. It has been confirmed that the dimensions of the welded part met the tensile test, bend test, and hardness test. Investigations of material susceptibility, water chemistry, and stresses in the material against SCC were performed by accelerated SCC test. The low susceptibility of the SUS316L and Inconel 82 used for the replaced structures has been evaluated using the data referred to in the joint study program of utilities. Evaluation results are summarized in Table IV.

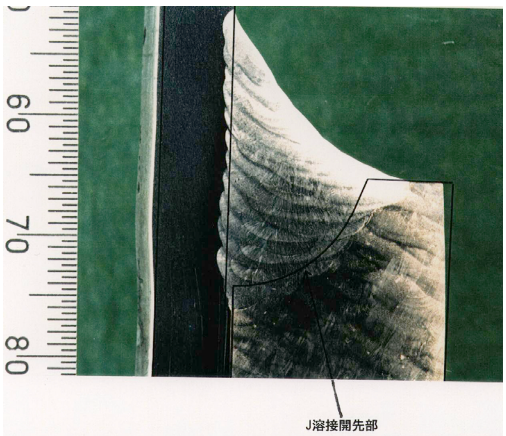

FIG. 9. Cross-section view of ICM housing J welding.

TABLE IV. SUMMARY OF STRUCTURAL INTEGRITY EVALUATION

EVALUATED PORTION	ITEM	EVALUATION METHOD	RESULTS
(1) Buildup Welding (2)Housing J- Welding	Quality of Welding -	(1) Non Destructive Test (PT, UT) (2) Metallographic observation (3) Hardness measurements (4) Constitution analysis (5) Mechanical property test. (Tensile test, Bend test)	(1) No defects were found by NDT and Metallographic observation (2) Mechanical properties of the welding portion were found to be good enough
	Corrosion Resistance	(1)Accelerated SCC test	(1) No SCC was found on the welding portion on this test
	Residual stress caused by welding	(1) Measurements of welding residual stress	(1) Residual stress was found to be low enough
	Thermal effects of the build-up welding to the RPV low alloy	(1) Metallographic observation of mockup test piece (RPV) cross section	(1) Observation results showed no metallurgical effects on low alloy
	Fatigue evaluation of J welding toe	(1) Fatigue analysis under estimated stress concentration factor	(1) Analysis shows small fatigue damage
(3) Shape Memory Alloy (SMA)	Characteristics of Shape Memory Alloy (Ni-Ti-Nb alloy)	(1) Composition analysis (2) Metallographic examination (3) Measurements of mechanical and thermal property (4) Corrosion resistance (5) Characteristics under high temperature environment (Creep, aging, transformation, etc) (6) Irradiation effect	(1) It was recommended that Ni-Ti-Nb alloy was suitable for In-Reactor use because of its good mechanical strength, corrosion resistance and irradiation characteristics
	Evaluation of SMA coupling parts	(1) Evaluation of shape recovery treatment temperature (2) Coupling strength test of SMA joint (3)Coupling strength test under repeated load (4)Corrosion resistance evaluation (5) Creep characteristics evaluation	(1) Mechanical joint strength of SMA coupling was evaluated under the seismic load, fluid vibration load and other thermal load under operating condition, and it was found to be suitable.

3.4. Summary

The test of the ICM housing replacement method was performed in 1996. The mockup test results have shown that the ICM replacement could be performed using the prescribed procedure. These evaluations indicated that the test of the ICM housing replacement method had demonstrated that the procedure was reliable. The method has been applied to the repair of an actual plant; ICM housing replacement was done at TEPCO's Fukushima Daiichi Unit #4 in 1997.

4. BWR CRD HOUSING & STUB TUBE REPLACEMENT METHOD

4.1. Test plan

CRD housings & stub tubes as shown in Figure 10 are the penetration for control rod drive mechanism and located on the RPV bottom head. For most of aged plants, CRD housings are made of SUS304, which has a SCC susceptibility similar to ICM housings. Figure 10 illustrates the CRD housing and stub tube structures before and after replacement.

The sequence of the CRD housing & stub tube replacement method is shown in Figure 11. Because of the structure and location of the CRD housings are similar to the ICM housings, some of the basic replacement technique such as electric discharge machining, remote UT and remote PT techniques are common technologies. On the other side, CRD housing needs severe alignment accuracy and special three-dimensional welding technique. Then the mock up test plan is optimized to demonstrate the characteristic technology of this method such as welding steps with on-line alignment measurements. Main steps of the mockup test are described below:

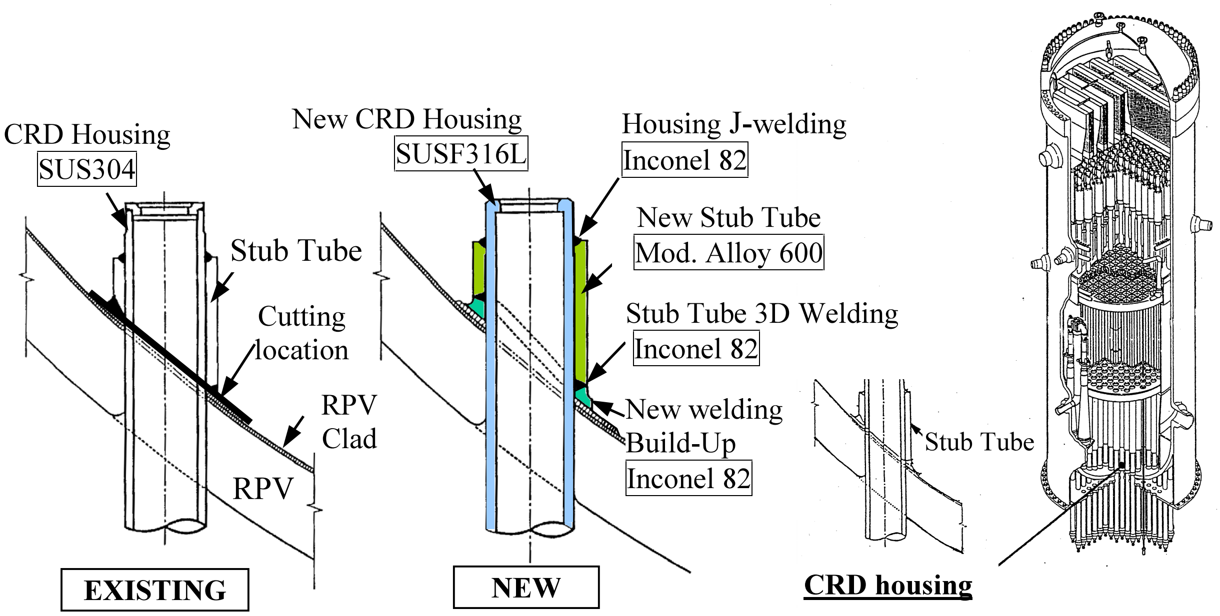

FIG. 10. New CRD housing and stub tube structure.

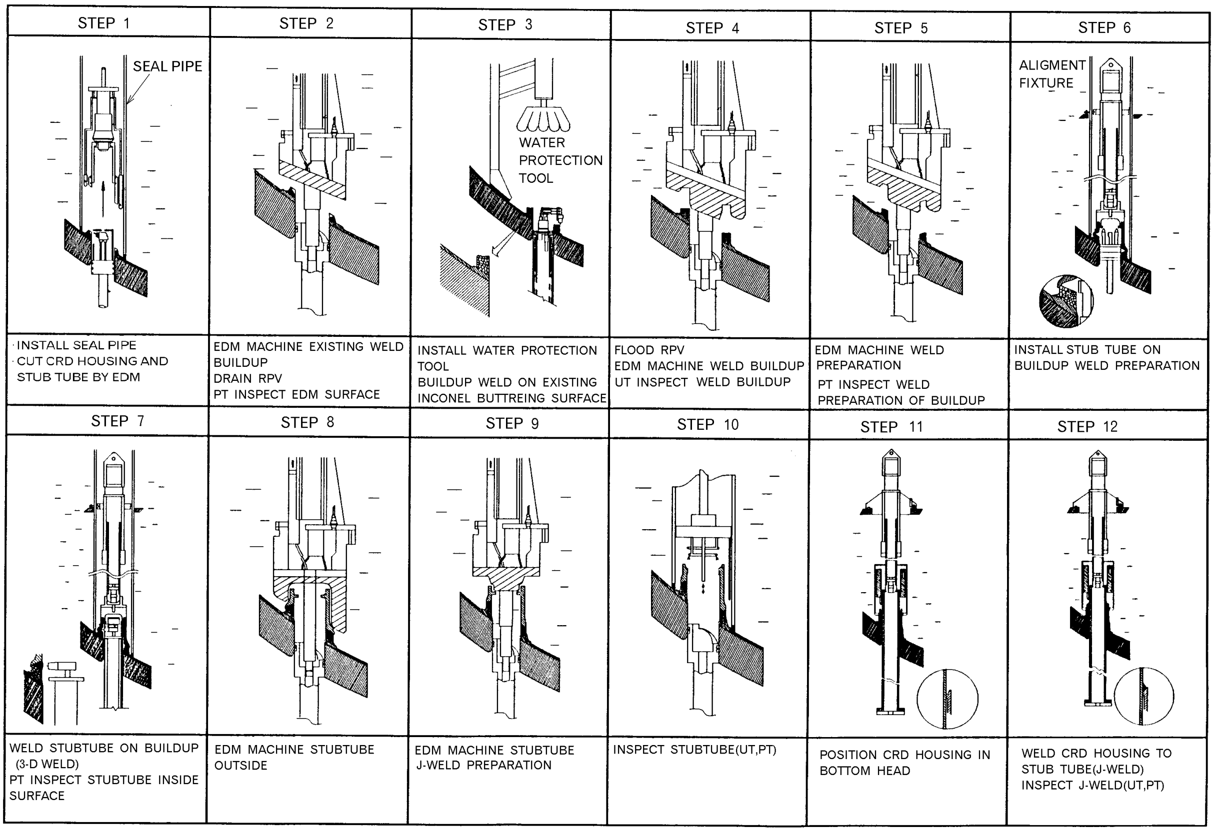

FIG. 11. CRD housing & STUB tube replacement sequence.

(1) Welding build-up and PT inspection of stub tube weld preparation.
(2) New stub tube fit-up and Stub tube 3-D welding with laser alignment tool
(3) Inside machining after stub tube welding and PT inspection
(4) New CRD housing fitting up and J welding with laser alignment tool

4.2. Test results

4.2. 1. Reliability of the replacement procedure

It was confirmed that all mockup test operation was carried out successfully and requirements were reviewed at each step. Other steps were also evaluated by the ICM housing replacement mockup test data and the results of joint study program. of utilities and manufacturers.

4.2.2. Welding buildup

The new welding buildup was formed on the bottom surface of the reactor pressure vessel (RPV) using a remote controlled welding machine. No flaw was detected in the weld buildup by a penetrant testing (FIT), and an ultrasonic testing (UT).

4.2.3. Stub tube 3D welding

The new stub tube was connected to the weld buildup by a three-dimensional shaped preparation welding after installation on the RPV bottom, using a remote controlled welding machine. To reduce radiation exposure of the worker, welding was executed under water for shielding with seal pipe device, which made partial dry condition. The alignment of stub tube was measured during and after welding as shown in Fig. 12.

After the installation, inside surface of the stub tube welding was machined by remote precise grinding machine. PT and UT inspections after ID machining discovered no defects in the connection.

4.2.4. Housing installation welding

The new CRD housing was installed inside the new stub tube and welded by the remote housing J welding machine. Figure13 shows the new stub tube and CRD housing after installation. The alignment of CRD housing was measured during and after welding. The test results met the required alignment deviation as shown in Table V.

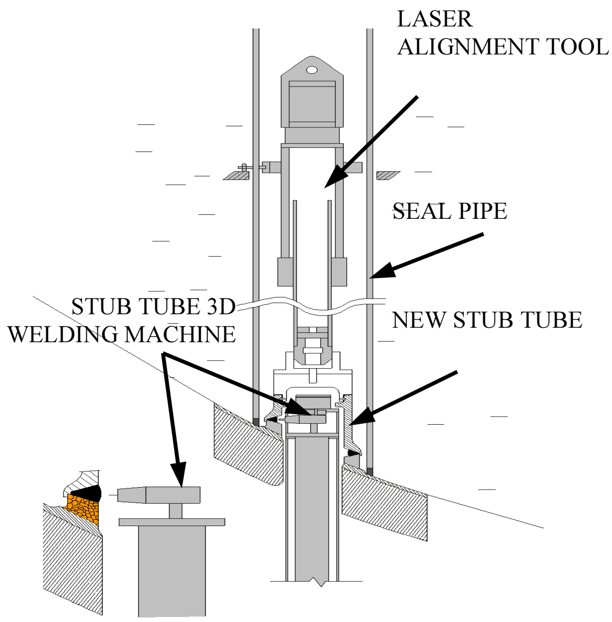

FIG. 12. New STUB tube welding.

TABLE V. ALIGNMENT OF STUB TUBE AND CRD HOUSING

INSTALLED COMPONENTS	Permissible deviation (mmDIA)	Measured deviation (mmDIA)
Stub tube center	1.5	0.7
CRD housing center	0.8	0.5

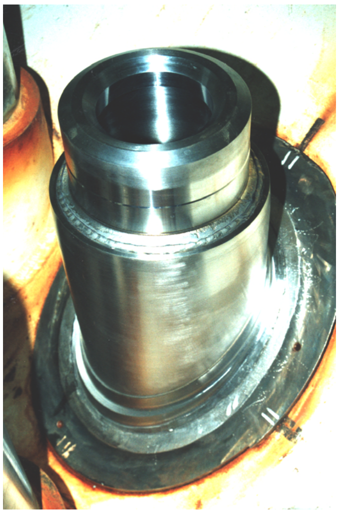

FIG. 13. New CRD housing and STUB tube.

4.2.5. Structural integrity tests

PT and UT examinations of build-up welding, stub tube welding and housing J-welding were performed. Metallographic examination of a section cut through the welding material was performed. No defects were found by these tests. Hardness of the welding material cross section was measured and the obtained data has shown good results.

4.3. Structural integrity evaluation

Structural integrity of replaced CRD housing & stub tube is evaluated by the mock up test results and the results of joint study program of utilities and manufacturers. Evaluation results are summarized in Table VI.

4.4. Summary

The test of the CRD housing replacement method was performed in 1998. The mockup test results have shown that the replacement could be performed using the prescribed procedure and these evaluations indicated that the test of the CRD housing & stub tube replacement method had demonstrated that the procedure was reliable.

5. SUBSEQUENT EFFORTS

As shown in Table I, NUPEC is now executing the reliability test for PWR Core Barrel replacement method, from 1997 to 2000. Full-scale mockup test will be demonstrated in the early 2000. Two additional tests of PWR BMI adapter replacement method and BWR jet pump riser brace replacement method are planned after the Core Barrel replacement method mockup test. These test results will be reported after their reliability tests have finished.

TABLE VI. SUMMARY OF STRUCTURAL INTEGRITY EVALUATION

EVALUATED PORTION	ITEM	EVALUATION METHOD	RESULTS
(1) Buildup Welding (2) stub tube welding (3) Housing J-Welding	Quality of Welding	(1) Non Destructive Test (PT, UT) (2) Metallographic observation (3) Hardness measurements (4) Mechanical property test (Tensile test, Bend test)	(1) No defects were found by NDT and Metallographic observation (2) Mechanical properties of the welding portion were found to be good enough
	Corrosion Resistibity	(1) Creviced Beam Bend test	(1) No SCC was found on the welding portion on this test
	Residual stress caused by Welding	(1) Measurements of welding residual stress	(1) Residual stress was found to be low enough
	Thermal effects of the build-up welding to the RPV low alloy	(1) Metallographic observation of mockup test piece (RPV) cross section	(1) Observation results showed no metallurgical effects on low alloy
	Fatigue evaluation of J welding toe	(1) Fatigue analysis under estimated stress concentration factor	(1) Analysis shows small fatigue damage

6. CONCLUSIONS

Plans to test the technologies used in the methods to rejuvenate reactor internals and results obtained so far have been presented. Especially, a summary has been given to show how the reliability of various steps in the replacement work such as machining, welding, inspection, and so on have been demonstrated in the tests using full scale model mockups. It is concluded that NUPEC activities concerned with countermeasures for the aging degradation of nuclear plants have been successful in Japan because the ICM and the shroud replacement methods have already been applied to actual plants after the NUPEC tests were completed.

ACKNOWLEDGEMENTS

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EFFORTS FOR OPTIMIZATION OF BWR CORE INTERNALS REPLACEMENT

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Abstract

The core internal components replacement of a BWR was successfully completed at Fukushima-Daiichi Unit #3 (1F3) of the Tokyo Electric Power Company (TEPCO) in 1998. The core shroud and the majority of the internal components made by type 304 stainless steel (SS) were replaced with the ones made of low carbon type 316L SS to improve Intergranular Stress Corrosion Cracking (IGSCC) resistance. Although this core internals replacement project was completed, several factors combined to result in a longer-than-expected period for the outage. It was partly because the removal work of the internal components was delayed. Learning a lesson from whole experience in this project, some methods were adopted for the next replacement project at Fukushima-Daiichi Unit #2 (1F2) to shorten the outage and reduce the total radiation exposure. Those are new removal processes and new welding machine and so on. The core internals replacement work was ended at 1F2 in 1999, and both the period of outage and the total radiation exposure were the same degree as expected previous to starting of this project. This result shows that the methods adopted in this project are basically applicable for the core internals replacement work and the whole works about the BWR core internals replacement were optimized. The outline of the core internals replacement project and applied technologies at 1F3 and 1F2 are discussed in this paper.

1. INTRODUCTION

Since the core shroud cracking was found [1], the Japanese BWR owners and the plant manufacturers have conducted several R&D programs to establish the countermeasures for core shroud cracking. Based on the evaluation of the effect and applicability of them, TEPCO decided to replace the core shroud made of 304SS which has relatively higher possibility of IGSCC. Among 17 BWR plants owned by TEPCO, four plants have the core shrouds made of 304SS. In accordance with the long term outage schedule, 1F3 was selected as the first plant to replace the core shroud with the one made of 316L SS which has much less possibility of IGSCC [2].

In addition to the core shroud, the majority of the internal components made of 304SS are replaced at the same outage. These are (1) core shroud, (2) top guide, (3) core plate, (4) core spray spargers, (5) feed water spargers, (6) jet pumps, (7) differential pressure and liquid control system (DP/LC) pipe, (8) in core monitor (ICM) guide tubes, and internal pipes and nozzle safe ends connected to these components as illustrated in Figure 1.

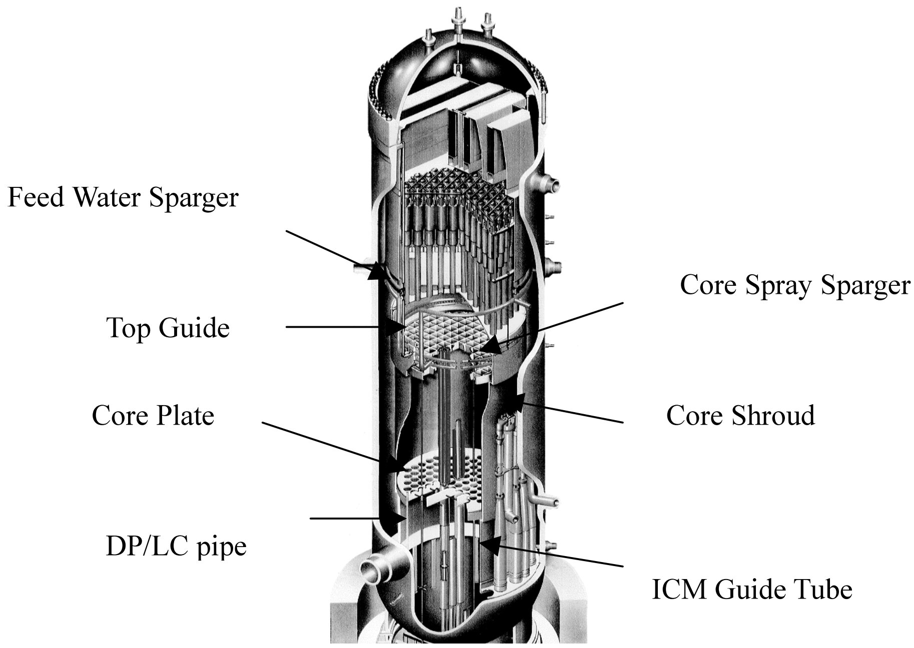

FIG. 1. Replaced components.

2. CORE INTERNALS REPLACEMENT AT 1F3

During the 16th outage of 1F3 started in May 1997 the core internals replacement project had been carried out, and 1F3 was back on line in July 1998.

Reduction of radiation dose level is one of the key elements through the core internals replacement work because of the workers to be able to enter inside the reactor pressure vessel (RPV) for installation work of new components. Prior to the removal work the chemical decontamination was conducted to reduce the radiation from the radioactive metal oxide deposited on the components, and the in-vessel shielding panels were set after the removal work to reduce the radiation from the irradiated RPV. As a result of them, the in-vessel dose level decreased generally below 1.0mSv/h, and it was nearly 0.2mSv/h at the bottom of RPV where operations were performed for many hours [3].

The Removal of the core shroud and most of other internals was conducted by means of Electrical Discharge Machining (EDM) process underwater [5]. Subsequently, the removed core shroud and other internals were sliced into small pieces, and put in the containers in the Dryer Separator Pool (DSP). The underwater plasma slicing was employed for the slicing process because of its slicing speed. The stuffed containers were decontaminated and transferred to the site bunker pool and the dry storage facility in the site.

The period of this outage, including the core internals replacement, was initially expected to be 300 days, but the actual period of outage was 423 days. It was partly because the removal work of the internal components was delayed. The main reason of delay is that, considering the core shroud and jet pumps replacement was the first experience, a cautious attitude was taken in carrying out replacement work to secure safety and reliability. The initial plan imposed rather stringent requirement. All these factors combined to result in a longer-than-expected period for the outage.

3. IMPROVEMENT FOR THE REPLACEMENT WORK AT 1F2

As the 1F3 core internals replacement work ended, an experience to implement the world's first welded core shroud and jet pumps replacement was acquired. Hence, many examinations for shortening a shutdown period had been conducted to gain economical efficiency in next replacement work at 1F2. Reflect on a delay in replacement work at 1F3, the schedules of all works were corrected adequately ,considering about the results at 1F3.

And at the same time, for some works, conspicuous about delay and radiation exposure, the new methods were adopted. Those works are removal work and slicing work and jet pumps installation work.

3.1. Improvement for removal work

3.1.1. Appropriation for removal process

At 1F3 removal work of core internal components was conducted by the EDM process and the mechanical cutting process. And most of the components were removed by means of the EDM process for its technological reliability. The EDM process allows a very high degree of precision and complicated operation remotely. So the EDM process is suitable to cut the components precisely underwater as previously planned. However the cutting speed is low and it takes long time to set the machine to the right position.

To shorten the schedule of removal work, it was took into consideration to enlarge the scope of adoption of the mechanical cutting process at 1F2. The mechanical cutting process, using some conventional tools, is appropriate for cutting the small component rapidly which has simple shape.

Therefore considering the acquired conditions of each components after cutting, adequate process were selected for each components. The EDM process was adopted basically to the part of component which is to be prepared for welding with new components. And the mechanical cutting process are adopted to the part no need to be cut precisely. The core spray pipings and guide rods are cut by sawing. DP/LC pipe and ICM stabilizers are cut by hydraulic cutting. The bolts of top guide and core plate are removed by pliers.

As the result of these improvement, the total schedule of removal work was shortened compared to 1F3's by 30 days approximately.

3.1.2. Adoption of mechanical cutting tool for core shroud

The mechanical cutting process was adopted to core shroud too. The core shroud is cut at an upper and lower location to be sliced underwater in the DSP. As an alternative to the

EDM process, it is interested in using a mechanical roll cutting technique to sever the upper core shroud section to shorten the cutting time. The roll cutting tool, as illustrated in Figure 2, consists of 10 cutting modules and a support structure. Each module has one cutting blade. The blades are pressed by and rotated inside core shroud by hydraulic and electric motors. After the mock-up test, this tool was adopted at 1F2. As the result of the adoption of this tool, the schedule of cutting the upper core shroud were shortened compared to 1F3's by 10 days approximately.

3.2. Improvement for slicing work

3.2.1. Appropriation of sequence for removal and slicing work

The internal components removed from RPV are sliced into small pieces in the DSP and stuffed in the containers to be transported. To store dryer and steam separator in the DSP, only a half of the whole space in the DSP is available for slicing work. So when the slicing work is delayed, sliced components cannot be transported from the DSP, and then the components removed from RPV cannot be carried into the DSP.

That is to say, the removal work and the slicing work are closely related each other. Actually the removal work was detained by delay of the slicing work at 1F3. So it is very important to examine the whole sequence considering many factors other than cutting and slicing speed. Those are the time for setting tools, the order of slicing, the arrangement of components and tools in the DSP, and utilities available at refuel floor for instance. Additionally a new process of slicing was adopted at 1F2 to shorten the schedule of slicing work.

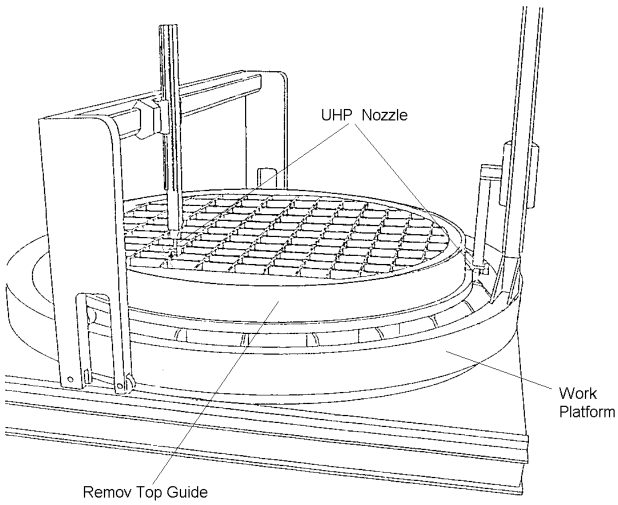

FIG. 2. UHP slicing tool.

3.2.2. Adoption of new slicing process

At 1F3 the plasma slicing process was adopted because of its slicing speed. But the slicing work took time longer-then-expected. Mainly because the sensitivity in setting the plasma torch to the right position. As an alternative to the plasma slicing process, a new slicing process was examined. The ultra high pressure abrasive water jet (UHP-AWJ) slicing process was adopted at 1F2. The UHP-AWJ process mechanically cuts off components by throwing a stream of abrasive-containing high pressure water to them. It is a practical method that has been used lately in the bolt-type core shroud replacement work at the Oskarshamn Nuclear Power Station in Sweden as illustrated in Figure 3.

3.3. Improvements for the installation works at 1F2

The installation works of new internal components start with jet pumps installation work and then continues to core shroud installation work. After that, the other components are installed, those are core plate, top guide, ICM guide tubes, DP/LC pipe and so on. Most of installation works were made good progress at 1F3. But as concerns about jet pumps installation work, the schedule was longer than previously expected. And the radiation exposure of this work is relative large compare to another works. So the examination was conducted to shorten the schedule and reduce the radiation exposure of this work.

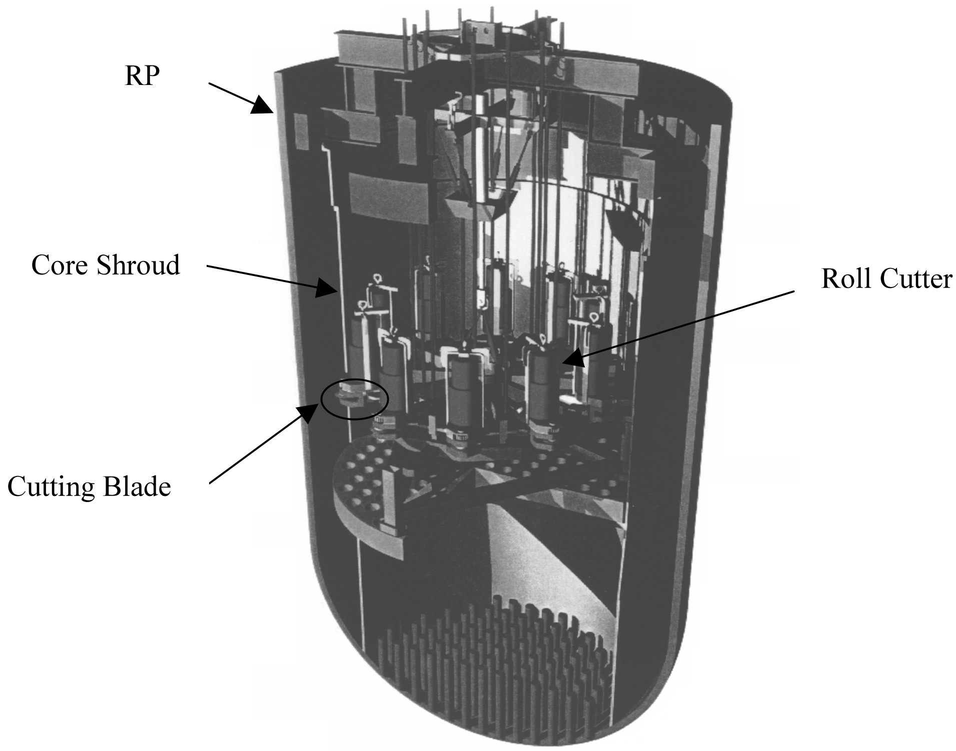

FIG. 3. Roll cutting tool.

3.3.1. Improvement for jet pump raiser brace installation work

There are 20 jet pumps in the RPV and one jet pump is mainly consists of three assemblies, those are raiser assembly and inlet-mixer assembly and diffuser assembly. As the result of estimation for works to install those tree assemblies, the raiser brace installation work was focused to be improved. Raiser brace is one of the parts of raiser assembly to join raiser pipe with RPV by welding. A raiser brace has four thin plates, called raiser brace leaves, and the tip of them are directly attached to the cladding pad inside RPV at core region. Therefore to install raiser brace, the in-vessel shield panels must be opened partially for welding. The automatic welding machine was adopted to decrease the radiation exposure at 1F3. But setting and testing operation previous to welding could not be done remotely. So workers operated closely to the welding point. Therefore the radiation exposure of these operations was relative large compared to the other operations.

To reduce the radiation exposure, a new welding machine was adopted at 1F2. This machine, as illustrated in Figure 4, has two manipulators which has six axis to move independently. With the large degrees of freedom, setting and testing operation can be done remotely. Therefore the time of those operations became shorter and the radiation exposure was reduced.

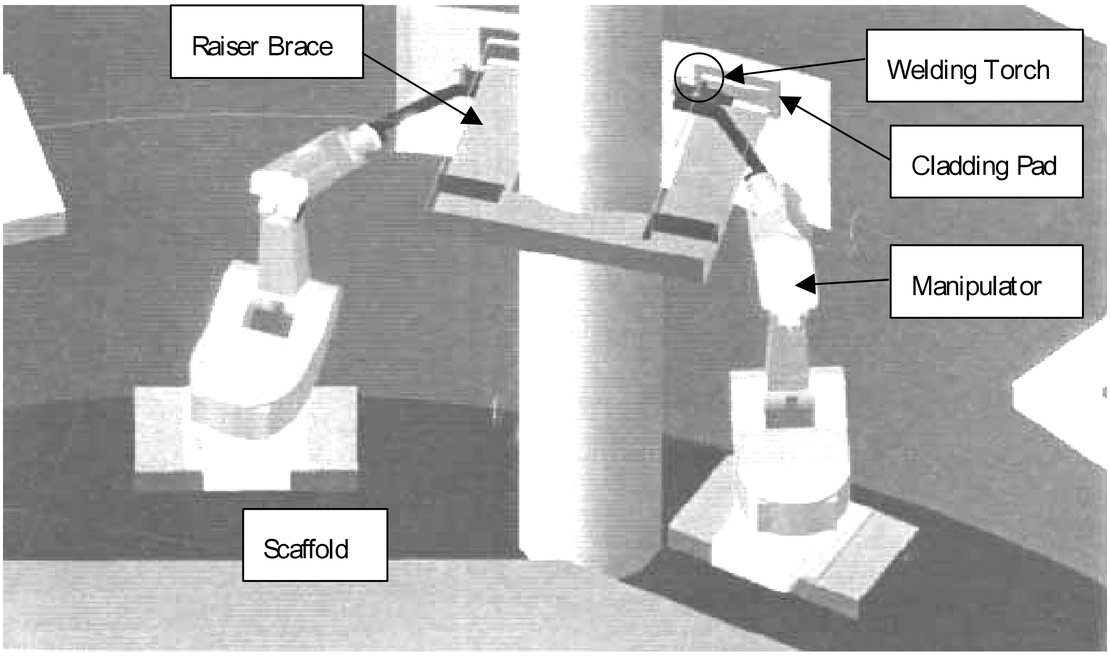

FIG. 4. Raiser brace welding machine.

3.3.2. Result of improvement

In addition to the improvement for the raiser brace installation work, some examination were planned to reduce the total radiation exposure. As the way to access to working area in RPV, an elevator was attached to reduce the burden of workers and the radiation exposure at

1F3. But the station of the elevator was located on the top floor of the scaffold where existed at the core region in RPV and relative high dose level. So the location of the station was changed to the top of scaffold, as illustrated in Figure 5, to reduce the radiation exposure.
As the result of those improvements, as mentioned above, the schedule of jet pumps installation work was shortened compared to 1F3's by 30 days approximately. And the radiation exposure of this work was reduced compared to 1F3's

3.4. Result of core internals replacement work at 1F3

The 17th outage of 1F2, including the core internals replacement work, was carried out and back on line in July 1999. The period of outage and the total exposure at 1F3 and 1F2 are shown in the Table I. Considering about them, many kinds of improvements, as mentioned above, were functioned effectively and resulted in shortening the period of outage and reduction of total radiation exposure.

At the same time, the period of the outage was planned to be 334 days, and actual period was 331 days. There is no significant difference between the planning and actual period, so it seems that the schedule and sequence of core internals replacement work were almost optimized.

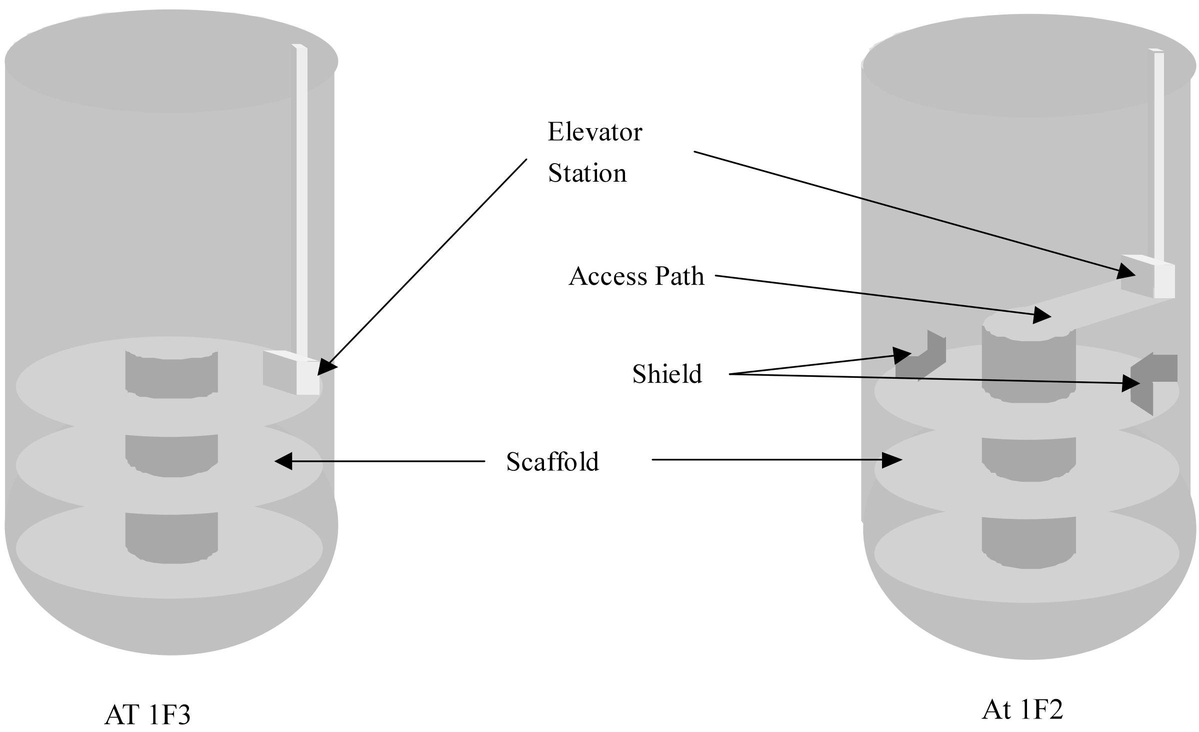

FIG. 5. Circumstances in RPV.

TABLE I. COMPARISON OF CORE INTERNALS REPLACEMENT WORK AT 1F3 AND 1F2

Unit	Period of outage (days)	Total radiation exposure (person Sv)	Points of improvements
1F3	423	11.5	-
1F2	331	7.7	• Enlargement of the scope of mechanical cutting • Adoption of water jet slicing • Adoption of new welding machine for jet pumps Installation

4. CONCLUSION

The core internals replacement project at 1F3 were successfully completed. Because of the several factors, the period of outage at 1F3, including the core internals replacement, was longer-than-expected. To shorten the schedule, some works were focused to be improved. Those were removal work, slicing work and jet pumps installation work. So some examinations about work sequence and new processes were conducted prior to the internals replacement work at 1F2.

As the result of adoption, the period of outage at 1F2 was 331 days, almost as long as expected, and the total radiation exposure was reduced. Therefore the core internals replacement work is seemed to be optimized and the methods, those were adopted at 1F2, are basically applicable for core internals replacement work.

And now TEPCO is planning the core internals replacement work at Fukushima-Daiichi Unit#5. The period of outage and the total radiation exposure are expected to the same degree as 1F2's approximately.

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METHODS FOR REDUCING REFUELING OUTAGE DURATION

(Session III)

Chairperson

N.S. FIL
Russian Federation

KEY ISSUES FOR THE CONTROL OF REFUELING OUTAGE DURATION AND COSTS IN PWR NUCLEAR POWER PLANTS

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Abstract

For several years, EDF, within the framework of the CIDEM project and in collaboration with some German Utilities, has undertaken a detailed review of the operating experience both of its own NPP and of foreign units, in order to improve the performances of future units under design, particularly the French-German European Pressurized Reactor (EPR) project. This review made it possible to identify the key issues allowing to decrease the duration of refueling and maintenance outages. These key issues can be classified in 3 categories: Design; Maintenance and Logistic Support ; Outage Management. Most key issues in the design field and some in the logistic support field have been studied and could be integrated into the design of any future PWR unit, as for the EPR project. Some of them could also be adapted to current plants, provided they are feasible and profitable. The organization must be tailored to each country, utility or period: it widely depends on the power production environment, particularly in a deregulation context.

1. INTRODUCTION

The duration of refueling outages is an important factor for the improvement of availability and maintenance costs of a Nuclear Power Plant, and therefore to ensure its competitiveness.

For several years, EDF, within the framework of the CIDEM[1] project and in collaboration with some German Utilities, has undertaken a detailed review of the operating experience both of its own NPP and of foreign units, in order to improve the performances of future units under design, particularly the French-German EPR project.

This review, made it possible to identify the key issues allowing optimization of the duration of refueling and maintenance outages.

These key issues can be classified in 3 categories:

- Design
- Maintenance and Logistic Support
- Outage Management

Most of these key issues are directly linked to the design, widely discussed with the EPR Designer, NPI (Framatome & Siemens). But many fruitful discussions, during the Basic Design Phase of the project, with German operators involved in the project have made it possible to emphasize some maintenance and management practices which allows them to realize refueling and maintenance outage shorter than in France.

[1] French acronym for Design Integrating Availability Experience Feedback and Maintenance

2. KEY ISSUES

2.1. Design

One of the major contribution of EPR studies has been the detailed analysis of the refueling and maintenance outage very early in the design phase.

This has made it possible to define and to take into account upstream design requirements allowing a reference duration of 16 days (from breaker to breaker). Then, key issues which are presented below are applicable to new plant under design. Their application to current units needs initial studies on their technical feasibility and profitability.

2.1.1. Main phases of the refueling and maintenance outage

The main phases of the outage have been studied in detail and the critical points concerning the design related to the process, layout and components, directly impacting the outage duration have been clearly emphasized.

2.1.1.1. Reactor cooling

One of the main objectives is to cool the vessel head as soon as possible to 70 °C, temperature from which it is possible to prepare the vessel opening.

As an example, the procedure defined for the EPR is described below:

- cooling and depressurizing of the primary circuit carried out automatically,
- cooling from 296°C (hot shutdown condition) to about 120°C (RHR start-up condition) conducted automatically by the steam generators, with 4 reactor coolant pumps in service. Cooling starts 5 hours after the control rods drop for a planned outage, and 3 hours after it for a forced outage. During this operation, RHR is conditioned automatically,
- when the temperature of 120°C is reached, RHR is started,
- cooling from 120°C to 50°C is conducted automatically by RHR. The objective is to reach the cold shutdown conditions (RCS temperature of 50°C and vessel head temperature less than 70°C) about 16 hours after the control rods drop,
- cooling after RHR start-up would have to be continued with steam generators and condenser by-pass as so long as it can operate.

The number of MCP in operation during this phase has to be determined by the Designer. It depends on the design of the internals. *The main requirement to consider is the time necessary to reach 70°C on the vessel head.*

2.1.1.2. Two-phases cooling

A two-phase cooling seems to be a very efficient practice to reach the cold shutdown conditions. This requires:

- decreasing the minimum pressure for reactor coolant pumps
- increasing the ΔT value endured by the pressurizer surge line (set at 110°C on many current units).

The best way is to decrease the RCS minimum pressure for RCS pump starting conditions.

Provisions have to be made to avoid all problems of thermal shocks or fatigue in all spray lines (normal and auxiliary), for any MCP in service arrangement.

The solving of the thermal shock problems has to be proved by the designer.

The authorized cooling gradient for the pressurizer automatically controlled could be 100°C per hour.

2.1.1.3. Purification and gas sweeping

The purification and gas sweeping flow is such that the required radiological conditions for opening the primary circuit are reached less than 30 hours after the control rods drop (reactor shutdown).

The duration of the gas sweeping could be about 6 hours (the vacuum system could be used for this operation).

2.1.1.4. Vessel opening

For this operation, the best way is to use an optimized Multistud Tensioning Machine (MTM), despite its cost. In this case, a duration of less than 10 hours could be reached.

The Rod Cluster Control Assemblies (RCCA) can be disconnected in less than 6 hours with a special tool.

2.1.1.5. Fuel unloading

To simplify the handling devices for the core unloading, a good practice is to realize a total core unloading operation at each outage.

New Technologies for Fuel Handling Machine allow a core unloading rate of 6 fuel assemblies per hour, at least.
The fuel handling machine underwater test activities, before core loading, which are on the critical path, have to be optimized and could be realized in 2 hours.

2.1.1.6. Design of cooling systems

Reactor Cavity and Spent Fuel Pit Cooling System, Component Cooling System and Essential Service Water System have to be carefully designed in accordance with the fast RCS cooling procedure and fuel unloading rate to allow handling of the first fuel assembly very soon after the reactor trip. As an example, for the EPR project, the first and the last fuel elements are respectively handled 71 hours and 111 hours after reactor trip.

2.1.1.7. Safety trains

The number and the architecture of safety trains have a strong impact on the outage progress and, more generally, on the maintenance of the plant.

Within the framework of of the EPR project, very detailed comparison of the French (1300MW series) and German (Konvoi series) current units have shown the interest of a 4x50% safety train design.

This option adds a considerable bonus to performance in several ways.

- Firstly it allows preventive maintenance of a certain number of safety equipment during power operation, thus reducing the amount of maintenance to be carried out during shutdown.
- Likewise, this system of four independent and physically separated trains, thanks to the redundancy it gives, allows carrying out preventive maintenance on the entire safety train as soon as the reactor pool is filled prior to unloading. Engineered-safety train maintenance is never, because of this, on the shutdown critical path.
- What is more, the fact that a complete train can be taken out of service at the beginning early and furthermore for a sufficiently long length of time, involves only one acceptable tag-out whilst of course fully satisfying safety requirements (one whole train) in a wide field of application which only has a small number of isolation devices to lock-out in fail-safe position. This way of working has the advantage of reducing operating teams workload in matters of intervention tag-out equipment. It is vital not to underestimate this load insofar as the length of shutdowns is becoming very short.
- Another result, also, of this four independent train structure, for four-loop reactors, is that the safety equipment is smaller than the corresponding machinery of a two-train structure. This gives the obvious advantage of a saving in intervention time during the outage if a maintenance policy of replacement for rotating equipment is adopted; as the servicing of the equipment replaced is carried out in workshops when the plant is operating at power.

Electrical interconnections between trains

Electrical interconnections between certain train switchboards could also be provided. These allow:

- undertaking preventive maintenance of DC batteries and diesels, in operation
- during outages, doing preventive maintenance on electrical switchboards which cannot be switched off when the plant is operating at power.

2.1.1.8. Steam generators inspection

In many countries, regulation requires all tube bundles must be controlled once, at least, during each 8 or 10 year operating period.

To reduce the inspection duration, some dispositions could be adopted, such as:

- manholes opening and closing works are robotized,
- steam generators are designed with nozzle dams installed by robots,

– plugs have to be installed at the nozzle level before the works begin, to avoid falling into the loops of foreign material.
– all the 4 steam generators are controlled in parallel and simultaneously with the use of improved robots

2.1.1.9. Low level nozzle works

The duration of these works must be as short as possible. With optimized practices, they could not exceed 60 hours.

2.1.1.10. Core refueling

The loading rate could be the same as for the unloading, i.e. 6 fuel assemblies per hour at least, To take into account the difficulties which may occur during the loading of some strongly bent assemblies, some additional devices to the Fuel Handling Machine such as shoehorn or dummy fuel element could be used.

An optimized core mapping is possible in less then 3 hours.

2.1.1.11. From end of loading to vessel head seal installation

During this phase, some decontamination works (vessel flange cleaning, pool draining and cleaning) have to be performed on the critical path of the outage. These works should be easily optimized and robotized.

2.1.1.12. From vessel head installation to reactor cooling system closing

During this period, most of activities is on the critical path. With the use of the MTM, this phase could be completed in less than 12 hours.

2.1.1.13. Venting and filling of the reactor cooling system

The most efficient way to vent and to fill the RCS is to use a vacuum system.

2.1.1.14. From start up to hot shutdown state

When possible, the start up could be conducted in a two-phase process, RHR shutdown but available as soon as the reactor coolant pumps start. This process allows a full heating capacity of the pumps.

The conditioning of the main steam system up to the turbine stop valves and of the reactor coolant system can be achieved.

2.1.1.15. Tests before synchronization

It should be possible to realize reactor protection system tests without any influence on the critical path.

2.1.1.16. Power increasing

This phase has very important influence on the availability.

A very important point concerns the In-core Instrumentation and the Flux Map Acquisition System. The solution adopted for the EPR is the Aeroball System which presents some advantages such as:

- no bottom vessel head penetration,
- short stabilization period before flux map acquisition (few hours)
- proven technology.

Another issue is to remove the flux map at low power and to realize a flux map at an intermediate power level (50-70%) before a complete flux map at 100% power, after a long stabilization period. The suppression of the low power flux map needs to develop a procedure to verify that the fuel loading is correct before installation of the vessel internals.

With such dispositions, operation at full load is possible within 24 hours following the unit synchronization to the grid.

The last important point during this phase is the Technical Specification limiting the power increasing rate to 3% per hour due to the Pellet Clad Interaction phenomenon, currently encountered in many countries (USA, France, etc.). This issue is not directly linked to the plant design but more to the fuel. Relaxation of this constraint could save more than 10 hours.

2.1.2. Other issues

2.1.2.1. Workshops, handling and storage area

For the control of outage duration, it is very important to have some surfaces available such as workshops and handling or storage areas in the vicinity of the equipment hatch.

As an example, a hot workshop could be installed in front of the equipment hatch. This allows increasing the working area for:

- storage of the Multi-Stud Tensioning Machine,
- cleaning and control of the vessel studs,
- storage of the vessel dummy head,
- handling of large components,
- etc.

The containment could be extended to the limits of the workshop when the RCS is depressurized and the hatch opened. This extension avoids several openings of the hatch during the outage.

This workshop allows the opening of the equipment hatch when RCS temperature is under 100°C and RCS is depressurized.

When the equipment hatch is open (primary circuit depressurized), the containment function is extended to the workshop limit assisted by a ventilated anti-contamination system. This device should resolve any radioactive release outside the containment in case of fuel element drop during core unloading.

A complementary hot workshop could also be designed for the following purposes:

- total maintenance of main coolant pumps and motors,
- maintenance of valves (from the containment),
- maintenance of pumps, for example RHR pumps,
- maintenance of valve drives and motors,
- maintenance of tools stored inside the containment during power operation.

2.1.2.2. Equipment hatch handling

In order to ease the burden of the polar crane, a specific equipment hatch handling should be provided to allow opening and a normal closing in 2 hours and a rapid closing in 30 minutes. If the design of the hatch needs a leaktightness test, it would not exceed 2 hours.

2.1.2.3. Accessibility to the containment during power operation

It should be interesting to have access to the Reactor Building during power operation, some days before the outage (about 10 days) and after startup (about 5 days), to prepare the refueling and maintenance outage, to close work sites after startup and to carry out, if necessary, limited interventions.

For this purpose, the basic EPR concept provides a cover over the entire pool of the Reactor Building by very thick concrete slabs which thus ensure effective biological protection. The specific arrangements chosen concern also ventilation systems (Reactor Building ventilation in open circuit as in Konvoi series) and guarantee satisfactory intervention conditions for personnel.

Main operations carried out inside the containment boundary before shutdown concern commissioning and requalification of the Reactor Building polar crane, the "valves check round" in the accessible part as a whole, and preventive maintenance of the part of Reactor Building ventilation (filtration and dome ventilation) which must remain in operation during shutdown. The expected gain in outage time is 2 to 3 days.

2.1.2.4. Polar crane

Maintenance and testing of the polar crane could be possible without any effect on the critical path, according to the regulations. In case of accessibility to the Reactor Building before deconnection, this test would be realized just before unit shutdown.

2.1.2.5. Piping control

These controls can be automatized with the use of displacement sensors and remote control devices.

2.1.3. Effect on outage duration

The design options presented above have a significant influence on outage duration.

Their implementation for the EPR project have allowed a reduction from 31 to 16 days (from breaker to breaker) of the duration of the refueling and maintenance outage, compared with the 1997 reference outage of EDF's current 1300 MW series. The loading to 100% phase is reduced from 4 to 1 day. This evaluation has been made

The Figure 1 gives the gain for each phase of the outage.

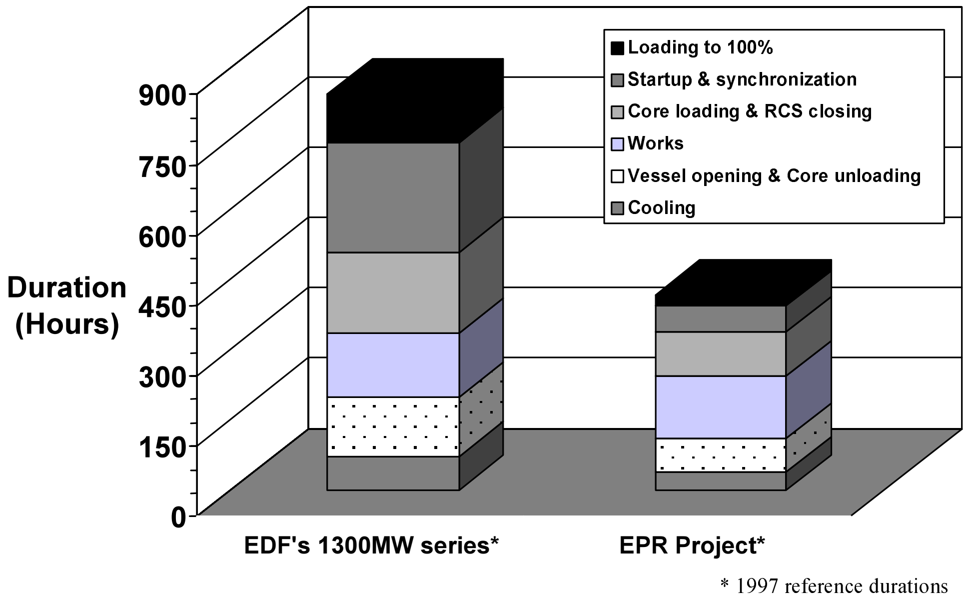

FIG.1. Refuelling and maintenance outage. Comparison EPR – EDF 1300 MW series.

2.2. Maintenance and logistic support

2.2.1. On-line maintenance

Performing on-line maintenance has significant advantages:

- to reduce the maintenance works burden during outages by spreading out the workload on the whole operating period,
- to reduce the outage duration by 2 to 3 days
- the safety systems, especially the cooling chains are longer available in outage, when they are really needed.

In any case, depending on the number of safety trains, probabilistic studies are needed to assess the Allowed Outage Times for on-line maintenance.

2.2.2. Standard exchange concept

This concept, which consists in replacing some components by serviced or new ones instead of maintaining them "in-situ" allows the operator to decrease works duration during outage, to reduce workers radiation exposure and to spread out the work load of the maintenance team over the plant operation period.

However, the impact of such as strategy on the supply, storage and management of parts has to be carefully evaluated, especially from the economic point of view.

2.2.3. Optimization of logistic support

Many utilities have already implemented a RCM approach to optimize maintenance works. It becomes more and more evident that an optimization must be also extended to the logistic support elements optimization.

This covers optimization of:

- maintenance plan
- spare parts,
- human resources and skills (sizing of teams, training …)
- facilities (warehouses, workshops, offices)
- outsourcing.
- special tools (handling or dismantling devices, data acquisition …)

2.2.4. Information management system

A computerized and integrated Information Management System makes computer resources available to all the users and contributes to the management of all activities.

It allows preparation of files from standard facts, implementation of automatic processing and management of complex tasks.

The main function of such a system are:

- maintenance works management (work orders, work permits, tagging),
- spare parts management,
- ensuring operating experience feedback (recording of events, elaboration of reliability data, failure analysis…) which can be used as the starting-point for specific studies (PRA, availability studies…) and for modification or improvement applications.
- allowing budget preparation and follow-up.

The computer system for maintenance allows optimizing organization and maintenance activity and cost control.

2.3. Outage management

On this topic, 2 main aspects must be considered:

- outage planning
- organization

2.3.1. Outage planning

Precise and detailed preparation is one of the major issues to control duration and cost of an outage: operation, maintenance and inspection and control activities have to be meticulously planned. This planning phase must consider the general operation policy defined for the plant, including fuel cycle length, grid needs, regulatory constraints, maintenance requirements. A long term planning (10 years) could be a good practice to take into account all these aspects allowing operators to optimize outage positioning and scheduling, work load, internal and external resources and design modifications or improvement.

Another important issue is to prevent outage extension as far as possible. For this purpose, the use of computerized tools allowing simulation of the outage progress and emphasis on very critical tasks which could induce extension and which have to be carefully prepared, particularly from the logistic support standpoint. EDF is currently developing such a tool, based on Monte-Carlo methodology.

2.3.2. Organization

Good practices from utilities are often difficult to adapt to others, mainly due to marked differences of size, experience, organization or culture.

However, it is possible to list the best practices which seem to be particularly efficient to control and reduce outage duration and costs, and sufficiently general to be applicable in any plant or country.

Some of them could be highlighted:

- optimization of resources (internal and external) implemented during outages, in particular the size of teams and a well adapted outsourcing contract policy,
- definition and implementation of an outstanding organization for tag-in and tag-out activities to avoid the risk of outage extension (3 shifts, large teams),
- well planned test and requalification activities,
- development of an “outage culture” involving all the hierarchy levels of the plant.

The permanent presence on the site of representatives of the Safety Authorities is an other issue, which should favorably influence safety , outage duration and cost. An additional presence of Safety Representatives within the control and test team during outage period, should allow immediate validation of these control & test without delays. This practice is currently performed in some countries such as Germany or Finland.

3. ECONOMICAL ASPECT

All the key issues highlighted above, particularly those in the design field, have a cost. Detailed studies, taking into account all the components of the kW.h cost (investment, availability, O&M costs, fuel cost) are needed to determine the most economical solution.

For a NPP under design, the best way is to choose solutions leading to short outage, provided these choices are realistic regarding the investment costs. Actually, the interest of every utility is to operate units with intrinsic high capability factor, even if, depending on the electricity market, these units are not systematically requested.

However, it appears that the shortest outage duration could lead to high O&M cost (increased outsourcing for example) and it could not be necessarily the best solution. This outage duration should be continuously adapted to the market conditions, more and more deregulated.

4. CONCLUSION

Most key issues in the design field and in the logistic support field have been studied and could be integrated into the design of any future PWR unit, as for the EPR project. Some of them could also be adapted to current plants, provided they are feasible and profitable. The organization must be tailored to each country, utility or period: it widely depends on the power production environment, particularly in a deregulation context.

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DESCRIPTION OF A PROJECT MANAGEMENT SYSTEM SOFTWARE TOOL (SUGAR)

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Abstract

Toshiba has developed a project management tool that can be applied to large-scale and complicated projects such as the outage of a nuclear power station. The project management tool (Sugar) which Toshiba developed is excellent in **operative visibility** and **extendibility**, and has been developed from the beginning for use in nuclear periodic-inspection project control. Here, the development circumstances of this project management tool (Sugar) and the feature are described, and an easy demonstration is provided as an example.

1. INTRODUCTION

The Sugar is the project management tool which also incorporates many cultural aspects of Japan to make it "friendly" and familiar to the user. It has been jointly developed with Artemis International Limited. The official name of Sugar is Artemis Sugar™. The copyright regarding Artemis Sugar™* is held by Artemis International Limited and TOSHIBA CORPORATION.

2. INTRODUCTION OF ARTEMIS Sugar™

2.1. History

The Toshiba nuclear energy division has performed examinations of project management systems since 1976. The benchmark test of a project management system was performed in 1986, and Artemis was chosen and introduced as Artemis-9000. However, since it was developed for a main-frame computer, the user interface was inadequate, and handling was difficult. The input/output operation of activity data required about 2 hours, by the time it carries out data correction and the output of a bar-chart was obtained, since it was the character base.

The Artemis Sugar™ was developed by 1994 as GUI software of the project management tool (Artemis-7000) of the Unix version. This Unix version of Sugar was used for outage schedule

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* Artemis Sugar™ is a trademark of Artemis International Limited.

examination of some nuclear plants. Then, the personal-computer version of Sugar is developed in 1998.

2.2. The purpose of development

The Sugar system was developed with the objective that many of the **man-machine interface** difficulties experienced in the past with the main-frame computer version should be improved.

Specific goals included:

* Perform input and edit in a **handwritten image**.
* A bar chart can easily be **created and output by an in –experienced user.**
* Treat **symbols** peculiar to **Japan**, such as post mark.
* Treat the activity of a time (part) unit.
* Have **PERT** function and excel in extendibility.

2.3. The function and the feature of Sugar

* **Client/Server**
  - client(W95,98/NT client)
  - server(NT):Artemis 9000/7000
* **Performance**
  - 24,000 activities
  - 255 resource profiles
  - grouping : 1600 top groups, 1600 main groups, 4000 sub -groups
* **Breakdown**
- 4 levels (Top, Main, Sub, Work)
  - calendar in each level
  - resource in 2 level(Sub, Work)
  - MS explorer style
* **Japanese Characteristics**
  - user **defined symbols** by symbol editor
  - **multiple bar on 1 line**
  - symbols indicate **holidays** , **rest days**, **time**.
  - milestones (Hotel **script-Vertical**)

GUI : Graphical User Interface

* **Map**
  - move by using mouse
  - radar function
* **Calendar (Work Pattern)**
  - multi work pattern

- minimum time unit : **1 minute**
- **calendar editor**

* **Multi windows**

- **warp between windows**

* **Query**

- various combinations
- wild card search

* **Search**

- **tree views , constrain**

* **Ease of use**

- **various views** by grouping (activities , resource…)
- **layout**
- critical , save views

* **Basic Features**

- **PERT**(time analysis : compatibility with Artemis 9000/7000)
- resource handling
- online help (Japanese)

* **Customization**

- interface files between **Artemis 9000/7000**
- **link with other windows based software**

A Sugar standard screen is shown in figure-1.

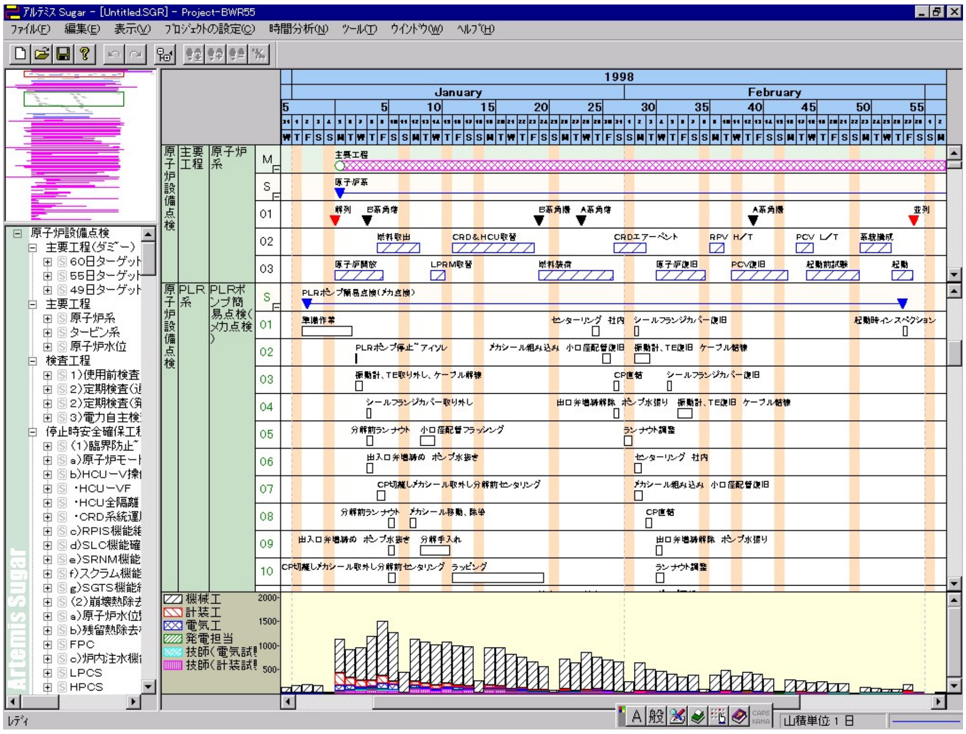

Figure-1 Sugar standard screen.

The newest personal-computer version Sugar (Ver5.0) was demonstrated at the world PC exposition in October, 1998, and the Makuhari Messe hall.

2.4. Practical use of Sugar

2.4.1. Practical use for periodic-inspection outage period examination

Simultaneously with development, the Sugar was used to examine possibilities for shortening the periodic-inspection outage for an operational plant.
Subsequently, Toshiba proposed **outage period shortening examination** consignment business by the PERT technique **using Sugar** to some electric-power companies, and received the order for conducting such examinations.

The handwriting bar chart of a Japanese unit was first begun from **changing into PERT figure** of a time unit conventionally. In the case of conversion, an actual-result bar chart needs to be **analyzed** and **work form**, net being **working hours**, and **order being related** it become important. These data were inputted into the Sugar, **time analysis** was performed, and the present **critical path** was clarified. Thereby, the effect of various processes on the critical path became clear.

A concrete examination of process shortening set up the **target schedule** (milestone) of shortening, and was performed by **repeating analysis** of critical-path work, and examining measures for shortening the schedule, one by one.

Shortening examination flow is show in figure-2.

In shortening examination, there have to some key points. Shortening examination needs to take into consideration **system operation for plant safety**. Even if it is under plant outage, in-service of cooling water etc. has the system which cannot be stopped in a nuclear power

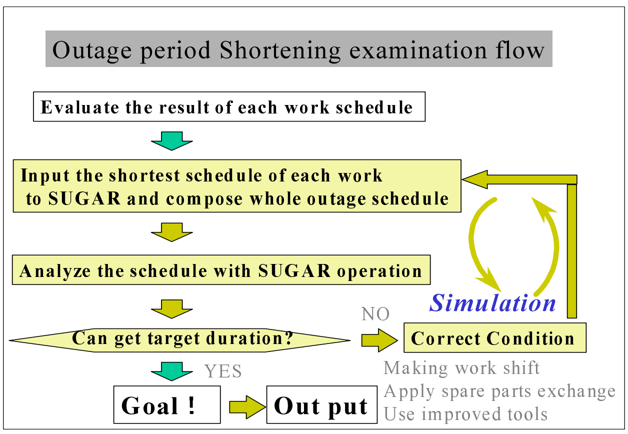

Figure-2 shortening examination flow.

station. Therefore, the grouping of the activity bar is carried out by system, and it clarifies the relation of system operation and maintenance works. The next is the **resources information** on each activity. There is the necessity that the next considers the “required resources quantity” and “resources availability” for each activity that are the resources information of a crane and worker etc.

2.4.2. Application in a power station

The client server system of Artemis Sugar™ TOSHIBA CORP is carrying out introduction proposal in a nuclear power plant. Personal computer version **Sugar (Ver5.0)** has **already been used** as the schedule and work control tool in power stations of several power companies. A typical system configuration is show in figure-3.

The characteristic from a power station system application include:

* **client server** method.
* **dispersion** input **integration.**
* preparation/allocation of a master schedule.(utility)
* input of a contract plan schedule.(contractor)
* allocation of the adjustment latest schedule. (utility & contractor)
* input of progress/achievement. (contractor)
* distribution of the newest progress schedule. (utility)

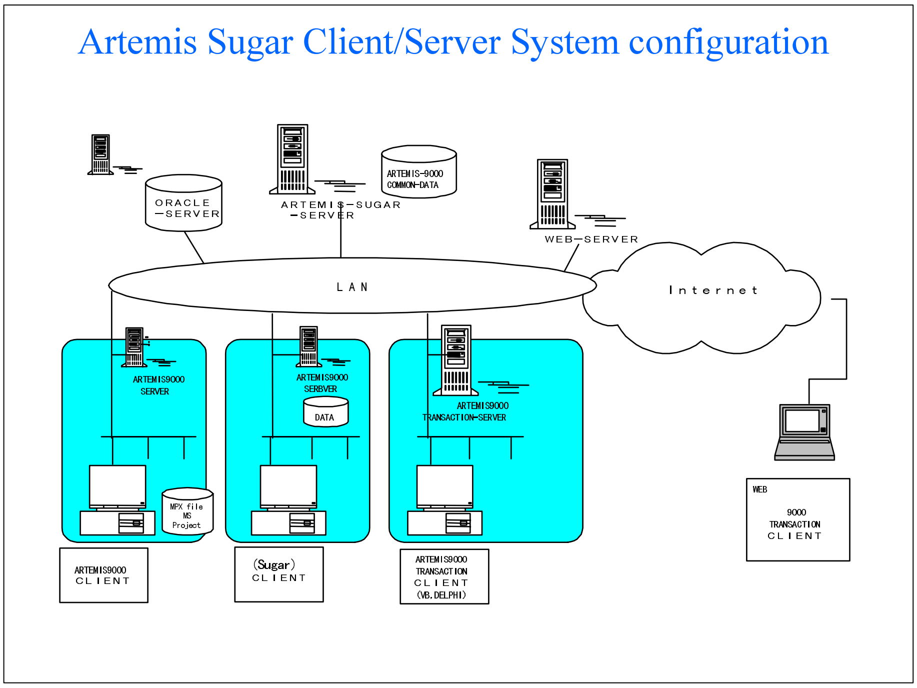

Figure-3 Typical system configuration.

2.4.3. Practical use for on-site work schedule control

A Sugar is applied at companies conducting periodic-inspection planning in Japan, and in 6 project sites within Toshiba involved in its outage management business. Work schedule control is carried out. Each site is connected with the Toshiba intranet. The actual result is that 50 periodic-inspection projects have been conducted up to the current time.

2.4.4. Practical use to a modification construction plan

Toshiba is developing operation plant business for 19 units. These plants are fixed targets. Although modification construction is carried out, Sugar is utilized in Toshiba to examine potential modifications to the construction plans (an engineering schedule, on-site process examination, resources examination) .

2.4.5. Practical use for a construction project

Application of Sugar has also been started for project management of a plant construction project.

3. DEMONSTRATION

4. FUTURE DEVELOPMENT AND APPLICATIONS

The introductory proposal of Sugar is promoted in each electric-power company, and it is also attended in Toshiba from now on. Introduction is preceded and template **data preparation business** or system introduction **consulting business** is also considered if needed. Moreover Sugar, a **work management system** (work order), and the cooperation system with a **plant risk monitor** are proposed in the electric-power company to which introduction is going, and it will be further developed into a tool for "**realization of safe and positive outage work**" in the future.

USE OF PASSIVE SYTEMS TO IMPROVE PLANT OPERATION AND MAINTENANCE

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Abstract

In a deregulated future, a utility's strength will depend on its ability to be cost competitive in the marketplace. However, the competitive advantage of nuclear power will depend on each owner's ability to reduce Operating and Maintenance (O&M) costs without sacrificing nuclear safety. The use of passive systems (i.e., systems without any moving parts) can reduce plant O&M costs while increasing safety in nuclear power plants.

1. INTRODUCTION

This paper discusses the significant O&M cost reduction and improved safety achieved at Indian Point-2 Nuclear Power Plant (IP 2) by the use of two newly installed passive systems. IP 2, located 60 Kilometers north of New York City, is the first nuclear plant in the United States to replace the conventional thermal recombiners with Passive Autocatalytic Recombiners (PARs) for post accident hydrogen mitigation in containment. Similarly, the Chemical Additive System with sodium hydroxide (NaOH) for post-accident pH control of the Emergency Core Cooling solution in containment sump was replaced with a Passive pH Control System that uses TSP baskets. Both of the original plant systems required significant maintenance and testing. The replacement passive systems are completely passive and require no maintenance. Use of these passive systems at IP 2 has resulted in a significant reduction in maintenance and testing and, hence, O&M cost.

2. HYDROGEN RECOMBINER

2.1. Function of hydrogen recombiner

U.S. Nuclear Regulatory Commission (NRC) regulations require the control of hydrogen concentration at a safe level following a loss-of-coolant accident (LOCA). Hydrogen may be generated inside the containment as a result of the zirconium fuel cladding and the reactor coolant reaction, radiolytic dissociation of water, and corrosion of metals. The function of the hydrogen recombiner is to limit containment hydrogen concentration below the flammability limit of four volume percent hydrogen so that containment integrity is not jeopardized due to a hydrogen ignition.

2.2. The conventional recombiner

Most nuclear plants use electrical or flame type thermal recombiners to reduce the hydrogen concentration in containment. These conventional recombiners use high temperature heat to recombine hydrogen with oxygen, and they depend on electric power and other support systems for their operation. Because of the complexities of these systems they are inherently unpredictable and require greater levels of maintenance and testing than the new PAR technology.

2.3. Hydrogen recombiners at IP 2

At IP 2, two conventional recombiners were located inside the containment. They were the flame types, which used hydrogen as fuel. They required an external source of hydrogen and oxygen to maintain a high temperature stable flame, which recombines the hydrogen and oxygen in the containment atmosphere. Ironically, this meant that hydrogen as well oxygen would have had to be supplied into containment to reduce hydrogen. Following an accident 15 truck loads of hydrogen and 7 truck loads of oxygen would be required to reduce the hydrogen in containment. Piping connections were provided outside the containment to hook up the hydrogen and oxygen supply from trucks. Containment penetrations were provided for the supply piping, which in turn required containment isolation valves. The hydrogen and oxygen flow and pressure must be controlled to maintain proper flame, which required control valves, instruments, and control panels to start, stop and operate the recombiners.

Fabricated in 1969, replacement parts are no longer available for this 30 year old system. Several environmentally qualified (EQ) components of the system were located in a harsh environment, had a limited qualified life, and had to be replaced regularly. There was significant preventive maintenance (PM) work for instrumentation, mechanical, and electrical components. Several components were tested on a monthly or quarterly basis to satisfy surveillance requirements. Actual recombiner flame lighting tests were performed at every refueling outage for both recombiners. During those tests the containment was evacuated for safety purposes which, resulted in a loss of approximately one day of critical time during every refueling outage. Moreover, this system had operating limitations, e.g. to maintain the flame it could not be started unless containment pressure is below 5 psig (1.358 bar). Following an accident, the system required manual start and stop by an operator stationed at the control panel located within a radiation area.

2.4. Description of PAR

The PARs used at IP 2 are designed and manufactured by NIS Ingenieurgesellschaft mbH of Germany (NIS). Each PAR consists of a stainless steel sheet metal box that is approximately 1m x 1m x 1m and which is open at the top and bottom. There are 88 catalyst cartridges, which are fabricated from perforated steel plates, inserted in each box. The cartridge frame measures 45 x 20 x 1 cm. and holds more than 400 grams of catalyst pellets. The 3 to 6 mm in diameter catalyst pellets are made from aluminum oxide spheres and are coated with palladium and hydrophobic polymers. The palladium coating acts as a catalyst and the hydrophobic coating provides water proofing. Comb-like spacers in each box secure the catalyst cartridges vertically and spaced 1 cm apart. The spaces between the cartridges serve as flow channels for the gases. Airflow enters at the bottom and the catalyst recombines hydrogen and oxygen in the flow channels to gaseous water. The exothermic reaction of the recombination produces heat, which results in a convective flow that draws more gases from the containment atmosphere into the unit from below. PARs are self-starting and self feeding. They require no electrical power or any other support system. The catalyst is not consumed as it functions, and is not subject to long term aging degradation.

2.5. PAR qualification tests

In order to qualify the PARs for use as safety-related equipment inside the containment, IP 2 completed environmental and seismic qualification tests for the NIS PARs. These are “first of a kind” tests performed for a PAR using a 10 CFR Appendix B quality assurance program.

The qualification tests were performed using a scaled down PAR model. Several cartridges were first subjected to radiation and thermal aging, seismic simulation, and exposure to post-accident temperature, pressure, humidity and chemical spray. The functional performance tests were conducted inside an environmental chamber. The chamber had instrumentation to measure hydrogen concentrations, chamber pressure and temperature at several locations. A scaled down PAR housing with exposed cartridges was installed inside the environment chamber. Hydrogen was then added into the chamber. The hydrogen concentration and other test data were monitored and recorded.

The tests demonstrated that the PAR will fulfill its intended function during and following an accident, and were crucial for NRC approval for the use of PARs at IP 2.

2.6. PARs at IP 2

Two full-size PARs are installed inside the IP-2 containment. The old hydrogen recombiner system was retired and is currently being removed. One dozen containment isolation valves, 45 control valves, hundreds of manual valves, 215 instruments and two control panels will be removed. In fact, more than 800 total components in the system are no longer required. This has eliminated numerous tests, PMs, training requirements, EQ requirements, and substantial maintenance. Several procedures including operating procedures are eliminated. PARs do not require operator interface for its operation and has freed crucial operator time during a postulated accident. . PARs do not have any active components whose failure can result in the unavailability of the system, hence it is more reliable which improves nuclear safety. The old flame type recombiners were tested as required by the operating license every refueling outage. PARs have eliminated this testing, thereby increasing the availability of IP2 by approximately one day every operating cycle.

The end result is reduced O&M costs and improved nuclear safety.

3. CHEMICAL ADDITIVE SYSTEM

3.1. Function of chemical additive system

The function of the Chemical Additive System is to control the pH of the ECCS recirculation solution in the post-accident sump following a LOCA. Maintaining the sump solution pH greater than 7.0 maximizes the iodine retention in the sump, which in turn minimizes the amount of airborne iodine in the containment atmosphere. With the inventory of air born iodine available for leakage to the environment reduced, the off-site thyroid dose following a LOCA is reduced.

The pH control also minimizes the hydrogen produced by the corrosion of galvanized and zinc based paint, and minimizes the potential for chloride-induced stress corrosion cracking of stainless steel components.

Since the initial pH of the ECCS solution, will be approximately 4.5, a chemical additive must be utilized to raise the pH of the solution in the sump.

3.2. Spray additive system

In the original design of IP 2, sodium hydroxide (NaOH) was used for pH control of the ECCS solution. The NaOH was stored in a tank and for delivering to the containment via the

Containment Spray System. This Spray Additive System included a tank, piping, valves, eductors, and instruments.

To prevent decomposition of the NaOH during long-term storage, a blanket of nitrogen gas was maintained above the NaOH solution. The tank was also equipped with level transmitters and sampling connections to assure that the proper amount of NaOH with the correct concentration was available. The tank was provided with heating elements and temperature monitors to ensure that the temperature of the tank contents was above the precipitation point. A pressure relief valve and a vacuum relief valve were provided for tank protection. A piping system with automatic valves was provided for NaOH delivery, and eductors and flow transmitters to control NaOH flow rate. In all, the Spray Additive System was very complex and required extensive maintenance and testing. Further, NaOH is also very caustic and is an occupational health hazard. It is considered a potentially toxic chemical that requires special handling precautions.

3.3. Passive pH control system

The Spray Additive System at IP 2 was replaced with a Passive pH Control System that uses TSP for pH control of the ECCS solution in the post-accident sump. TSP is stored inside the containment building in four baskets, each made of stainless steel. The top of the basket is hinged to facilitate filling and inspection, and solid to provide some drip protection. The bottom and sides are an open grating covered with fine mesh stainless steel screening to enhance wetting and dissolution of the TSP post-LOCA and yet retain the TSP granules during normal plant operation. The baskets are located on the bottom elevation of the containment. This floor gets flooded following a LOCA, such that the TSP will dissolve into the ECCS solution prior to flowing to the sump.

This change has eliminated several testing requirements of the Spray Additive System, as well as monthly sampling and testing of the sodium hydroxide. It has completely eliminated maintenance work including PM requirements on the Spray Additive System components such as valves, tank, instruments, etc. The replacement system does not require operator interface for its operation. It does not have any active components whose failure can result in the unavailability of the system, hence it is more reliable which improves nuclear safety. Moreover, it eliminates the need to handle hazardous chemicals, which improves industrial safety.

3. CONCLUSION

IP 2 is the first nuclear plant in the United States to install PARs for post accident hydrogen control. PAR qualification tests performed by IP 2 pave the way for the use of PARs in US nuclear plants. Replacing the flame type recombiners with totally passive PARs has reduced O&M costs, increased the availability of the plant, and improved nuclear safety.

Replacing the Spray Additive System with a pH control system that is totally passive has also reduced O&M costs and improved safety.

PRACTICE OF FUEL MANAGEMENT AND OUTAGE STRATEGY AT PAKS NPP

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Abstract

The Paks Nuclear Power Plant generates almost 40% of Hungarian electricity production at lowest price. In spite of this fact the reduction of operational and maintenance costs is one of the most important goal of the plant management. The proper fuel management and outage strategy can give a considerable influence for this cost reduction. The aim of loading pattern planning is to get the required cycle length with available fuel cassettes and to keep all key parameters of safety analysis under safety limits. Another important point is production at profit, where both the fuel and spent fuel cost are determining. Earlier the conditions given by our only fuel supplier restricted our possibilities, so at the beginning the fuel arrangement changing was the only way to improve efficiency of fuel using. As first step we introduced the low leakage core design. The next step was the 4 years cycle using of some cassettes. By this way nearly half of 3 years cycle old cassettes remained in the core for fourth cycle. In the immediate future we want to use profiled cassettes developed by Russian supplier. Simultaneously we will load new type of WWER cassettes with burnable poison developed by BNFL Company. Hereby we can apply more BNFL cassettes for four years cycle even more. Both cost of fuel and number of spent fuel can be reduced besides keeping parameters under safety limits. The Hungarian in service inspection rules determine that every four year we have to make a complete inspection of reactor vessel. Therefore earlier we had two types of outages. Every 4 years we planned a long outage with 55-65 days duration and normal ones with about 30-35 days duration between the long ones. During the normal outages this way did not give us enough room to utilise the shortest possible critical path determined by works on reactor. Some years ago we changed our outage strategy. Now we plan every 4 years a long outage, and between them one normal and two short ones. As a result the overall outage duration can be reduced by 10-15 days every year.

1. INTRODUCTION

The Paks Nuclear Power Plant is the only nuclear power plant in Hungary. It has 4 units with capacity of 460 MW each. The original contract to build nuclear power plant in Hungary was signed by Hungarian and Soviet Union Government in the second half of sixties. The contract was about building of two units, type WWER-440 model 130. Some years later the contract was modified for building of 4 units, type WWER-440 model 113. The construction works started in the middle of seventies and the commissioning of the units was in 1983, 1984, 1986 and 1987 respectively. The original capacity of the units was increased by 20 MW on each unit in consequence of different modifications in secondary site that resulted in higher efficiency of turbines.

The operating organisation was established on 1 January 1976 and it became a shared company on 1 December 1991. Our shareholders are the Hungarian Power Companies Ltd (99,9 %) and local governments (0,1 %). Since the shareholder of the Hungarian Power Companies Ltd is the Hungarian Government therefore the owner of the Paks Nuclear Power Plant is the Hungarian Government as well.

During the last 10 years the average electric power generation of the plant was 13928 GWh. In different years this value was changing very little - less then ! 2 % - which gives an evidence of the stable operation of the units. During the same period the plant average load factor (Fig.1) was changing between 85 - 87,7 % and the lifetime load factor of the plant reached 85,2 % by the end of 1998. According to these high values of the load factor our units are regularly included in the list of the Top Twenty-five units. As stated at 31 December 1998 our units were on the following places:

– 9. Paks 4 - 86,5 %
– 12. Paks 3 - 85,6 %
– 13. Paks 2 - 85,6 %
– 21. Paks 1 - 84,1 %

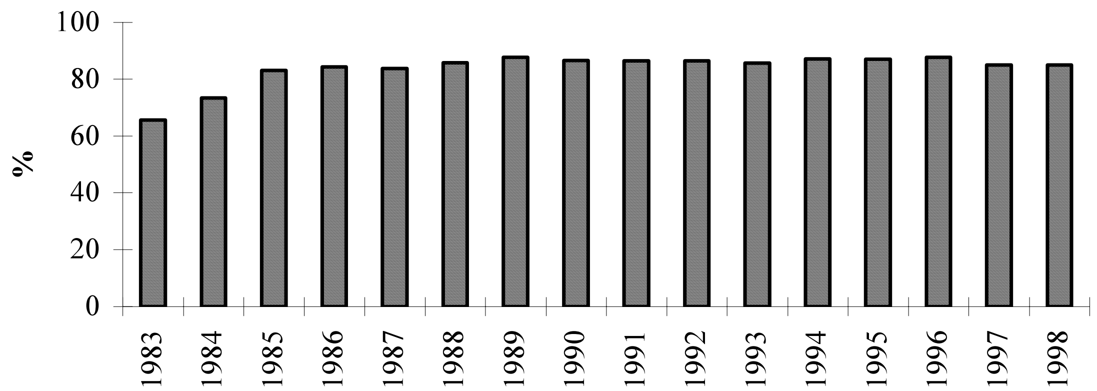

FIG. 1. The average load factor.

In Hungary the total amount of built-in capacity for electric power generation is 7850 MW (1998). The capacity of the NPP is 1840 MW which gives 23,4 %. Since we operate the NPP in basic load the electric power generation of the plant gives 38 % of the total Hungarian generation (Fig. 2.). During last years the price of the nuclear power generation was changing and slightly increasing and in 1998 the price reached 4,72 Ft/kWh (0,022 USD/kWh) (Fig. 3.). The price of the power generation from the other sources was in 1998: 7,26 Ft/kWh (0,034 USD/kWh) for coal-hydrogen and 10,34 Ft/kWh (0,045 USD/kWh) for coal. The comparison of these prices gives that the nuclear power generation has the lowest price in Hungary.

In spite of the above mentioned favourable facts besides meeting the regulatory (and of European Union) safety requirements the reduction of operational and maintenance costs was and remains the one of the most important goal of the plant management. The fuel management and the outage strategy can give a considerable influence for this cost reduction. In this presentation we would like to outline the optimisation process of the fuel loading including further development and the way of changing the outage strategy.

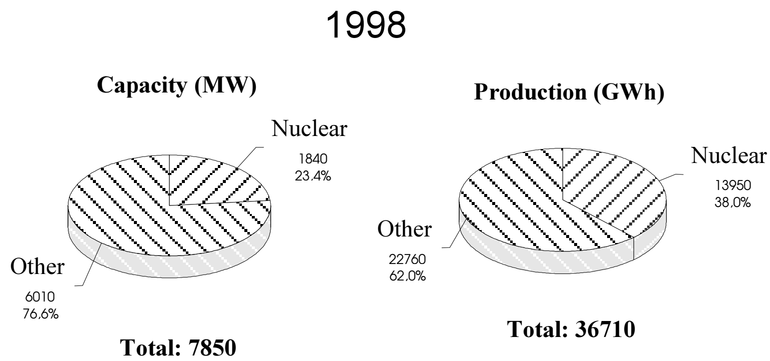

FIG. 2. Portion of NPP in Hungarian electric capacity and power generation.

2. PRACTICE OF FUEL MANAGEMENT

2.1. Nuclear fuel

VVER-440 type rectors have special nuclear fuel called cassettes. The core consists of 349 fuel cassettes. A hexagonal shroud surrounding the assemblies with pins forms cassettes. The shroud separates the cassettes, the water gap between them results higher peak in pinwise power distribution. The control and safety protection system consists of boron-steel rods. A fuel part called follower is attached to it. It has a special feature as by pulling out the boron steel absorbing rod from the core, the follower as fuel is pulled in at the same time.

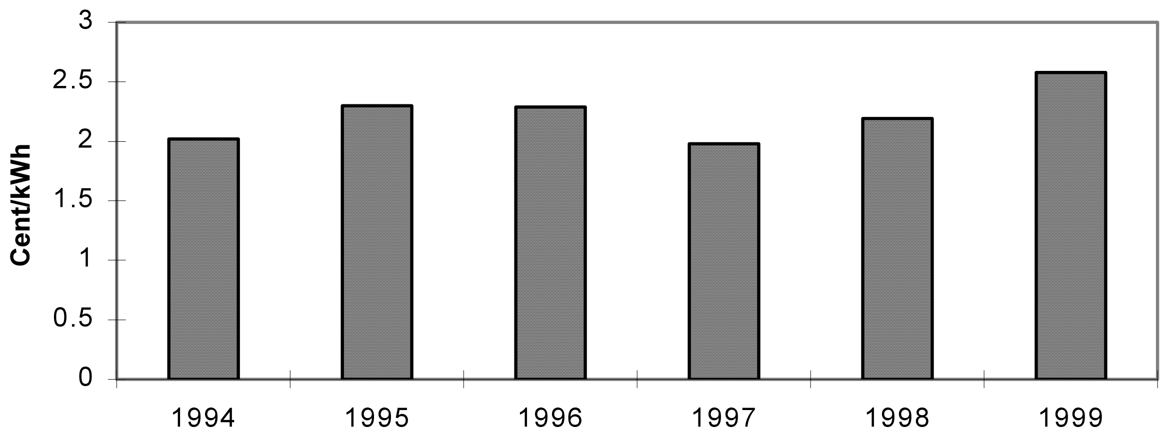

FIG. 3. Price of nuclear power generation in Hungary.

2.2. Requirements for load design

For designing the load, the most important conditions are the safety and operational procedures that are to be met.

The most important requirements are the restrictions for **safety** operation, like negative moderator temperature reactivity coefficient, both pins and assemblies power distribution, pinwise linear heat rate, as well as the heat-up of the coolant in cassettes. The differential valuability of the control group should not be grow too high since it is not allowed to take in a reactivity above the allowed speed. The shut down margin could not be worse than -2 %, even if the most valuable absorption rod is fails and remains in upper position. The allowed maximum burn-up determines the lifetime of cassettes, utilisation of which is very important from point of view of spent fuel number.

Having the safety requirements met, we have to comply with the requirements of **availability** specified by the plant operation. It means that date of the outage, the calendar day of its beginning and completion is fixed by the thorough influence of the electrical energy system. That in turn determinates in advance the beginning and length of the operational cycles, to which the initial reactivity reserve of the reactor load by all means shall be adjusted. In design work we have to take into account such limitations that only one unit could be in outage, the outages should not be conducted in the winter period, at least one week pause should be left between the outages, and there is a determined sequence in the maintenance.

Further on, it is a natural claim of **economy** that the individual cycles should be performed with the possible lowest fuel consumption and specific fuel cost. This last condition could be taken into account by an optimisation process enhancing the economy.

Accounting with the availability requirements of the units implies some uncertainties. As the milestones of the outages are bound to calendar days, but for designing we have to consider the real effective operational time. The context between the calendar and effective operational time is drawn into the calculations by a utilisation factor of 0,97. If the real value is deviating from it, then as a perturbation affecting the load design. The same effect could be caused by modification of the original date of the outage. All these deviations are affecting the economy through the specific fuel consumption. Unfortunately as the result of all these circumstances, in spite of the optimisation, the best fuel utilisation could not be realised in all cycles.

2.3. Optimisation of fuel loading

Both the cost of fresh fuel and final disposal of spent fuel constitute the considerable amount of cost, so improvement of fuel management is important question. The safety operation requires a lot of effort generally. We had to work out optimisation process to find the most economic loading at required safety operation. More possibilities considering boundary limits give more tools us to find more economical fuel management as optimal one. Perturbations of real life can disturb this optimal loading of course. After a cycle, when we analyse it, we have to take this into consideration.

The aim of optimisation is to find the cycle series cost of which and the number of spent fuel are the minimum during the reactor operation. It is a complicated attached problem, because the fuel is used for three or four cycles in the core. There are several approaching solutions. We have made up a simplified model to solve this task in reasonable computer time. We separate the problem introducing equilibrium cycle. In this case we assume, that both loading pattern and length of cycles are same cycle by cycle. By this way the most economical equilibrium cycle has the maximum reactivity, i.e. the maximum length of cycle. The global optimum until recently has not been solved.

In the practice, due to the perturbations of real operation, equilibrium cycle is not realised. On the other hand, these perturbations are small, so we can rule out the extreme non-equilibrium cycles too. Nevertheless we can assume that the operation of our reactor can be described with equilibrium cycles. Our optimisation method is aimed at finding the maximum reactivity arrangement for the given length of cycle. This loading pattern planning method has given us good results for years. Furthermore, it is flexible enough to follow the changing operational conditions.

2.3.1. Original loading pattern

Equipment of NPP was delivered from Russia before 1982 as well as the technological documentation. There was only one starting loading pattern in this documentation. By this way, we could perform only fixed length of cycles. Our four units started one after another nearly yearly. It was necessary to meet the inevitable requirement of planning with different length of cycles to co-ordinate the dates and length of their reloading. Because of this we changed the original loading pattern plan in unit 3 and unit 4 and later but not basically. We reached the required length of cycle by variation of number of 3.6 % and 2.4 % uranium enriched cassettes. The followers were produced only with 2.4 % uranium enrichment. Another unfavourable condition was the 3 years (3 cycles) lifetime of the cassettes. Arrangement of loading was similar as original. The high enriched and valuable cassettes were on the periphery of the zone while high burned up and less valuable cassettes were mixed in the central part of zone.

2.3.2. Low leak arrangement

Relayed on information scientific literature and experience of our colleagues in abroad, we examined our possibilities to improve our fuel management. Our possibilities were limited. Cassettes with enrichment 1.6, 2.4 and 3.6 Uranium % were obtained from our only supplier, and followers only with 2.4 Uranium %. Their lifetime was 3 cycles. So we had only on way to optimise our loading: to change its arrangement.

It was obvious from neutron physical calculation in case of equilibrium cycles, that moving some fresh cassettes from periphery to inside region and putting the most valueless cassettes to periphery can increase efficiency of fuel using. It was necessary to improve our computer codes for this low-leakage fuel arrangement calculation.

The low leakage fuel arrangement was introduced from 1989, first in unit 1 cycle 7. As a result the average enrichment necessary for same length of cycle was decreased comparing to the original fuel arrangement. Due to this, the fuel cost reduced by 3 %, although the number of spent fuel remained. Additionally this fuel arrangement reduced the neutron influence for reactor vessel.

2.3.3. New loading strategy from 1993

Two favourable improvements were applied in fuel product technology of Russian supplier from beginning of 1990. We could buy 3.6 % uranium enriched followers, and we started to use them from 1993. By this way the given length of cycle could be reached by lower number of fresh fuel, then earlier. The other improvement made possible to use fuel cassettes for four cycles. So we introduced the four-cycle fuel refuelling strategy from 1995. The required average reactivity was reached in the cycle even we placed more low-value cassettes in periphery. Application of fewer fresh fuel increased degree of freedom of loading pattern planning, so the parameters did not exceed the limits for maximum local power at reduced neutron escaping and higher average power of valuable cassettes. The number of spent fuel reduced as well.

In our new practice of loading pattern planning, we always use 3.6 % uranium enriched cassettes. The length of cycle was set by number of fresh fuel. Nearly half of the 3-cycle fuels remain for the fourth cycle. This reduced the fuel cost with 16 %. The cassettes with highest burn-up were taken out from zone after 3 cycles, so the maximum permitted average burn-up was not reached and the reliability of cassettes was retained. The first arrangement was introduced on unit 1. Because we did not use all the 3-cycle cassettes in the fourth cycle, a better name for this fuel arrangement is "**3.5-cycle burn-up practice**".

2.3.4. Further development

Our plan is to apply more cassettes for 4-cycle even more. This is possible by increasing the enrichment of fuel cassettes. We have two ways for this now.

The Russian supplier has developed a cassette with different enrichment in radial direction, a **profiled cassette** for WWER reactors. The average enrichment is 3.8 uranium % for these cassettes. We prepare our system for this task simultaneously, performing required safety analysis. We have started permission process. We want to introduce these types of fuels first in unit 3 cycle 15 in 2000 and in unit 1 cycle 19 one year later.

An alternate solution is another supplier and cassettes developed by BNFL company. Analysis was accomplished to develop a new WWER-type cassette together with Finnish IVO Company and NPP Paks. First a WWER fuel geometry was chosen considered optimal for NPP Paks too. The selecting process based on computed features of equilibrium cycles. These features were the loading included 349 cassettes, 1375 MW nominal power, about 320 effective days cycle length a year, yearly 90 unloaded cassettes consisting of 12 followers and 78 fixed cassettes. The fixed fuel cassettes would be in core for four cycles while followers for three ones. These conditions determined the necessary enrichment of **BNFL** cassettes. To provide necessary subcriticality in spent fuel pool cassettes with high uranium enrichment and burnable poison were selected. The less enriched followers to fixed fuel assemblies give us two advantages. Followers are used only for 3 cycles so their less enrichment reduce the fuel cost by 1%. Enrichment could be chosen according to requirement for subcriticality of spent fuel pool so burnable poison was not necessary in the followers as the number of them is less to fixed cassettes. This step reduced the extra cost connected with burnable poison by 13 %. Enrichment in fixed cassettes should be increased to reach the desired cycle length but this is relatively not too much because of their higher number. We plan to load these fuel assemblies first in unit 4 cycle 16 in 2002.

2.4. Evaluation of the different load types

We can estimate the profit of different loading on basic of two economic considerations. Both the cost of fuel and the number of spent fuel are important. However the boundary conditions considered in the load design do not enable us to reduce both of the cost elements simultaneously. In the case of the Paks NPP, applying the LLP array did not reduce the production of the spent fuel, but by utilisation less amount of the 3,6% enriched fuel reduced the fuel consumption, while the same amount of assemblies were unloaded from the reactors.

By implementing the 3,5 years fuel cycle, above the cost reduction the production of spent assemblies is also decreased.

As an example, here we introduce the characteristic data of the loads applied in the Unit 1 of the Paks NPP. Fig. 4 shows the average burn-up attained and planned in the 3-21 operational cycle. The number of unloaded assemblies is shown on Fig. 5 for the same period. One can see that while there is no significant change in the number of spent fuel assemblies the level of the average burn-up is diminishing which is because of increase in the number of lower enrichment. Fig. 6 introduces a cost of the individual cycles formed with a fictitious price. There the compulsion determined by the Plant operation causes a high variation in the fuel cost of one kWh, and the optimal solution could not be applied in each case.

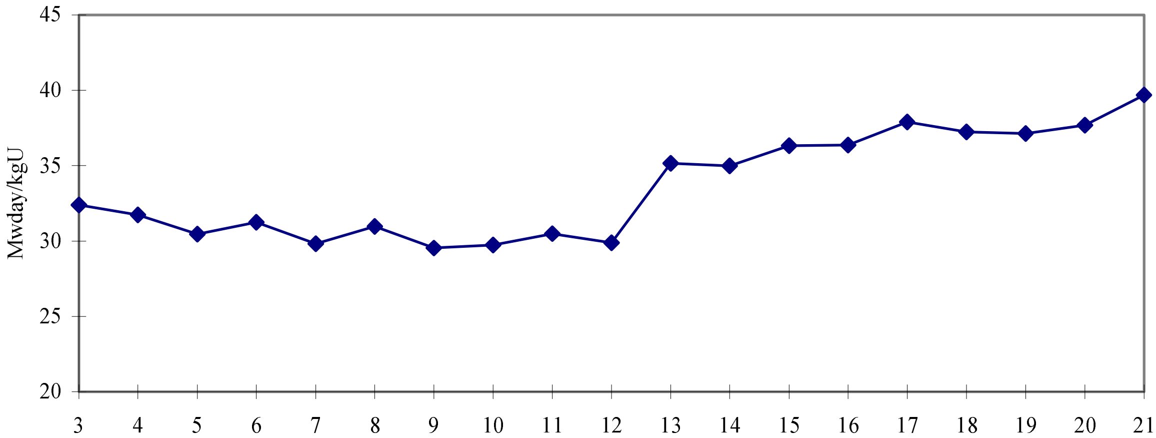

FIG. 4. Average burn-up of unloaded cassettes on Unit 1.

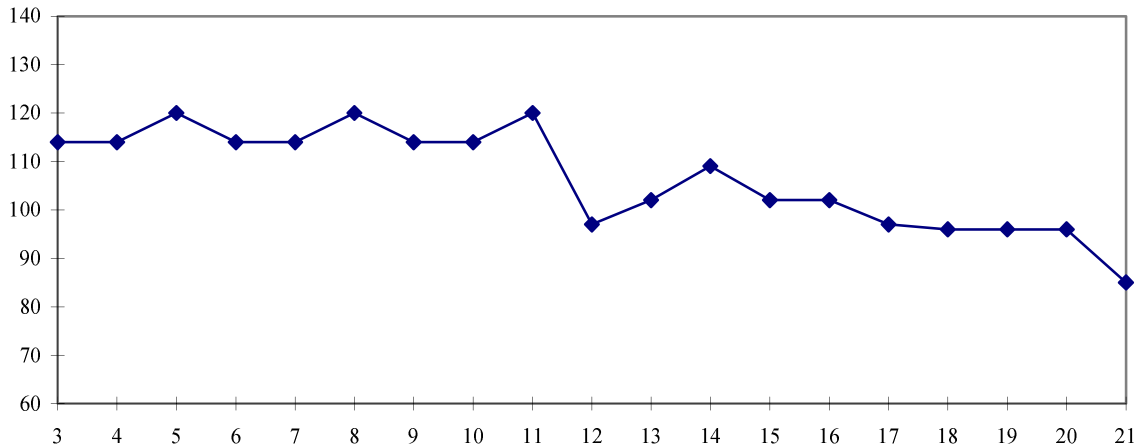

FIG. 5. Number of unloaded cassettes on Unit 1.

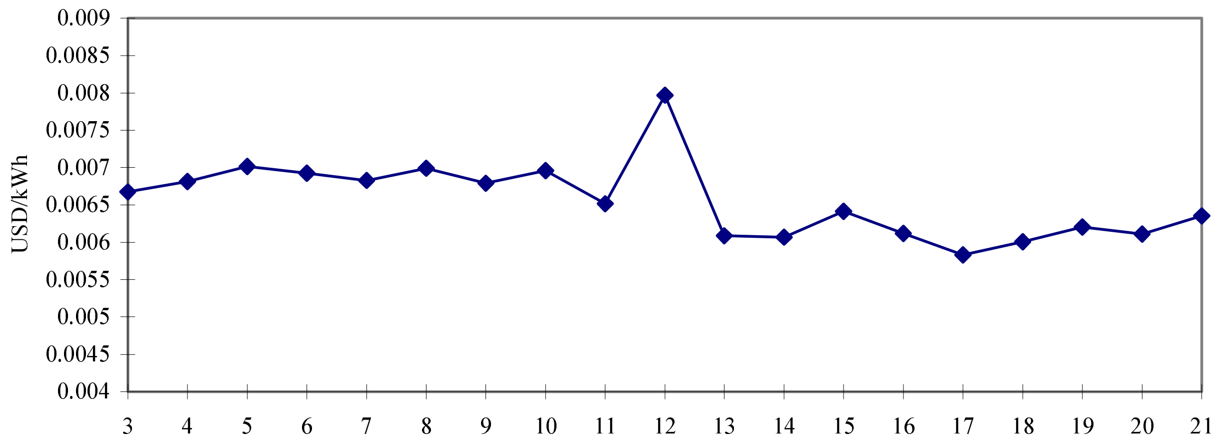

FIG. 6. Specific fuel cost on Unit 1.

To estimate the theoretical fuel consumption for the applications of the individual load types - without the perturbations caused by the real operation - we evaluated the equilibrium cycles. Fig. 7 shows the average and maximum burn-up that could be attained in the conventional, LLP, in the 3,5 year - and in the aimed 4 years cycles. Fig. 8 shows the assemblies to be annually unloaded and Fig. 9 the specific fuel consumption with the fictitious price. It is evident that both in the cost and in the number of spent fuel assembly a considerable reduction could be attained by the 4 years utilisation of the assemblies.

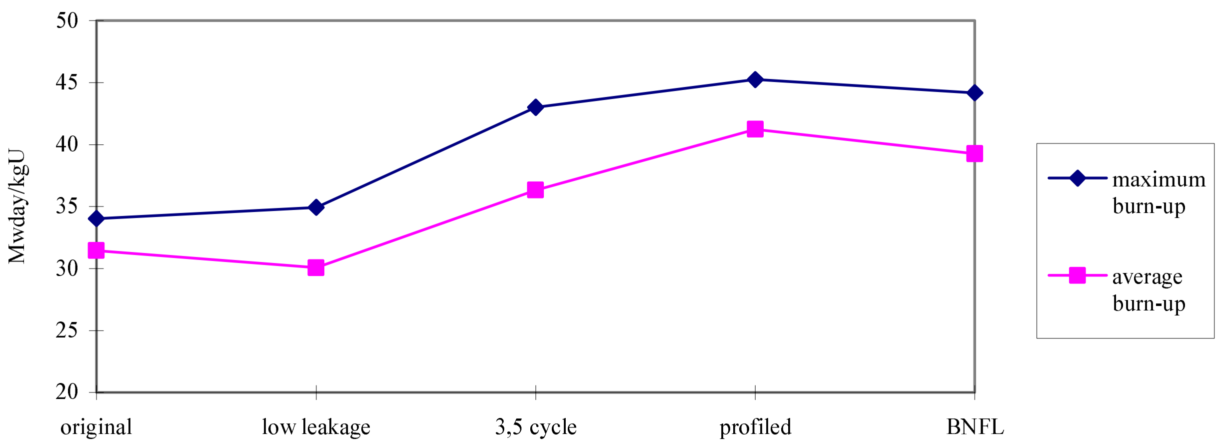

FIG. 7. Burn-up of unloaded cassettes.

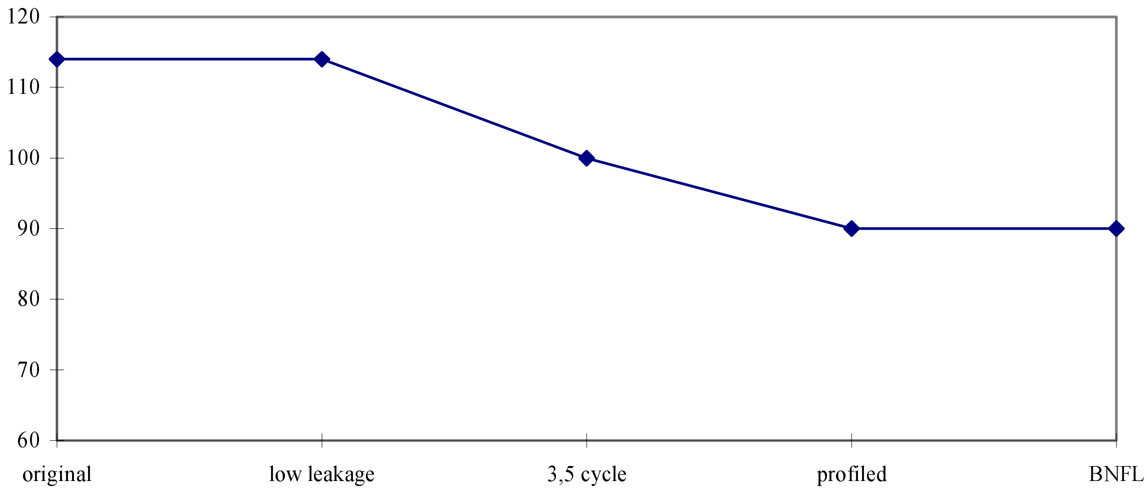

FIG. 8. Number of unloaded cassettes.

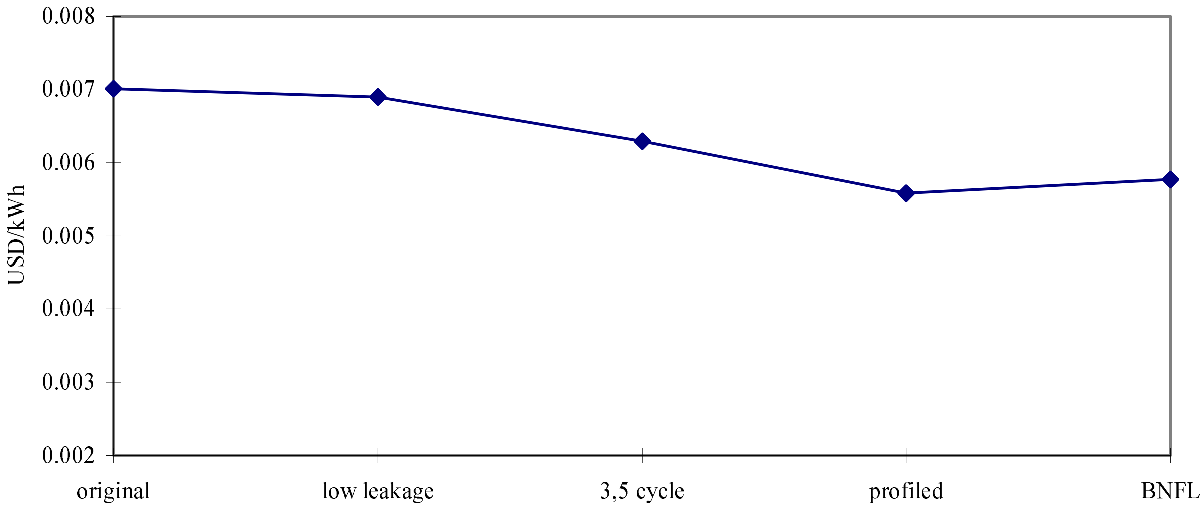

FIG. 9. Specific fuel cost.

3. OUTAGE STRATEGY

As it was mentioned above we operate the NPP in basic load. Practically it means that we can run the units at rated power almost all the time. Therefore we are interested in reducing the outage duration. The total outage duration is shown in Fig. 10. In 1996 we introduced a new outage strategy which has yet promising results and in the future can be resulted in reduction of the overall outage duration by 10-15 days.

3.1. Old outage strategy

The Hungarian in service rules determine that every four year we have to make a complete in service inspection of reactor vessel. Therefore earlier we had two types of outages. Every 4 years we planned a long outage with 55-65 day duration. The critical path of the long outages was on the reactor works and included the full unloading of all the fuel cassettes, the removal of reactor internals, the full scope in service inspection of the reactor vessel, putting back the reactor internals and the load back the fuel cassettes one third of which has been replaced by new ones. Since the above outlined critical path was quite long during the long outages it was possible to perform a wide range of different works like in service inspection of tanks and pipelines, modifications and reconstruction of all size.

Between the long outages we planned three normal outages with about 40-45 days duration at beginning and about 30-35 days duration later on. The critical path of the normal outages was on the reactor works or on the works of safety systems or sometimes on other works. In fact the problem was that there were no clear rules for determination of duration of normal outages and there was no practical limitation for the work volume. The duration we usually planned the same time as the minimum of the actual duration of normal outages during previous years. In the work volume all type of works was included except the inspection of reactor vessel and reconstruction of major size.

The way of planning of long outages was considered good but the planning practice of normal outages was not enough sufficient. Sometimes such works that could have been postponed to the long outage of the next year determined the duration. Another time the works on the original critical path (reactor works) were performed during a shorter time, but this earlier finish could not been realised for the entire outage.

3.2. New outage strategy

The solution of the above mentioned problem was the introduction of three types of outages instead of two ones with type dependent limitation of works. The long outages remained practically the same as were earlier. All the work of major size has to be planned for the long outages. The critical path can be on the reactor works or on the other works that are usually reconstruction or modifications of major size. The plant goal for the duration of long outages is 55-60 days.

Between two long outages we have short - normal - short outages. The short ones are based on the minimum possible critical path of the reactor. It is usually not allowed to plan for this type of outages such works which require longer time then the critical path has. It means that for the short outages we do not plan in service inspection, reconstruction and modification work of medium and major size, only minor ones. The plant goal for the duration of short outages is 25-30 days.

In principle it would be possible to have only short outages between the long ones. But in this case we might have the following situation. Some modifications, reconstruction and in service inspections of medium size can not be included into the short outages therefore they have to be postponed by 3 years in worst case.

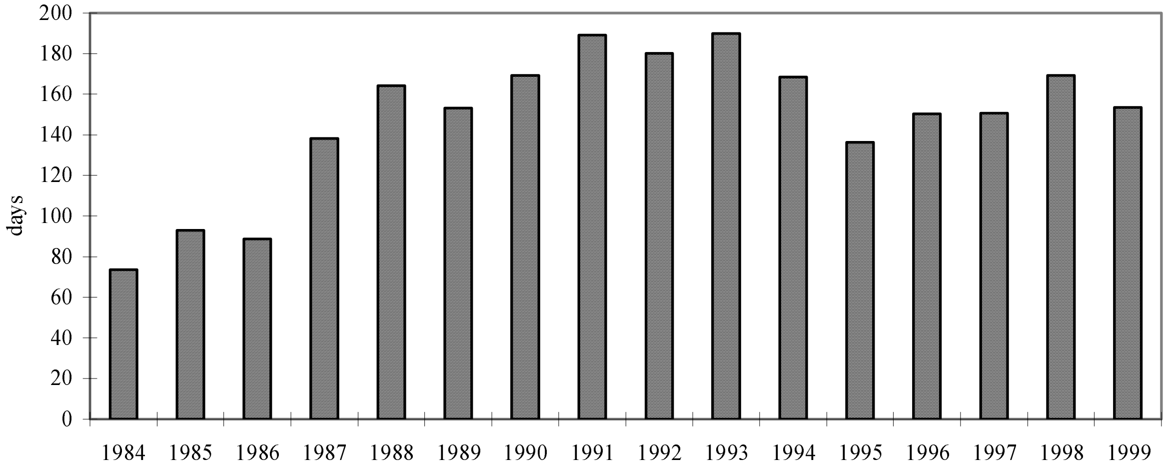

FIG. 10. Total outage duration.

This is usually not allowed because of regulatory requirements or from point of view of the balanced budget. The other problem is that the work volume of the long outages will be to big which resulted in too long outage duration. Therefore we decided to have a normal outage between two short ones. For the normal outages we plan modifications, in service inspection works and reconstruction of minor and medium size. The plant goal for the duration of normal outages is 30-35 days. The long-term outage plan is shown on the Table I.

TABLE I. THE LONG-TERM OUTAGE PLAN

	2000	2001	2002	2003	2004
Unit 1	short	normal	short	long	short
Unit 2	long	short	normal	short	long
Unit 3	short	long	short	normal	short
Unit 4	normal	short	long	short	normal

4. CONCLUSIONS

In this paper there wer e introduced the practice of the fuel management and the history of changing of the outage strategy. The optimisation of the fuel load pattern has given a considerable decrease of fuel costs. In the future the implementation of the BNFL cassettes can give a further cost reduction because of competition of two fuel supplier. The introduction of the new outage strategy has resulted in decrease of the total outage duration by 5-10 days during last years. By the end of 2002 we will have finished our safety-upgrading program which has a big contribution to make longer outages then the plant goal. Beginning from 2003 we consider as real goal to reach 140 days of total outage duration.

Now in Hungary a new Low of Electric Energy is being prepared. It is planned that from the 1 January 2001 the market of electricity trading will be opened. It means that the authorised consumers can buy the electric energy even from abroad. (Presently in Hungary the only electric energy trader is Hungarian Power Companies Ltd). Therefore in the future the Paks NPP have to be competitive not only with Hungarian conventional power plants. This is the main reason that in the future even more attention should be given to reducing operational and maintenance costs at Paks NPP.

ADVANCES IN DESIGN AND TECHNOLOGIES FOR IMPROVING PLANT OPERATION AND MAINTENANCE

(Session IV)

Chairperson

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ADVANCES IN NEW WWER DESIGNS TO IMPROVE OPERATION AND MAINTENANCE

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Abstract

Economic operational indices of WWER-type reactors show their competitiveness in all the countries where these reactors operate. Advanced WWERs being designed and constructed now have the improved characteristics of economical efficiency and are more convenient for operation and maintenance. Many technical solutions aimed at improvement of the operational performance are implemented in the design of WWER-1000/V-392 and WWER-640/V-407, and these reactors are the important basis for the nuclear power expansion in Russia. Some of these solutions are considered in the present paper.

1. INTRODUCTION

Nowadays in eight European countries 48 commercial reactors of WWER-type are in operation with total power exceeding 32 GW, including 20 reactors WWER-1000 and 28 reactors WWER-440. More than 700 reactor-years of operation without serious incidents with radioactivity release outside NPP site have demonstrated high level of WWER type reactors safety. Economic operational indices of such reactors in Russia point out the competitiveness of WWER reactors as electric power producers in many regions of the country.

Nevertheless, operation experience, new national and international safety standards and changes in economy of Russia govern the necessity of development of new advanced reactors. New reactors of WWER type shall possess the improved characteristics of safety and economical efficiency, be more convenient for operation and maintenance. In Russia a complex of R&D work is being performed on development of power units of new generation with WWER-1000/V-392 and WWER-640/V-407 reactor plants meeting new requirements for safety and economical efficiency of electric power production.

2. REACTOR PLANT WWER-1000/V-392

The advanced design of WWER-1000/392 has been developed on the basis of standard reactor plant of V-320 design, which has been in operation for a long time at nuclear power stations in Russia, Ukraine and Bulgaria. New power units with WWER-1000/320 (in Czechia, Russia, Ukraine) are being constructed or planned. So, the design of WWER-1000/392 is based on the approved technical solutions and is an evolutionary development of the operating reactor plant with the following main characteristics:

– power - 3000 MW(th)
– primary/secondary pressure - 15,7/6,3 MPa
– coolant temperature at the reactor outlet - 320°C
– average burnup in equilibrium cycle - 40 MW.day/kg U
– effective time of operation at nominal power - 7000 hours.

Many technical solutions aimed at improvement of safety and operational performance of the plant are implemented in the design, in particular:(1) advanced reactor WWER-1000; (2) passive system of residual power removal; (3) passive system of the core flooding under loss-of-coolant accidents; (4) passive system of rapid boron injection for the reactor shutdown; (5) primary coolant pump preventing coolant leak under long-term station blackout.

2.1. Reactor vessel

The length of reactor vessel V-392 is increased in comparison with reactor V-320 at the expense of larger length of the supporting shell, keeping the possibility to transport the vessel by railway. With this, the core top elevation is decreased in relation to the elevation of the reactor supporting structure that allows to reduce considerably the personnel dose commitment in maintenance of reactor, steam generator and electric drive of the reactor coolant pump. Thus, neutron flux in the region of support is reduced almost two times in direct passing from the core through the vessel, and from the streaming out of the gap between the reactor vessel and «dry» shielding - more than 10 times.

The vessel extension allows also to reduce the neutron irradiation intensity of critical weld between the supporting shell and the shell of nozzles zone. Owing to this, the margin is increased for the vessel integrity under pressurized thermal shock. The vessel extension allowed to increase the coolant inventory between the core top and the lower generant of the inlet nozzle, that is, to improve the core cooling conditions under loss-of-coolant accidents.

Containers with surveillance specimens in reactor V-392 are placed on the vessel inner wall, whereas in reactor V-320 the surveillance specimens are placed on the upper end of the core baffle. Such upgrading brings together the conditions of neutron flux effect on the surveillance specimens and on the vessel metal allowing to predict more exactly the variation of the vessel mechanical properties in the course of operation.

2.2. Internals

In reactor V-392 the stops are installed on the core barrel bottom, and arms are placed on the upper end of the core baffle (these components are not provided in the operating reactors V-320). The stops are installed with a small clearance in relation to the core barrel bottom, therefore in case of hypothetical guillotine break of the core barrel a part, broken away, will move only slightly and the arms will keep the core baffle in engagement with the PTU lower plate under such downward motion of the core barrel.

Elastic clamping element is transferred from the core barrel flange end to the supporting shoulder of the protective tube unit. This upgrading simplifies maintenance of the clamping element.

On the cylindrical part of the core barrel in the zone of flow separator the compensating plates are placed with the help of which the design value of mounting clearance is achieved. In heating-up the reactor vessel and internals this clearance is decreased and the core barrel is clamped to the flow separator over the whole perimeter that reduces the vibration loads on the core barrel. Design of the core baffle channels is changed to smooth temperature fields in the baffle and to decrease the resulted deformations of the baffle.

Distance between the middle and upper plates of the protective tube unit is increased. This allows to increase the bending radius of the guiding channels where the in-core instrumentation elements are arranged. Owing to this upgrading all channels are brought into periphery nozzles of the upper unit that improves the reliability of the in-core measurement system and simplifies its maintenance.

2.3. Reactor upper unit

On the upper unit of V-392 reactor 121 nozzles are provided for the members of reactivity control system and reactor emergency protection (CPS) in comparison with 61 nozzles in V-320 reactor. This gives a possibility to vary the number and arrangement of CPS members and to optimize each fuel cycle for reaching the best characteristics of the core safety and efficiency.

In V-392 reactor the measurements of coolant temperature and core power are combined and brought through the common nozzles of in-core instrumentation (ICI), while in V-320 reactor there are separate nozzles for temperature monitoring and nozzles for core power monitoring. With this, all ICI nozzles are arranged on the periphery of the reactor upper head that facilitates the access to them when reactor assembling or removing the upper head unit, and reduces the repair personnel dose commitment.

In V-392 reactor the upgraded control rod drive is used with the improved maintainability and more simple procedure of the drive mounting-dismounting. The drive service life (including electrical part) is 30 years with the outlook of its further extension to the reactor service life. Position indicator, used in the drive, provides for monitoring the position of control rod in the core in each 20 mm (instead of 350 mm in the existing analogs). Monitoring of drop time and position of control rods in the core is also provided under the reactor scram, that is, the functions of diagnostics are also fulfilled.

2.4. Main coolant pipeline

Design of reactor plant V-392 is developed with application of «leak-before-break» concept that allows to give up the massive supports-restraints on the main coolant pipelines. Owing to this, all sections of pipelines become accessible for in-service inspection that improves their reliability. The personnel dose commitment during the inspection of pipelines bending is also reduced because the labour consuming procedures on removal of the upper parts of the emergency supports are excluded.

2.5. Other components of the reactor plant

Many other components of reactor plant V-320 are upgraded in design of V-392 with the aim to enhance the station safety and efficiency and improve the conditions for operation and maintenance. In particular, for reactor plant V-392 the reactor coolant pump GZN-1391 is used. This pump is the upgraded GZN-195M, which is used for the operating reactor plants V-320. In GZN-1391 pump water is applied as the lubricant and cooler of the main bearing; in combination with the improvements in the system of motor lubrication this allows to give up the outside oil system and exclude possible fire.

In reactor plant V-392 it is supposed to use some passive systems intended for fulfilment of the main safety functions (reactor shutdown, decay heat removal, core cooling). These systems, in the first turn, are intended to improve considerably the plant safety. Alongside with this, passive systems, as a rule, are simpler in operation and maintenance, and therefore improve also the plant economical characteristics.

3. REACTOR PLANT WWER-640/V-407

In the primary circuit of reactor plant V-407 the same equipment is used mainly as in reactor plant V-392, though there is a number of important differences. In particular, in comparison with V-392 in the reactor plant V-407:

– the core specific power intensity is decreased;
– neutron fluence to the reactor vessel is reduced;
– axial loads on fuel assemblies are decreased;
– no loop seals in the primary loops;
– passive safety systems are widely used.

3.1. Improvement of operational availability

High operational availability of the reactor plant is assured owing to:

– application of the equipment and technical solutions proven by operation;
– using of the approved materials and manufacturing processes for the main equipment;
– optimal water chemistry of the primary and secondary circuits;
– reducing the number of process systems and simplification of their schemes.

The design provides for on-line monitoring the state of equipment and components with the help of special diagnostic systems (noise diagnostics, monitoring the equipment vibration, detection of loose and poor fixed objects, primary-to-secondary leak monitoring). Special systems are provided for information support of the operation (safety parameters display system, equipment residual life assessment system, operator support system).

3.2. Maintenance improvement

To improve the conditions of maintenance of the reactor plant systems and equipment the appropriate experience is taken into account as well as the practices applied by Western specialists, including the results of international assessment of technical solutions accepted in the design of V-407. Many components of the reactor plant were designed in such a way that their service life will be equal to the station service life, as a whole.

Reactor plant designing was carried out with regard for the requirements related to inspection, maintenance and repair of equipment. In particular, good access is provided to the equipment requiring periodical examination and repair, the non-destructive inspection systems included. For the primary equipment the materials are used mainly with low content of cobalt and other elements with long half-life period. Together with designing of systems and equipment the process specifications for their maintenance and repair were developed.

3.3. Consistency of technical solutions

Design of reactor plant V-407 makes maximum use of technical solutions proven by operation experience of existing power units and justified in the designs of advanced WWER-1000. Such consistency improves technical characteristics of the reactor plant including also operational availability and maintenance.

4. CORE AND FUEL HANDLING

The cores of reactor plants V-407 and V-392 are similar as to design and use practically all technical solutions on the advanced core of operating WWER1000/320. The prototype of the advanced fuel is standard fuel assemblies (FA) with stainless spacing grids and guiding channels which have been in operation at WWER-1000 since 1982. Originally the standard fuel was operated in the mode of two-year fuel cycle, then the transition was done to three-year fuel cycle with the corresponding increase of average burnup.

Operation experience of standard fuel revealed certain drawbacks both concerning efficiency of fuel utilization, and design of fuel assembly (highly absorbing material within the active part; boron-based burnable absorber; low design service life; one-piece structure). Therefore designers and manufacturers of russian fuel for WWER-1000 have developed the advanced FA with zirconium structural materials and this FA is being implemented at present.

4.1. Advanced fuel assembly

Advanced fuel assembly (AFA) has been developed both for replacement of standard fuel at the operating reactors, and for new nuclear power plants with advanced WWER. The main difference of AFA, being the most effective as to economy, from standard fuel is application of only zirconium structural materials in the assembly active part. This allowed (in combination with specially developed refuelling patterns) to reduce the specific consumption of uranium approximately by 13%. Application of gadolinium burnable absorber instead of boron absorber allows to reduce this index by approximately 5% more. Application of AFA allows also to reduce enrichment of makeup fuel. Using of uranium-gadolinium fuel allows to reduce neutron fluence to the reactor vessel, to improve flexibility of fuel cycle, to exclude expenses for operation and storage of burnable absorber rods.

Guiding channels (GC) for absorbing elements are optimized by outer diameter and wall thickness in such a way that to improve the conditions of insertion of absorbing elements under the mode of free drop, to keep sufficient DNBR in the surrounding fuel rods and to provide for the required strength of GC, as the load-carrying component. Tests are being performed of GC of zirconium alloy Э635 of decreased radiation creep that could be used at higher burnup.

Difference in linear expansions coefficients of guiding channels of the assembly and reactor core barrel is compensated by increase of the working stroke of spring block of FA cap. For connection/disconnection of AFA stainless cap and GC a simple device is used not requiring replaceable fasteners or complicated fixtures with power nut drivers. As the inspection stand is available the procedure on dismounting or mounting the AFA cap takes a minimum time.

Absorbing elements are upgraded as well. The combined absorber is used in them comprising boron carbide and dysprosium titanate. This allowed for two times increase of absorbing element service life. For the absorbing element cladding the new alloy is applied with improved mechanical properties and radiation strength. This allows to decrease the cladding thickness and improve the efficiency of absorbing rod.

So, the advanced fuel provides for improvement of safety and economic efficiency of reactor plants of new generation. Nowadays all the mentioned improvements of the core are being checked and implemented at the operating NPP with WWER-1000, therefore the experience obtained is a reference one for the advanced reactor plants V-407 and V-392.

4.2. Fuel handling system

In design of fuel handling systems for the advanced reactor plants V-407 and V-392 some changes are introduced into the process and structure of fuel handling equipment. These changes provide for improvement of safety, of maintenance conditions and simplification of fuel handling procedures.

For example, in V-407 design the «lower» supply (shipment) of nuclear fuel to the reactor compartment is used at power unit. With this, the height of lifting of in-station packing set for fresh fuel and transport container for spent fuel is about 1,5 m excluding by this the occurrence of nuclear accident in case of drop of fuel packages. In the design of fuel handling system of reactor plant V-392 the nuclear accident in case of drop of transport packing set is prevented by installation of shock absorbers at the places of packages lifting to the height exceeding the design one for these packages.

All fuel handling equipment of reactor plants V-407 and V-392 (in-plant transport packing set for fresh and spent fuel, leak-tight bottles, bottles of defective assembly detection system) has the cells for fuel assemblies made of hexahedral tubes. This measure provides for improvement of nuclear safety under accident situations and also prevents mechanical damage of fuel assembly during its installation and withdrawal from the fuel handling equipment.

5. MONITORING AND CONTROL SYSTEM

The existing nuclear power plants were constructed by designs of 60-70-ties with the use of automatics, equipment, cables and actuating mechanisms manufactured mainly at the enterprises of the Soviet Union at that time. Monitoring and control system (MCS) of Russian plants may be conventionally divided into three generations.

The first generation includes MCS of the reactors commissioned before 1975. Specific feature of these systems is wide application of remote control from the operator's panels, remote control of process parameters and relatively simple automatic devices (process protections, automatic control, interlocks, signalling). MCS of the second generation is characterized by wide application of measuring and control instruments with the unified electric signal, logic control devices, aggregated monitoring systems for the plant process systems. In the systems of the second generation the links between the control devices for the reactor, turbine and other system were considerably extended, the scope of monitoring and automation of processes was increased. In the devices of automatic control and protection of the reactor the elements of microelectronic technique are applied. The specific feature of MCS of the third generation is wide application of microprocessing and computing technique for control of processes. For representation of information to operator both the mimic panels, and alphanumeric and graphic displays are applied.

For reactor plants V-407 and V-392 new systems of monitoring and control are developed. They apply widely the microprocessing technique for implementation of all MCS functions including safety functions. The requirements of new regulatory documents are taken into account as well as recommendations of international standards, up-to-date principles of system construction, such as:

– high automatization level of processes;
– regard for operation experience of the existing systems and the latest achievements in the world practice in creation of control rooms;
– developed information support to operator, high functional reliability and self-diagnostics of hardware;
– redundancy, independence, diversity, resistance to common cause failures;
– assurance of serviceability under internal and external impacts including accident conditions;
– reducing the maintenance work scope and the number of personnel engaged.

Development of hardware for new MCS is performed according to complex programme of Minatom of Russia prescribing the development of software-hardware for MCS engineering for the stations reconstructed, under construction and under design, including WWER-640/407 and WWER-1000/392.

6. CONCLUSION

In Russia a complex of R&D work is being carried out on development of power units of new generation with reactor plants WWER-1000/V-392 and WWER-640/V-407, meeting the new requirements for safety and economic efficiency of electric power production.

In the advanced WWER many new technical solutions are applied with a view to improve safety, to optimize economical indices and to minimize the expenses on maintenance of the station equipment and systems.

THE DEVELOPMENT OF KNGR CONTROL ROOM MAN–MACHINE INTERFACE DESIGN

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Abstract

KNGR MMI design has been developed for the last 7 years as a part of Korea Next Generation Reactor (KNGR) design development. The KNGR control room has the common features of advanced control room such as large display panel, redundant compact workstations, soft control, and computerized procedure system. A conventional type safety console is provided as a backup when operation at the workstations is impossible. The strong points of an advanced control room are based on the powerful information processing and flexible graphic presentation capability of computer technology. On the other hand, workstation based design has a weak point that the amount of information to be presented in one VDU is limited. This can cause navigational overload and inconsistent interfaces and provide chances for performance errors/failures, if not designed carefully. From this background, the regulators require licensees to follow strict top-down human factor engineering design process. Analysis of operating experiences and iterative evaluations are used to address the potential problems of the KNGR advanced control room MMI design. But, further study is necessary in design area like CPS design, where experiences or design guidance is insufficient. Further study topics for KNGR advanced control room MMI design development are discussed briefly in this paper.

1. DESIGN PROCESS OF THE KNGR ADVANCED CONTROL ROOM

A structured Man– machine Interface (MMI) design process is being coordinated throughout the KNGR design program, which is to develop a standard Advanced Light Water Reactor (ALWR) design for commercial operation after the year 2010 in Korea. The program is organized in three phases according to each developmental state. Major activities of the Phase I, which was completed in 1994, were to develop the plant design requirements and concepts. The concept of MMI design had been developed with other plant design concepts through the Phase I program. In Phase II, completed in February 1999, basic design activities have resulted in a MMI basic design and documentation supporting submittal of a Standard Safety Analysis Report (SSAR) for licensing. The current Phase III stage is planned to be a three year program creating detailed design information and performing additional design evaluations. This phase is scheduled to be followed by a construction project for the first KNGR units.

1.1. Creation of the control room design concept

The goal of the conceptual design phase was to establish a point of departure for evaluating and refining the MMI design through a formal human factors-based design process. To that end, basic features, called MMI resources, and their characteristics were identified. In addition, the control facilities including an initial Main Control Room (MCR) layout were created. The preliminary conceptual designs evolved from the current Korea Standard Nuclear Plant (KSNP) designs, with insight from the ABB-CE advanced control room(Nuplex 80+™) design and Design Certification process. Driving forces including the

EPRI ALWR Utility Requirements Document (URD) requirements, an OECD Halden Reactor Project study, and a KEPRI advanced control room survey transitioned the design towards a full-function compact workstation MCR.

In mid 1997, a MMI Joint Design Development team was convened at the initiation of Phase II. An experienced team of control complex designers and I&C engineers was assembled, complemented by experienced PWR operators. The mission of the team was to systematically assess the preliminary conceptual design features to determine those to be carried forward in the basic design.

The entire team and technical managers of KNGR design organizations formally reviewed the results and recommendations for each feature in periodic design review meetings. The resulting designs were integrated into a complete conceptual design, and were optimized where specific fundamental concerns were identified. The detailed evaluation results were documented in evaluation reports for each feature [1][4]. The ability of the design to address integrated design considerations, such as common mode failure of software based systems and continued operation with MMI equipment failures, was also assessed.

1.2. Development of a basic MMI design and documentation

The KNGR MMI design effort were made until early 1999 with design details and formal documentation being developed for each MMI feature and control facility. Human Factor Engineering (HFE) program plan [2] and other implementation plans for the HFE program element were established and executed to direct the HFE activities so that human centered goal can be met. Dynamic rapid prototypes of all MMI resources were developed for initial evaluation. Alarms, displays and controls were implemented for a selected set of plant systems and incorporated into a dynamic mockup. This part-scope control room workstation was driven by plant simulator models to allow operator-in-the-loop verification and validation testing. In parallel a set of human factors analyses supporting the design were performed.

A primary activity of Phase II was continued for the development and documentation of the MMI design. Early documentation of the design served three important functions; (1) to provide a concise design for mockup development, (2) to allow internal and external design review and (3) to support SSAR development and submittal.

Initially, an operation philosophy document was developed to establish a vision of how the KNGR would be operated. The operation philosophy provided staffing assumptions and operator functions in all modes of operation to allow a common perspective for MMI resource and system designers to proceed. A Human Factors Engineering Standards, Guidelines, and Bases (HFESGB) document was produced specifically for the KNGR MMI design. This program guidance document, based on an extensive set of accepted human factors sources [3], facilitates consistent application of the HFE principles by all design team members and organizations.

The focus of Phase II centered on the basic design development of the MMI resources and control facilities. MMI resource functional requirements and control facility system requirements systematically documented from a wide variety of sources. The conceptual MMI designs developed through the Phase I were refined to meet these requirements, as well as functional and task requirements obtained from the results of human factors analyses and the guidance of the KNGR HFESGB. The resulting designs were documented in design reports for MMI resources and system descriptions for control facilities.

The complete set of MMI design documentation has been captured in a document handling system that supports formal tracking of the review and comment process. Review of the design documents was performed by both MMI design team members and independent review organizations. The standard issues of the above KNGR documents have completed through the Phase II program.

1.3. Mockup developments and evaluations

A fundamental approach in the KNGR MMI basic design process is early development and use of prototypes and a dynamic mockup. Rapid prototypes were developed in parallel to MMI resource designs to facilitate interactive review and refinement. The rapid prototypes were used to evaluate design alternatives or confirm acceptability of design decisions, particularly for new and unique MMI resources.

A “deep slice” of the basic design process was conducted to create displays, controls and alarms for a small set of selected plant systems and computerized operating procedures for a specific event scenario. The designs were based on the KSNP systems and the results of a limited scope task analysis. The “deep slice” designs were used to create a dynamic mockup of one complete compact workstation. The mockup includes an alarm Video Display Unit (VDU) and three multi-function VDUs, three soft control devices with associated confirmation switches, and system level actuation switches. A complete Large Display Panel (LDP) is implemented in one-quarter scale using rear projection technology. The entire mockup is driven by KSNP full-scope simulator models.

Two principal evaluative activities conducted in Phase II were suitability verification and preliminary validation. Suitability verification is an iterative process to be continued throughout the MMI design. It addresses the issue of whether the form and arrangement of individual MMI resources support operator task accomplishment. This is accomplished by (1) using a top-down approach to assess usability of the mockup MMI resources for task performance and assess adherence to high-level human factors design principles, and (2) using a bottom-up approach to determine conformance of the mockup implementations to established HFE criteria in the HFESGB. The initial suitability verification effort was performed by human factors specialists and MMI designers, supplemented by licensed plant operators. The result was a number of formally documented findings that are providing feedback to designers early in the design process to avoid costly design changes.

Preliminary validation activities were also conducted on the mockup to allow an initial assessment of the integrated ensemble of MMI resources. For the assessment, two crews of licensed operators from a KSNP unit joined the design team for a while to conduct post-trip operations on the mockup for a hypothetical steam generator tube rupture event. The evaluations were performed for the cases of using the computerized operating procedure and paper procedure respectively. The specific topics including workload, task performance, crew errors, and situational awareness were also assessed.

1.4. The detailed design

Phase III of the KNGR MMI design will focus primarily on the application of the basic MMI design to a wider range of plant systems and on continued evaluation of the design. The detailed designs will be generated by cognizant engineers based on input from task analysis and plant systems requirements, and will be transmitted to I&C system designers for implementation and used for mockup expansion.

The continued evaluation of the MMI resources and control facilities will be performed as following:

- Continued suitability verification of individual MMI resources to maintain conformance to human factors guidance and task requirements as additional design details are developed and design changes are implemented on the mockup,
- Design validation as the mockup facility becomes more robust, to evaluate and demonstrate the integrated operational capability of the KNGR MMI and MCR during normal, abnormal and accident conditions,
- Demonstrations (and evaluation, if necessary) of KNGR MCR operations (1) with actual performance parameters, such as time responses, and (2) with postulated degraded MMI operating conditions including testing of proposed mitigation strategies, where required,

2. CONTROL ROOM AND MMI DESIGNS

Some of the most evident changes are in the KNGR Man–machine Interface Systems (MMIS), where state-of-the-art technologies are replacing conventional control room and the Instrumentation and Control (I&C) systems. The KNGR MMIS distinguishes itself from current designs by employing all digital I&C systems and data communications, and primarily video-based Man–machine Interfaces (MMI) which incorporate modern human factors principles. The MMI maintains well-proven KSNP features, such as critical function monitoring, while adding advanced design features. Advanced design features being incorporated include: (1) a compact workstation-type control room layout, (2) a large display panel, (3) computerized operating procedures, (4) soft controls, and (5) a reduced number of fixed location displays and alarms

The primary control and monitoring facilities for operators are full-function compact workstations, which provide unprecedented operational flexibility and integration. Each of three redundant workstations features paired sets of monitoring VDUs and soft control flat panels, allowing access to, and effect of, both safety and non-safety controls. Complementing these soft interface devices, computerized operating procedures reduce operator response time variability and operator error probabilities. This full complement of soft devices accommodates varied task requirements and operator preferences, and permits incremental MMI refinements with minimal impact throughout the plant design process and operating life. A Large Display Panel is located in the front of the control room. It provides fixed-location indication of high priority alarms, parameters and component status otherwise unavailable in the compact workstation environment. It also provides continuous display of critical safety function status as per regulatory requirements. A minimum set of fixed-position component controls and operator modules are located on a safety console, designed to complement the workstations during post-trip conditions. The control facilities support control room operator staff reductions, compared to conventional designs, while offering unprecedented expansion and reconfiguration potential. The control room is designed to explicitly meet the requirements of the Korean-URD. The control facilities feature the main control room, including its compact workstations, large display panel and safety console. Each of these will be briefly described in the succeeding sections. Other control facilities include a remote shutdown room and local control stations.

2.1. Main control room

The KNGR Main Control Room (MCR) layout is shown in Figure 1. The key features supporting the operating crew’s ability to maintain efficient and safe plant operation include:

– Two identical, full-function workstations supporting direct plant control and monitoring by a Reactor Operator (RO) and a Turbine Operator (TO), respectively,
– A third identical workstation supporting normal monitoring and crew coordination functions of a Control Room Supervisor(CRS) and serving as a backup to the RO/TO workstations,
– A Large Display Panel (LDP) providing overall plant operational and safety assessment,
– A Safety Console providing control capability for all Class 1E, safety-related components for the plant safe shutdown even in the event of complete workstation failure.

Advantages of this control room layout include enhanced communication between operators, good visibility of the extended scope LDP, ease of accommodating design and job allocation changes, and convenient access and egress routes.

2.2. Large display panel

The Large Display Panel is a wall-mounted overview display including Safety Parameter Display System (SPDS) and Bypassed and Inoperable Status Indication. The fixed display section of the LDP provides continuous, parallel display of key alarm, component, system, and parameter information. This complements the workstations' MMI with a spatially dedicated graphical depiction of the plant. A variable display section allows operators to selectively display pertinent information to support crew coordination. The LDP reference design used on the dynamic mockup is shown in Figure 2.

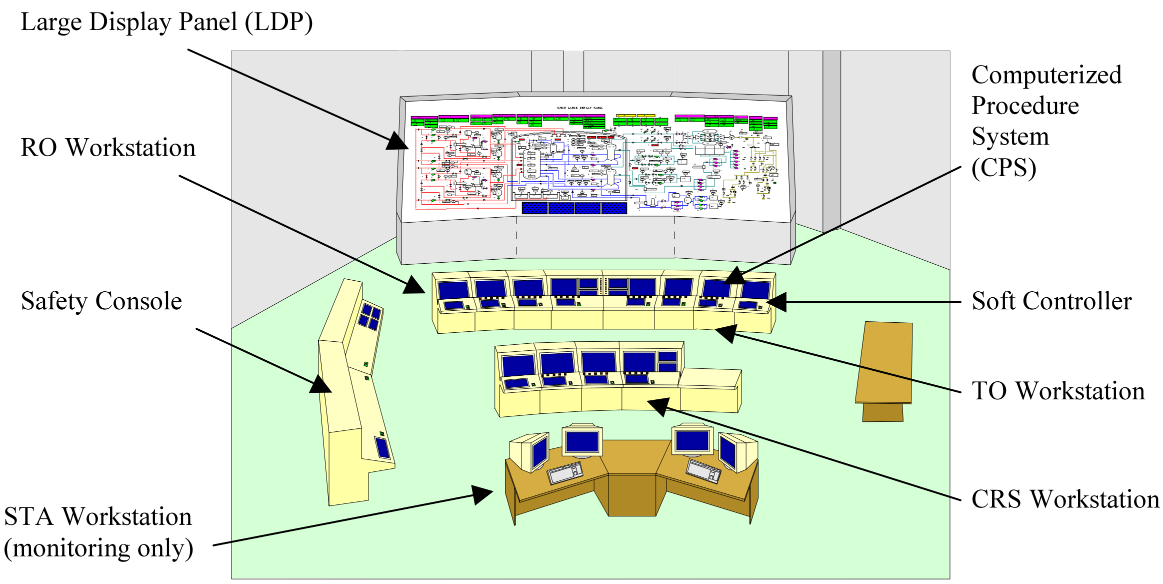

FIG. 1. Main control room layout.

2.3. Compact workstations

Each MCR workstation provides devices for access to all information and controls necessary for one person to monitor and control all processes associated with the plant operation and safety. This includes both safety and non-safety systems. The workstation design for Rector Operator (RO) and Turbine Operator (TO) is illustrated in Figure 3. Each workstation contains the following:

– One alarm VDU with trackball user interface,
– Three VDUs supporting process monitoring or electronic procedures with trackball user interface,
– Three flat panel displays used as soft controllers for process and component control; each working in conjunction with one VDU and using a touch sensitive user interface,
– Dedicated push-buttons for Manual Reactor Trip and ESF system actuation
– Laydown area for logs, drawings, backup paper procedures, etc.

The major advantages of the compact workstation approach are its (1) operational and design flexibility, (2) compactness and simplicity, (3) ability to cost effectively accommodate changes, and (4) provision of an enhanced integrated environment for CPS and operator aids.

2.4. Safety console

The control room includes a safety console providing Class 1E controls for all safety-related components, independent of the workstations. The safety console is intended to be used for surveillance testing during normal operations and to be staffed by an additional RO during post-trip operation. It provides all of the required safe shutdown capabilities, including mitigation of accidents, and helps address the common mode failure issue in the compact workstation design.

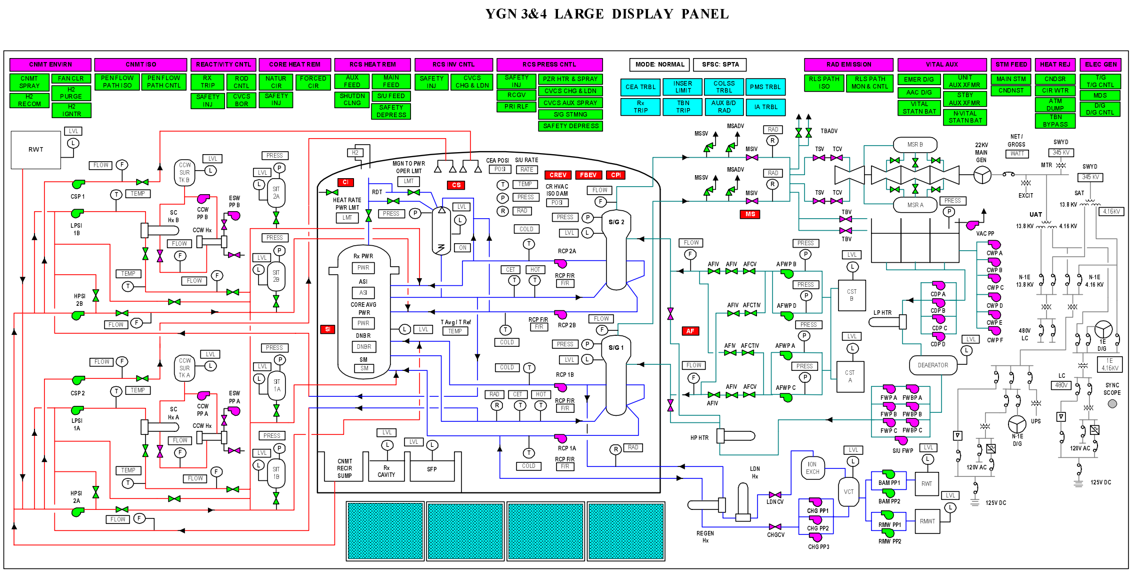

FIG. 2. Large display panel.

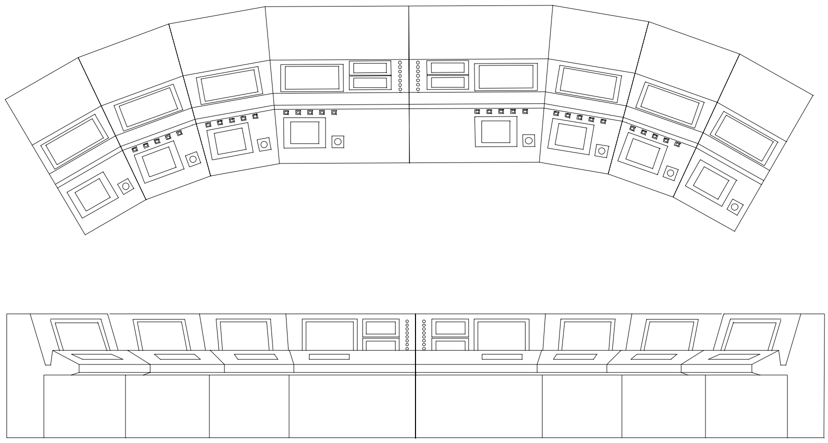

FIG. 3. Workstation design.

The MMI resources provide a standardized set of monitoring and control MMI designs that are designed to and evaluated by human factors principles. The MMI resources are used to implement the detailed MMI designs for all plant operator interfaces. Each of the primary MMI resources is described briefly below.

2.5. Workstation VDU displays

The MCR workstations allow simultaneous access to plant information through selectable displays on four VDUs per workstation. A wide variety of display formats include system mimics, major plant functions or conditions, technical data sheets, trends and graphical information, and application program access. All are designed to support specific operator functions. Multiple methods are provided for convenient access to the display set including navigational access through menus, direct access through format chaining from other displays (or alarms and procedures) and a dedicated mechanism such as function buttons or voice entry. A major function of the VDU displays is to provide a soft control link allowing the operator to quickly select a component or process control on the soft controllers directly from display pages.

2.6. Soft control

The KNGR program has adopted a “Soft Control” approach as the primary means to effect control actions. The “Soft Controls” utilize flat panel displays to emulate the physical switches and manual/auto stations that populate conventional plant control panels. Use of software-based control allows a standard interface device to assume the role of numerous physical devices. This has the advantage of allowing operator access to all plant controls from a single workstation, design flexibility and the ability to easily accommodate changes, and simplification of hardware procurement and maintenance. A set of divisional confirmation switches maintains divisional independence allowing safety and non-safety control to be effected from the same device.

2.7. Alarm system

The alarm system is designed to improve the annunciation process, by incorporating methodologies that:

- Reduce the total number of alarms that an operator must cope with
- Distinguish between true process deviations and sensor failures
- Minimize the occurrence of “nuisance” alarms
- Prioritize the relative importance of alarms so the operator can focus on the most critical alarm conditions first while deferring less critical alarm conditions
- Determine the impact of alarms on plant operations and distinguish these from lower level system alarms.

Highest priority alarms, such as those for critical safety functions, are presented in fixed locations on the LDP. All alarms are presented in list form on a dedicated workstation VDU as well as through relevant locations in the VDU display hierarchy.

2.8. Computerized procedure system (CPS)

The CPS is implemented as a passive design incapable of originating direct control action. The center section of a VDU screen shows a procedure overview in the form of a scrollable flowchart of all procedure steps. CPS allows assignment of procedural steps to specific operators (e.g., the RO and TO). An instruction window provides a general statement of the action(s) the operator should perform. An “action details” window provides the list of manual actions taken to perform the procedure step. To support decision-making the CPS provides real-time plant information or format chains to appropriate VDU displays. Direct access to the soft control MMI is afforded through format chaining as well. A “forcing function” approach keeps the operator in the evaluation loop, preventing the human from blindly following the computer for procedure execution.

Major advantages of the KNGR CPS include a significantly improved approach to continuously applicable procedural steps, direct access to information and controls for procedure execution, improved procedure integration in the compact workstation environment, and reduced error probabilities and response variability.

3. ISSUES OF KNGR CONTROL ROOM MMI DEVELOPMENT

3.1. Challenges in designing an advanced control room

Even though the computer technologies provide the high functionality for design flexibility to make the man machine interface more suitable for human being, they have several problems in application to nuclear power plant operation. The computer systems are usually highly centralized performing several functions in one processor. This is very undesirable in view of system reliability because it is prone to simultaneous failure of multiple functions. Conventional control rooms are better in this sense because they are highly distributed. Another problem with the computer system is that its software verification is very difficult because of the complexity of the software itself.

The information processing provides opportunities for relieving operators from complex data handling that are not suitable for operators in complicated operational situations. The computer graphic technologies allow presentation of operational information in a form that facilitates operators’ intuitive use of information without big cost. However this inherent computer capability for flexibility and automation brings the complexity of MMI and provides chances for performance errors/failures by operation staff in nuclear power plants if not designed carefully. In an advanced control room, many resources are typically used to support the operator task. Therefore, the lack of integration of multiple MMI resources can create performance problems. The limited amounts of information that can be presented in one VDU at a time make the advanced control room design difficult. This is because of navigational overload problems and inconsistent interface representing the same part of the plant. From this background, the regulators require licensees to follow strict top-down HFE design process.

3.2. Human factor engineering process

Systematic application of HFE is the key element for the successful development of safe and efficient control room and man machine interface[3]. For the successful completion of the complex KNGR MMI design process, a multidisciplinary team of human factor specialists, computer specialists, system engineers, and plant operators work together as a team from the stage of conceptual design through the validation process. Starting from the KNGR MMI design concept, the design has been going and will go through several iterations of analysis, design and evaluation. The system hazard analyses such as operator experience review, analysis of plant safety functions, analysis of critical operator actions from PRA, are performed to identify high risk tasks and critical operator errors against plant safety. The results are addressed or used in the MMI design. For the systematic application of HFE principles, related principles are searched and adapted to KNGR MMI concept to become HFE standards and guidelines for MMI design. The usability of the MMI design will continuously improve as iterative evaluations and improvements are implemented until final validation of the MMI design. The acceptability of the design will be demonstrated not only by the final validation but also by the repeated evaluation and studies throughout the design process. In this process, high fidelity full scope simulators will be used to facilitate and accelerate the improvement. A group of plant operators has been and will be extensively involved to provide operating experiences to identify any unforeseen issues in the design. An independent multi-disciplinary team will review the design to reflect diverse input for the design and thereby to correct the design and remove any bias in the design.

3.3. Control room design against computer system failures

Defense in depth strategy is reflected in KNGR control room design. There is a safety console where operator can perform Emergency Operational Procedure (EOP) or safe shut down operation when the workstations are unavailable for operation. There are three redundant workstations in main control room. When one of the RO or TO workstation fails, CRS workstation is used by the RO or TO. When one of the Cathode Ray Tube (CRT) or Flat Panel Display (FPD) is not working, there are still three pairs of information display and soft control display whose operation can continue. In case of CPS failure, operators can use the paper procedure whose format is consistent with the computerized procedures. Thus, the KNGR control room is designed to be very robust to any computer failures.

3.4. Design against common mode failure(CMF) of software

Design and analyses were performed to resolve the CMF issue as per the NRC positions stated in SECY-93-087. It should be noted that the KNGR MMIS differs from other advanced MMIS designs in that the non-1E soft control MMIs provide control input signals both to the safety and the non-safety equipment. Since the soft control MMIs provide a common interface for both safty and non-safety system control inputs, it is necessary to evaluate the potential impact of a common mode failure of the software used in the soft control MMIs to ensure it does not compromise the basic CMF protection provided by the diverse safety and non-safety systems.

CMF of Safety System Software: An overall strategy has been developed for the KNGR MMIS and is to make sure that the occurrence of a CMF of safety system software can be safely mitigated. In other works, the KNGR MMIS with diverse non-safety controls and indications at the RO and TO workstations, and non-safety systems such as Alternate Protection System(APS) provide sufficient means to bring the plant to a safe shutdown condition, and that the capability is adequate for any of the Chapter 15 event initiators. One notable feature of the KNGR MMIS would be that control actions could be taken through the soft controls. No credit would be taken, however, for the safety analyses, except for the hard wired switches incorporated on the workstation panel to comply with the NRC SECY-93-087 Position 4, which requires, independently from safety system, system level controls for safety functions.

CMF of Workstation Soft Control: For the KNGR MMIS, careful consideration is made to evaluate how a failure of this non-1E qualified soft control is related to the software CMF issue. A criterion for the acceptance of the soft control is if the plant can be brought to a safe shutdown condition when a CMF of the soft control MMI occurs. For the KNGR design, the failure will have no impact on the automatic protection system actuation or manual safety control through Class 1E MMI. This will be achieved through the safety console capabilities which will be credited to address this failure. Since the soft control MMI provides input signals to the safety system, it must be demonstrated that the CMF failure in soft control MMI can not impact the operation of the safety systems.

Four Quadrant I&C Architecture: The KNGR I&C architecture maintains a four quadrant architecture for both monitoring and control. This architecture provides distinction between 1) safety and non-safety I&C systems and 2) monitoring and control systems as shown in Figures 4 and 5. The distinction between safety and non-safety systems is particularly important because it is the defining line for design diversity which provides a means to address the concern of common mode failure of digital I&C systems. The distinction between control and monitoring is also important to separate the processing intensive monitoring functions from the less processing intensive, yet more time critical, control functions. Safety control MMI is provided by Class IE, flat panel display operator modules and switches on the Safety Console for credited control capability of safety equipment. Non-safety control MMI is provided by soft control MMIs on the workstations, which use diverse technology from the operators modules. As shown in Figure 4, the soft control also allows a control of safety equipment. Figure 5 illustrates similar diversity between qualified non-safety monitoring systems and MMI. Qualified monitoring is provided by the Type 1 Qualified Indication and Alarm System (QIAS). Non-Safety monitoring is provided by the diverse Information Processing System (IPS). The qualified MMI is provided by seismically qualified FPDs and LDP. Non-safety MMI is provided by the workstation CRTs, which are diverse from the flat panel displays.

3.5. Communication independence of soft control MMI

With regard to the independence of safety system, IEEE 603-1991 states in section 5.6 that "redundant portions of a safety system provided for a safety function shall be independent... and that no credible failure on the safety side of an isolation device shall prevent any portion of the safety system from meeting its minimum performance requirement."

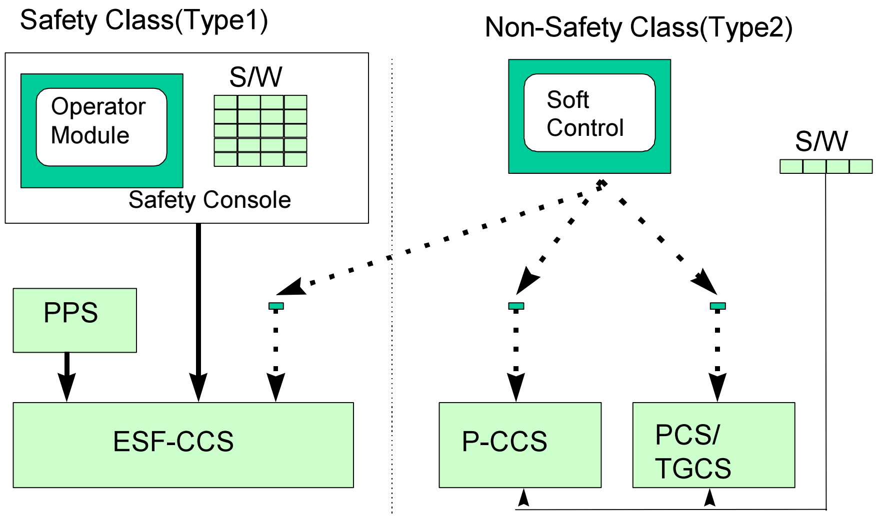

FIG. 4. The concept of control MMI diversity.

US NRC SRP provides guidance for the evaluation of conformance to IEEE 603 for computer based systems like the following: "If a digital computer system used in a safety system is connected to a digital computer used in a non-safety system, the review should confirm that a logical or software malfunction of the non-safety system cannot affect the functions of the safety system." With regard to the communication independence of computers that are part of the safety system, IEEE 7-4.3.2-1993 states that "No data communication between safety channels and between safety and non-safety systems shall inhibit the performance of the safety function."

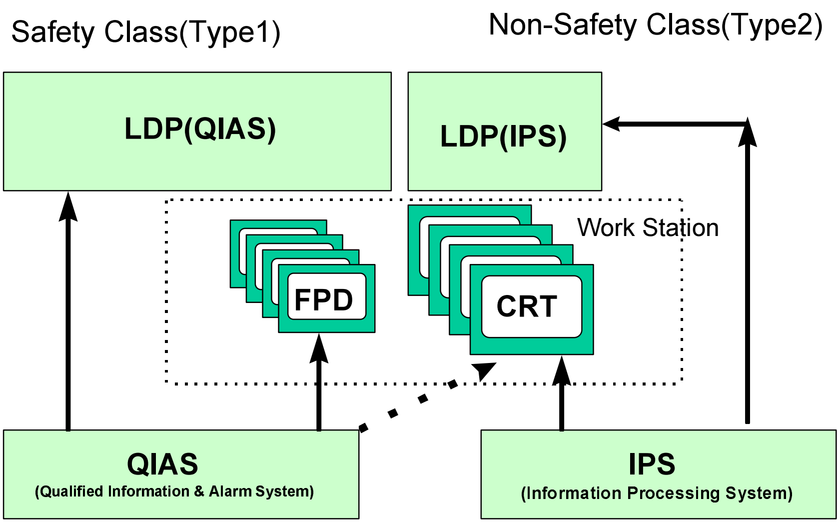

FIG. 5. The concept of monitoring MMI diversity.

In KNGR, the signal from a soft control MMI is connected to only one division of the Component Control System (CCS) at a time by using two design features. One is the divisional confirmation switches which allow control signals from a soft control MMI to take effect to the confirmed division only. The design approach is to select the control component via non-safety soft control MMI, then initiate the control action using a Class 1E actuation circuit confirm switch at each control station. The confirm switch is completely isolated from soft control MMI to ensure independence. The other feature is the use of a simple de-multiplexer which connects the communication lines between the soft control MMI to one division of CCS using an addressing signal from the soft control MMI. This is the second line of defense against soft control malfunction impacting more than one division. A simple non-1E de-multiplexing scheme that does not use any processor or memory is adopted to ensure the high reliability of the de-multiplexing function.

Non-safety Computer Function Overridden By Safety Function: In order to meet the basic independence requirement, the safety computer (ESF-CCS) must be able to override the non-safety computer (soft control MMI) when the safety system is performing its safety function. One of the design method to ensure this capability is a priority interlock. The priority interlock blocks any effect on ESF component control (system level) from the soft control MMI during safety function performance. ESFAS signals from the PPS override soft control signals at all times. The actuation signal from Class 1E ESF-CCS MMI (switches and operator modules) can also override the component actuation from soft control MMI. Control signals from the soft control MMI have the lowest priority. The operator can override the priority 2 interlock of ESFAS by using the soft control MMI or ESF-CCS MMI if the plant is in a safe state.

Communication Isolation Between Soft Control MMI and ESF-CCS: Buffering circuits in ESF-CCS division gateway are used to allow the handshaking with address checking between the soft control MMI and one division of ESF-CCS. This will assure the integrity of safety function by detecting and blocking the connection of soft control MMI to unintended division of ESF-CCS.

Operator Detection of Soft Control Malfunction: Prior to initiating the control action, it is expected that the operator would press the confirm switch for actuation after he checks the display of component selection which is not based on the selection signal generated from soft control, but based on the selection information fed back from the ESF-CCS. This enables the operator to detect any discrepancy between what he demanded at soft control MMI and what was actually received at the ESF-CCS.

3.6. Integration of SPDS and emergency operating procedure (EOP)

KNGR does not provide a stand-alone Safety Parameter Display System (SPDS), but the SPDS functions are integrated into the overall control room design. The violation of critical safety functions are annunciated through LDP tiles and CRT display to indicate the entry conditions to the proper optimal or functional EOP. The critical plant variables sufficient to provide information about critical safety functions are also provided on the LDP as an integral part of the fixed mimic displays. Plant function displays to assist the operator for the execution of EOPs in verifying and planning mitigation for violated critical functions are available in workstation CRT displays for RO, TO, CRS and all other operation staff, and personnel in TSC and EOF. Critical function and success path alarms are a meaningful framework to aid the operator in quickly identifying the significance of important alarm information. In KNGR, the Safety Function Status Checks (SFSCs) are a post-trip monitoring supplement to the emergency procedures. In the course of an event, violation of one or more SFSCs alerts the control room staff to emerging problems with the on-going mitigation strategy. Success path monitoring (SPM) algorithms provide alarms and displays of system/component availability and performance for each of the success paths.

4. FURTHER STUDY

4.1. Addressing the advanced control room (ACR) issues

As a compact workstation type control room is adopted for KNGR, It is necessary to address the potential problems of advanced control room for MMI resources design. It is believed that new type of cognitive errors are highly likely in VDU based advanced control rooms. To address this, we have taken the approach of identifying potential problems/issues, designing MMI against the potential problems, and evaluating the design to verifying the existence of the potential problems in the design product. KEPRI will perform this analysis with collaboration with the foreign consultants who have experiences in advanced control room design.

4.2. CPS design issues

Despite many potential advantages, there are also challenges in designing an effective computer-based procedure system such as decrease of operator competence for operation by exercising knowledge and the decrease of operator vigilance during operation using a computerized procedure. Results of the studies on computer-based procedures show many types of usability concerns. For example, since computer-based display devices may not be able to display all of these documents adequately, and partial scope may inhibit personnel performance. Integration with other MMI resources in the KNGR MCR, coordination among operators in use of CPS, appropriate level of automation in CPS design are some of the important issues to resolve in the design of the KNGR CPS.

4.3. Configuration management of S/W and computerized procedures

While the functionality of advanced control increase as new operator aids are incorporated in the man machine interface system, the burden of developing, maintaining the vast amounts of S/W became high. To be economical, it became crucial to develop qualification process and tools to be used for verification, validation, and configuration management of S/W. The development of methodology and tools for validating computerized procedure are necessary to reduce the efforts required for initial development and maintenance of the procedure.

4.4. Plant detection/diagnosis aids

As computer technology can provide complex information processing aids, further research is necessary to explore the possibility of aiding operator in detecting slowly developing problems and diagnosing the faults from the plant process and alarm information. The endeavour to enhance plant control automation based on the modern computer technology will be also necessary to improve plant availabilty and safety.

5. CONCLUSION

The basic design of KNGR MMI has been completed and the detailed design is undergoing. In this paper, we discussed on the development process of the KNGR MMI, its major features, and some licensing issues for KNGR MMI. Some of the research topics to be undertaken in the future were discussed briefly as well. Presently, the licensing precedents in U.S. ALWRs provide KNGR MMI designers with general directions for addressing the issues related to full digital MMIS and human factor engineering design process of advanced control room. Systematic development of requirements and design documentation as well as early human factors evaluation of the design using a simulator-driven dynamic mockup have been achieved. This leaves the KNGR program well prepared to pursue licensing of the standard design, as well as to implement the detailed design in KNGR construction.

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DEVELOPMENT AND VALIDATION PROCESS OF THE ADVANCED MAIN CONTROL BOARD FOR NEXT JAPANESE PWR PLANTS

M. TANI, K. ITO, M. YOKOYAMA, M. IMASE, H. OKAMOTO

Abstract

The purpose of main control room improvement is to reduce operator workload and potential human errors by offering a better working environment where operators can maximize their abilities. Japanese pressurized water reactor (PWR) utilities and Mitsubishi group have developed a touch-screen-based main control console (i.e. advanced main control room) the next generation PWRs to further improve the plant operability using a state of the art electronics technology. The advanced main control room consists of an operator console, a supervisor console and large display panels. The functional specifications were evaluated by utility operators using a prototype main control console connected to a plant simulator.

1. INTRODUCTION

It is important to provide a Human-System Interface (HSI) system with which operators can easily pick up appropriate information among a large number of plant process parameters and correctly identify the plant state.

The design of main control room in Japan has been continuously improved from conventional single large board with hard-wired indicators and switches to functionally divided boards with CRTs as major information source. (see Refs. [1-2])

A fully digital I&C system including the advanced main control console is planned for the next Japanese PWR plant with a view to achieving increased safety, reliability, operability and maintainability.

It is desired to improve the safety and efficiency and to construct the HSI system suitable to the fully digital I&C system so that operators can correctly perform their tasks; diagnosis of plant problems, active planning and implementation of control actions.

A touch-screen-based main control console, which has the following features, has been developed to meet the above-mentioned objectives.

- Full-time sit-down-operation console for reducing monitoring areas and traffic.
- Touch operational HSI system with CRT and Flat Display panels (FDP).
- Plant information presentation, which should be shared by the shift supervisor and operators using large display, panels (LDP)
- Automation of high workload monitoring tasks such as plant trip, and presentation of its results.
- Suitable space allocation among large display panel, operator console and shift supervisor console for smooth communication among all the operators.
- Functionally distributed computer systems architecture for enhanced maintainability and system reliability.

This paper describes the design concept, enhanced operability features, system configuration and evaluation results of the advanced main control console.

2. ENHANCEMENT OF OPERABILITY

The recent design improvement trends of main control board are obviously directed toward the soft operation utilizing computer driven HSI devices. The benefits of the soft operation are to supply relevant process information necessary for the implementation of control as well as providing appropriate process parameters for facilitating tasks, in addition to saving spaces of instrumentation and Control switches. The advanced main control boards consists of an operator console, a supervisor console and large display panels.Figure 1 shows the configulation of the boards.

The advanced main control console also enhances the operability by taking advantage of the soft-operation as described below:

2.1. Touch operation

Control actions are composed of check process of their ready-condition before implementation, monitoring process of control feedback parameters (direct control and its side-effect parameters), and verification process after implementation. In order to improve the operational quality by providing all the necessary parameters for control actions, control switches and relevant parameters are integrated onto the same control display. In addition to the integration of control switches and relevant parameters, automatic check functions for implementation start are been introduced.

2.2. Automatic verification and display

In order to reduce peak load in the Trip/SI situation, automatic verification of system level interlock and sequence actions are performed.

The verification results are automatically presented on the CRTs, the FDPs and the LDP.

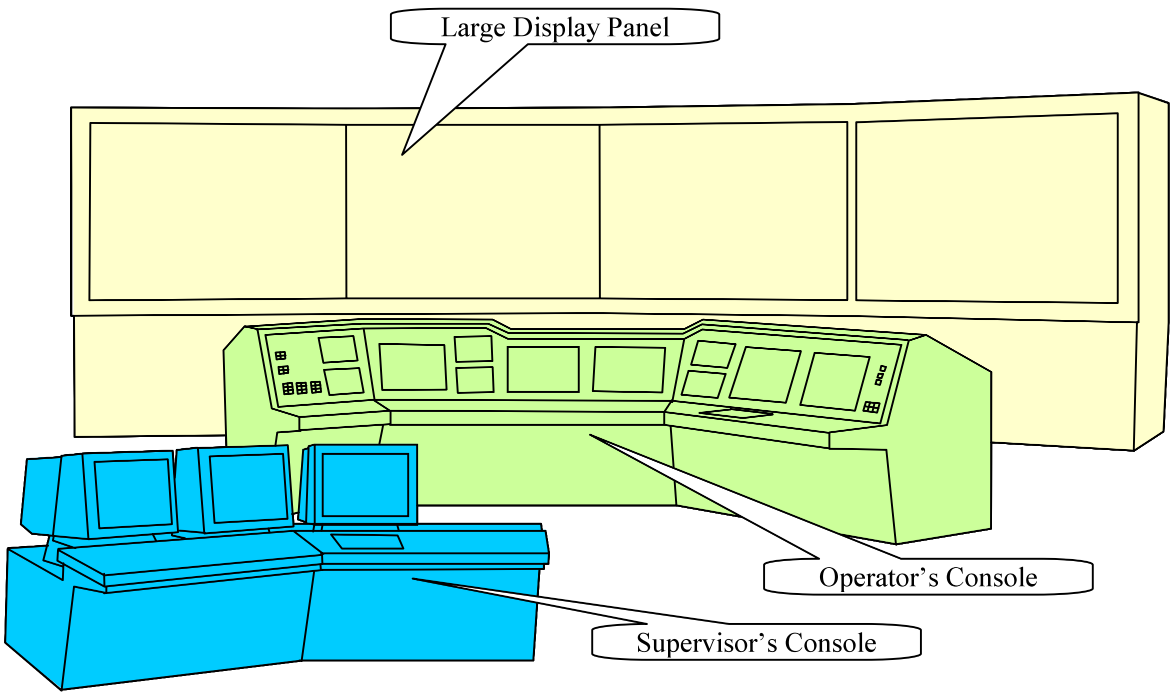

FIG. 1. Main control board configuration.

2.3. Inter-connected display request

All the control and monitoring displays are inter-connected in terms of functional and/or operational relationships and relevant displays can be readily requested by touching the keys presented on the rim of the displays. Figure 2 shows an example of integration monitoring and controling effect.

3. ALARM PROCESSING AND PRESENTATION

An alarm handling system that dynamically prioritizes alarms is used for avoiding information overflow and to facilitate plant state identification. The prioritized alarms and their relevant process parameters are provided in the graphically presented plant systems on the LDP with 3-level categorized color coordination. Individual and/or breakdown alarms are also presented on CRTs and FDPs with the priority categorization.

4. DESIGN RULES FOR TOUCH DISPLAYS AND LARGE DISPLAYS

Prior to adopting the touch-operation, various soft switches were tested, which included mouse, track ball, and touch screens, in terms of pointing accuracy, manipulability, and applicability to safety systems.

The manipulability of soft switches were tested in an experiment using a control system simulator. Equal manipulability between soft control switches and hardware control switches were verified for the steam generator feed water valve switch operation, which requires the most rapid and accurate manual control in the operation of PWR plant.

In order to establish appropriate soft switch button size, touch input test was performed experimentally, adopting touch input success ratio for a target area, touch point spatial distribution, and touch action time distribution as the evaluation indices. A control display configuration rule was specified based on these experimental results.

The large display configuration rules were also specified based on the examination concerning the distance and angle from the operator position, character and symbol size, and information density. An example of control display is shoown in Fig. 3.

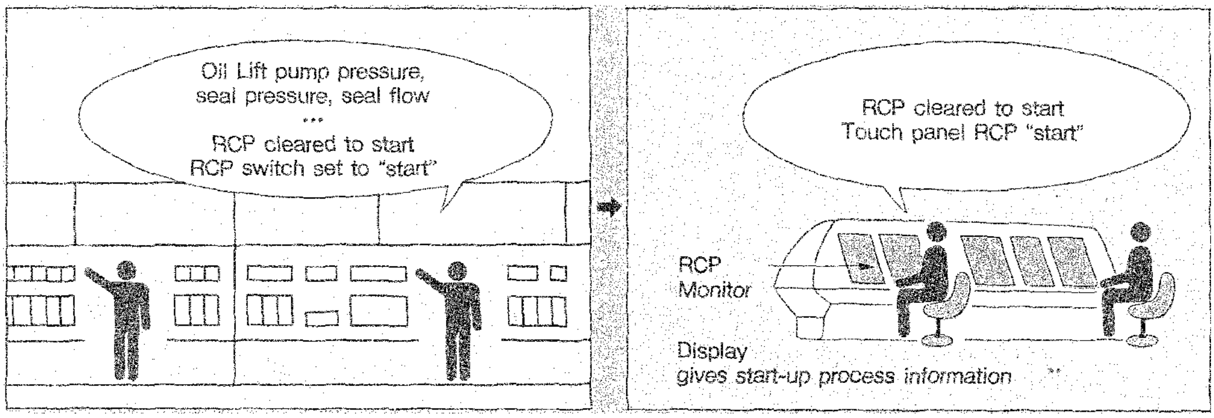

FIG. 2. Enhancement of operability.

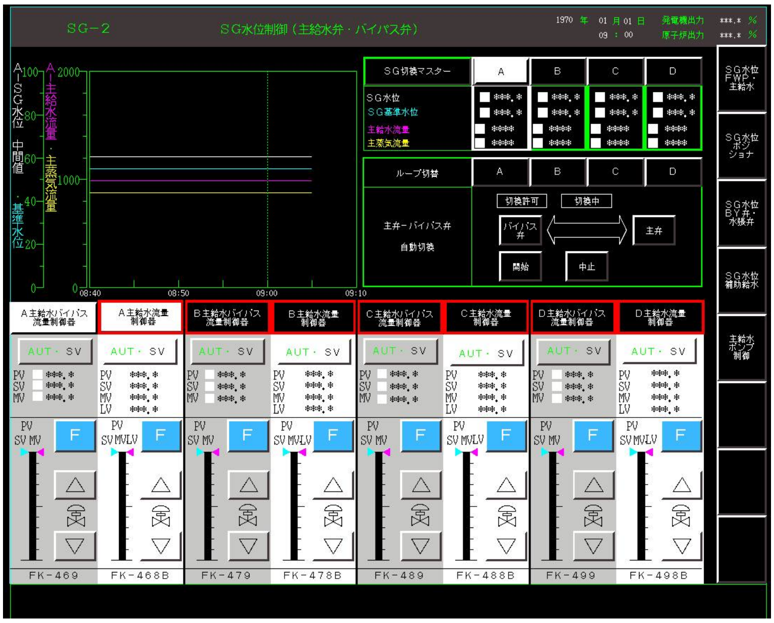

FIG. 3. Integrated displays for control & monitoring.

5. COMPUTER SYSTEM ARCHITECTURE FOR HSI SYSTEM

Because of its high reliability of using simplified software system and distributed maintainability, a functionally distributed computer system architecture, was adopted for HSI devices in which the LDP processors, integrated monitoring and control CRT processors, alarm processors, safety grade integrated monitoring and control processors are combined with a high speed data link system.

In terms of control function assignments between HSI devices and digital process control and protection systems, only the control request input function (i.e., input button function) was assigned to the HSI processors considering the software verification and validation.

Figure 4 shows the architecture of HSI systems.

Highly reliable software, which is as simple as the software of protection system, was adapted to the safety grade HSI device processor. Although control signals are separated from non-safety HSI device processors, display request signals can be input from non-safety processors for smooth monitor and control task procedure.

The two types of data link systems were used of data passing between the HSI device processors and the digital control and protection systems to meet the requirements for the process response time and the transmission data volume. The high-speed dataway systems are used for monitoring date transmission because of its large size data volume, and the multiplex systems for control data transmission because of its rapid response time.

6. OPERATOR WORKLOAD AND HUMAN ERROR PROBABILITY EVALUATION

The HSI improvement level, in comparison with the main control board of the latest plant in operation, was quantitatively evaluated in terms of operator physical/mental workload and potential human error probability.

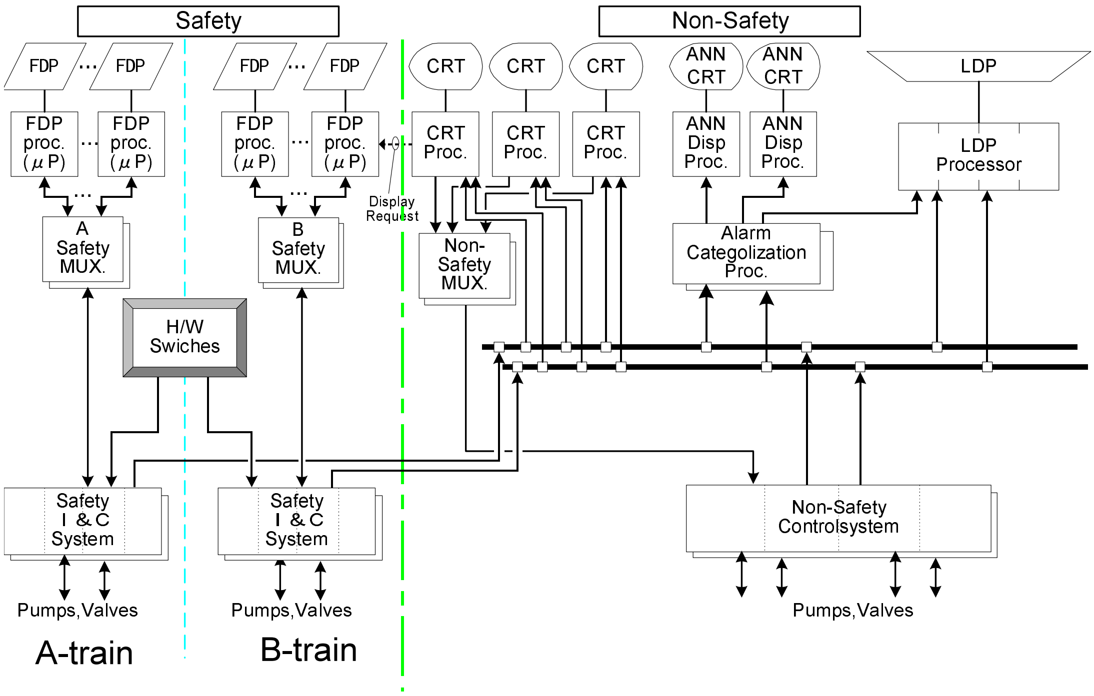

FIG. 4. HSI device processor architecture.

The physical workload was compared between the advanced main control console and main control board of the latest plant in operation, using the number of monitoring and control actions during accident operations as evaluation index.

Figure 5 shows the acuumulation method of mental workload in the operator model human processor model (see Ref. [3]).The mental workload was also compared similarly using a human information-processing model.

The physical and mental work load level of the advanced main control console appeared to reduce by about one third. Estimated potential human error on the other hand, appeared to be reduced by about one fourth. THERP (Techinique for Human Error Rate Prediction) were used in the valuation (see Ref. [4]).

Their results indicate that the integration of control switches and their relevant monitoring parameters, the inter-connected display request method, and the automatic verification function contribute greatly to the reduction (Table I).

7. VALIDATION TEST

In order to apply the advanced control console to nuclear power plant operation, the fully man-in-the-loop verification and validation (V&V) test has been planed. The design of the MMI shall be verified and validated with adequate facilities and resources at each phase of V&V phase in participant of the user (i.e. shift operators).

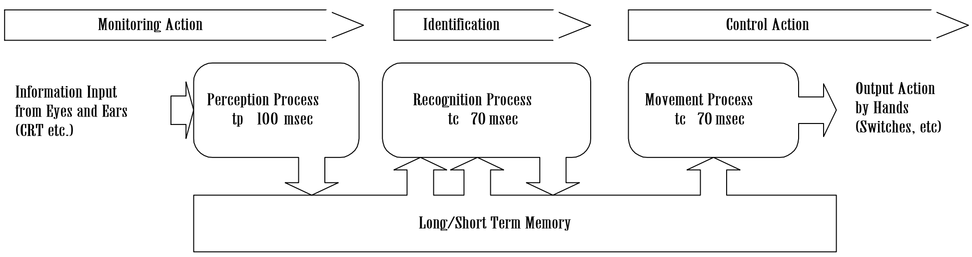

FIG. 5. Operator information processing model.

TABLE I. ANALYTIC EVALUATION RESULTS

Design goal	Evaluation method	Result (compared with improved control board)
Woakload	Operator Sequience Diagram	Abour 25% Off
	Model human processor	Abour 20% off
Human Error probability	THERP	Abour 25% off

The overall operability of the system was tested using a prototype advanced main control console connected to a full-scale plant simulator. This test was performed to confirm a shift supervisor, and one operator can monitor and control the whole plant under all plant conditions. The utility operating crews from different nuclear power stations joined the validation and went thorough under normal and accident conditions. The functional specifications of the displays integrated with control and monitoring functions, the display request method and alarm presentation system were validated by means of checking the implementation of operator's monitoring and control sequences defined in an operational manual. The validation results confirmed the improved operability of the advanced main control console.

In addition to the above mentioned operational sequence check, the operator's subjective rating was obtained through interviewing to confirm the qualitative evaluation points (e.g., cognitive process, control impression). The results suggested design goals of operability improvement were achieved.

8. CONCLUSIONS

It is believed that the advanced main control console would be applicable to the next Japanese PWR. The intermediate validation results confirmed the improved operability of the advanced main control console. The specification of the advanced control boards consoles will be finalized after continuing iterative validation tests. In the following validation phase, we also will attempt achieving further improvements of the advanced main control console. The effort will aim at the enhanced maintainability during the scheduled outage period and more advanced alarm presentation.

FIG. 6. Variation test facility of the advanced main control boards.

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TECHNOLOGIES FOR IMPROVING THE AVAILABILITY OF CURRENT AND FUTURE LIGHT WATER REACTORS

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Abstract

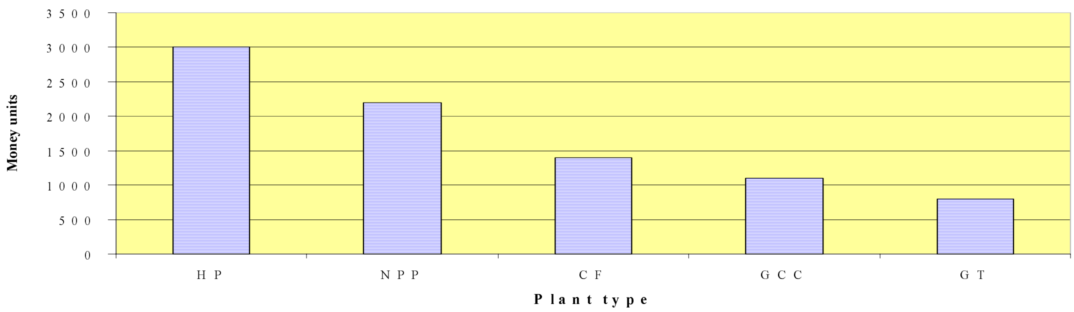
Construction of nuclear power plants involves large investments, much larger than for most alternatives for large-scale energy production. This is in particular valid when compared with the capital needed for building gas-fired power plants in areas or countries with an established infrastructure for distribution of natural gas. On the other hand, the fuel costs for a nuclear power plant are very low compared to the costs for natural gas and other alternatives, which means that the generation costs per kWhe of well performing nuclear power plants can be fully competitive. To this end, nuclear power plant owners and operators have been looking carefully at ways and means to achieve improvements with respect to power generation reliability and costs by equipment modernisation and/or modified procedures. In some countries, this has resulted in significant re-investments of revenues for modernisation programmes. In recent years, many such programmes have slowed down or been postponed due to reduced revenues in the wake of dramatic drop in electricity prices following deregulation. Typical areas of modernisation range from replacement of components or equipment to modification of system arrangements and structures, as well as introduction of new I&C systems and technology, for single functions or systems, or for the whole plant. Operating experience from well performing plants is compiled by utility organisations such as WANO (World Association of Nuclear Operators), EPRI (Electric Power Research Institute) and the EUR (European Utility Requirements) group, and also by the different nuclear vendors to serve as input for modifications and improvements in operating plants, and also as guidance for the design of new plants. A number of measures taken to overcome observed deficiencies and difficulties, suggestions for future improvements, and also implementation of specific design features were presented at the IAEA Technical Committee Meeting at Argonne National Laboratory (ANL), USA in September 1997. (proceedings of this TCM are presented in IAEA-TECDOC-1054). Utility views and vendor considerations are also reflected in the IAEA-TECDOC-968 "Status of advanced LWR designs, 1996". This paper reviews some ideas presented in these TECDOCs – and at other activities of the International Working Group on Advanced LWR technology (IWGALWR) – with respect to potential benefits for existing plants and for future designs, supplemented by some practical examples from modernisation programmes and new design development.

1 INTRODUCTION

In the 1970s, very ambitious nuclear power plant construction programmes were underway in many countries, but in the wake of the TMI accident in 1979 ordering of new capacity has been at a very low level. Orders for new plants have predominantly occurred in Asian countries, while quite a number of plant projects in some Western countries have been cancelled, postponed, or shelved; some operating plants have even been shut down.

Now, the reason for this negative trend in Western countries is not entirely due to the TMI accident, but it has beyond doubt had significant impact, in particular with respect to the public's perception of nuclear power. The accident also led to introduction of new safety requirements and new safety reviews, as well as delays and disturbances for operating plants and plants under construction.

Construction of a nuclear power plant represents a very large investment, and delays and disturbances during construction, or after commissioning, involves significant cost increases, which deteriorate the future economic viability. So, even though the TMI accident gave rise to reduced



Legend: HP = Hydro power; NPP = Nuclear power; CF = Coal-fired plant;
GCC = combined-cycle gas-fired plant; GT = Gas-turbine

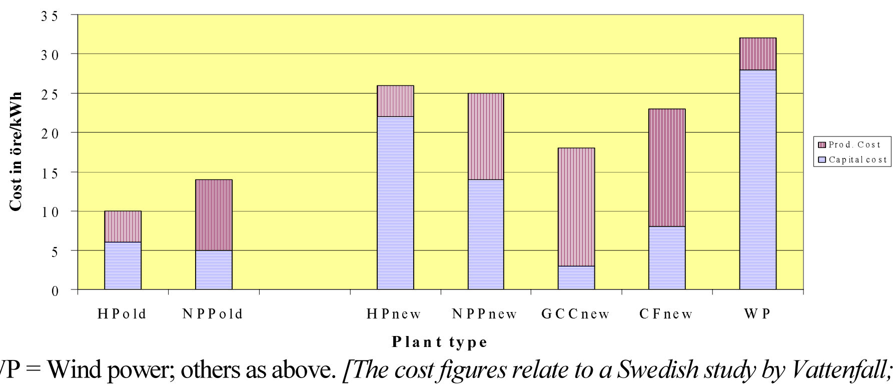
Figure 1. Schematic presentation of specific investment for some power plant types

public acceptance as well as anti-nuclear movements, the utility experience of cost increases and reduced economics – in addition to the "practical" demonstration of the financial risk of severe accidents – has been one major reason for being cautious about investments in nuclear power; another reason would be a lack of need for additions of large power generation capacities.

Continuous operation of the existing fleet of nuclear power plants without significant accidents or events, and at acceptable economics, will beyond doubt with time improve the nuclear power image – and its acceptability – by the public. And this will also be of decisive importance for a utility planning new generation capacity.

2 COST COMPETITIVITY OF NUCLEAR POWER

A quick comparison of the investment needed for construction of different types of power plants (cf. fig 1) indicates that there would be very little incitement for choosing the nuclear option. The relationships between the fuel costs and other production-related costs (fig 2) show a quite different picture, however. In reality, comparative energy assessments and experience bear evidence that nuclear power plants can compete economically, provided that they are operated reliably and at high energy utilisation. In other words, the better a nuclear power plant unit performs the better will the earnings, and possibly the revenues, be. The relationships are further affected by political decisions – with differences in taxes, and by inclusion of externalities (effects on the environment and the society). Figure 3 illustrates the situation when current Swedish taxes are added, and a minimum of external costs included. (The costs for externalities may rise rapidly in the future for the gas- and coal-fired alternatives. On the other hand, the net costs for wind power will drop due to a "green"



Legend: WP = Wind power; others as above. *[The cost figures relate to a Swedish study by Vattenfall; capital costs are based on 5% interest rate and 25 yrs (NP) annuities]*

Figure 2. Comparison between energy production costs for different power plant types

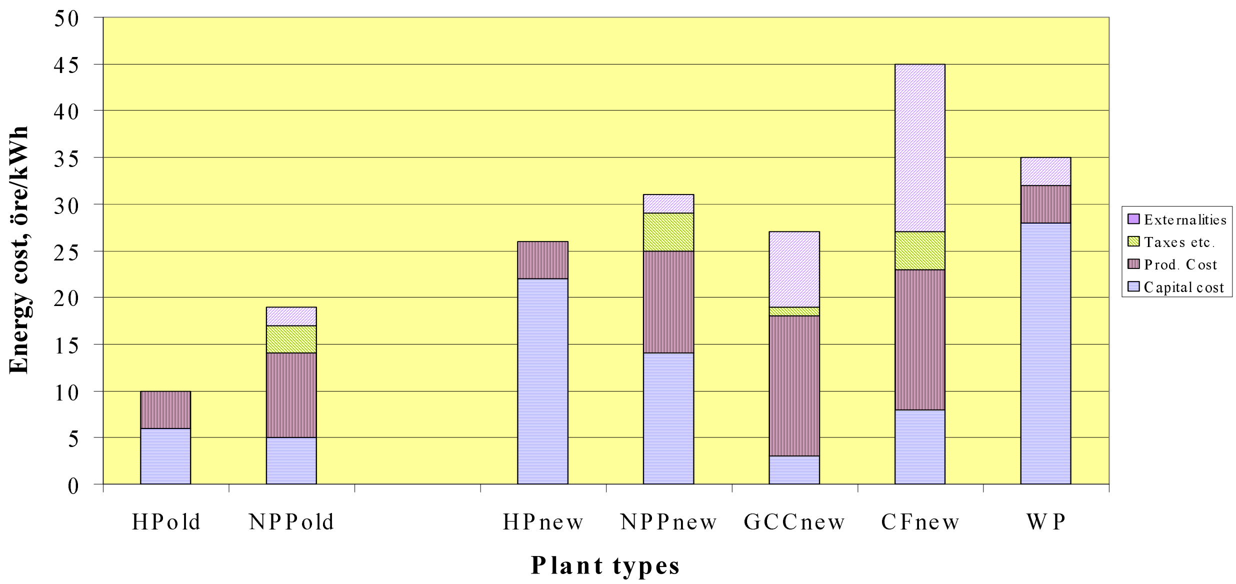


Figure 3. Cost comparisons including taxes and costs for externalities

subsidy of 25 öre/kWh!)

As noted by many sources, eg. in the paper "Designing for nuclear power plant maintainability and operability" presented at the ANL meeting [3], a proper plant design or configuration is a necessary but not sufficient prerequisite for achieving reliable and safe operation; the management, organisation and personnel of the plant owner/operator will be of paramount importance.

With respect to the possibilities for increased nuclear power deployment in the future attention must also be paid to the cost and duration of plant construction and for getting into operation. In the current economic "climate", investors clearly favour projects with low risk and rapid return on investment, ie. gas turbines and combined-cycle gas-fired plants have been preferred choices in many countries in recent years.

The nuclear industry, encompassing both designers/vendors and utilities, is working hard to improve the situation, and the viability for the nuclear power option. Specific goals have been formulated in the utility requirement documents that have been developed, eg. in the USA (by EPRI), in Europe (by the EUR group), in Japan and the Republic of Korea. The major focus of these documents is on operation and maintenance, but there are also requirements related to the initial cost, such as a ceiling for specific fore costs (1600 US$/kWe is typical figure, yielding a capital cost reduction of some 20% vs. the figures above) and a demand for shortened construction times (down to about 48 months). A brief review of utility requirements is provided in the "Status of advanced LWR designs, 1996" report [1] as background information with respect to design goals, and the plant design descriptions of the report present designer/vendor interpretations and design solutions adopted to meet the goals; the future will show whether these efforts will suffice to turn the tide.

3 DESIGN CHARACTERISTICS OF A WELL-PERFORMING PLANT

Trying to define a design configuration or structure that meets possible needs of a plant operator or owner would require very detailed assessments including cost-benefit analyses. The discussions of this paper will largely be limited to some of the characteristics described in the above-mentioned paper “Designing for nuclear power plant maintainability and operability" from the ANL TCM – with respect to possible application in new designs and modifications of existing plants.

The paper discusses design features of the ABB Atom BWR designs that contribute to the reliable operation and short refuelling outages that have been recorded; in the context of this paper focus will be on principles rather than detailed design features, however.

3.1 The reactor pressure vessel and ancillaries

The central - and largest - component in a BWR reactor plant is the reactor pressure vessel, which contains the reactor core and a number of reactor internals. The advanced ABB Atom BWRs are designed for "fast refuelling" - with no external pipe connections to the reactor vessel head. Hence, it can be removed directly when the flange bolts have been loosened by means of a "multi-stud tensioner" (an equipment that can handle a number of flange bolts [and nuts] simultaneously). The internals are stacked onto each other without bolt connections, and they can be lifted out directly when the vessel head has been removed.

The BWR 75 and its successors are equipped with internal recirculation pumps with pump impellers inside the RPV and wet motors in motor housings integrated with the RPV bottom. The pump deck at the reactor vessel bottom and the cylindrical support of the moderator tank are welded to the vessel. All other internals, including feedwater spargers and control rod guide tubes, are easily removable. This makes the whole inside of the reactor vessel accessible for In-Service Inspection (ISI) of welds.

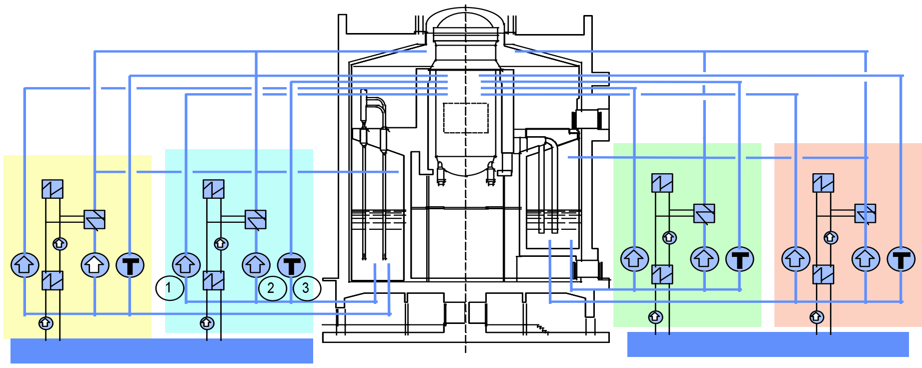
The arrangement of the internals can not easily be implemented in an existing reactor, but the multi-stud tensioner represents a practical alternative also for existing plants; it has been adopted by many vendors.

The reactor vessel bottom of the BWR 75 was modified compared with previous designs to reduce number of welds; in the new design the number of welds is reduced even further. This decreases the amount of ISI work that is required during a refuelling outage, i.e., it shortens the outage time with respect to inspections and reduces the associated radiation exposure.

An important advantage of the internal pumps with wet motors is the elimination of the shaft seals; another the elimination of external recirculation loop piping. Shaft seals require inspection and maintenance, and the piping calls for In-Service Inspection of welds; both activities that increase the workload close to radioactive sources and often imply extra critical path working time during maintenance periods.

The internal pumps and their static ”variable frequency-variable voltage” power supplies are advantageous with respect to operability; they permit rapid load following and power changes and with a suitable over-capacity in recirculation flow rate enable improving fuel cycle costs by ”spectral shift” operation towards the end of the operating cycle. The built-in over-capacity and the arrangement of the core inlet plenum serve another purpose, making it possible to continue power operation - at up to 100% power, if one of the pumps should fail; a positive feature for ensuring operability even though pumps and power supplies have proven to be very reliable.

In an ABB Atom BWR, the fine motion nut and screw control rod drives are hanging under the reactor vessel bottom. The control rod drives are, as the recirculation pumps, supplied with a purge flow of clean water to minimise contamination.



1: Low Pressure Coolant Injection
2: Containment Vessel Spray System
3: Auxiliary Feedwater System

Figure 4. Emergency core cooling systems

Wet motor pumps, without shaft seals, are used also for the reactor shutdown cooling system, reducing significantly inspection and maintenance needs; on the other hand, maintenance can be carried out rapidly and efficiently, using a special servicing equipment, when maintenance is needed.

3.2 Safety systems arrangement

In the BWR 75, the 2 × 100 % safety system configuration used earlier was replaced by a 4 × 50 % arrangement; i.e., the safety systems are generally divided into four separate subsystems, each with a 50 % capacity with respect to the design basis event (cf. figure 4). A strict physical separation of the four subsystems of each safety system was included. From the viewpoint of safety, availability, maintenance and cost 4 × 50 % represents an optimal solution - with enhanced redundancy and subsystem independence. The arrangement is in particular advantageous with respect to plant maintainability, since sufficient system capability/capacity will remain even if:

- one subsystem is taken out of service for repair or maintenance; and
- another subsystem fails upon request; the single failure criterion.

As a consequence, testing and preventive maintenance may be carried out during plant operation - on one subsystem at a time. This gives the plant management flexibility in planning the maintenance activities in the most efficient way; the workload during the annual refuelling and maintenance outage can be significantly reduced.

Another advantage relates to "permitted repair time". If a failure is detected in a 4 × 50 % safety system during normal plant operation, ample time is available for repair. As an example, the Technical Specifications for the Forsmark 3/ Oskarshamn 3 BWR 75 units permit a delay of up to one month for making the repair, without restrictions in plant operation. This means that the repair work can be carried out at a suitable time.

The BWR 75 design includes a certain redundancy also for non-safety-related systems, but primarily not for maintainability reasons. Typical examples in the turbine plant are the condensate and feedwater pumps and the circulating water pumps, with 3 × 50 % or 4 × 33 % pumps operating together on a common piping system, and providing a functional redundancy with respect to the power operation impact of a malfunction of an active component. (cf. Figure 5) Similar arrangements are utilised for service systems of the reactor plant, e.g., the primary and secondary cooling water systems, but there are also cases where 2 × 100 % have been found most beneficial, e.g., for the shutdown cooling system of the reactor.

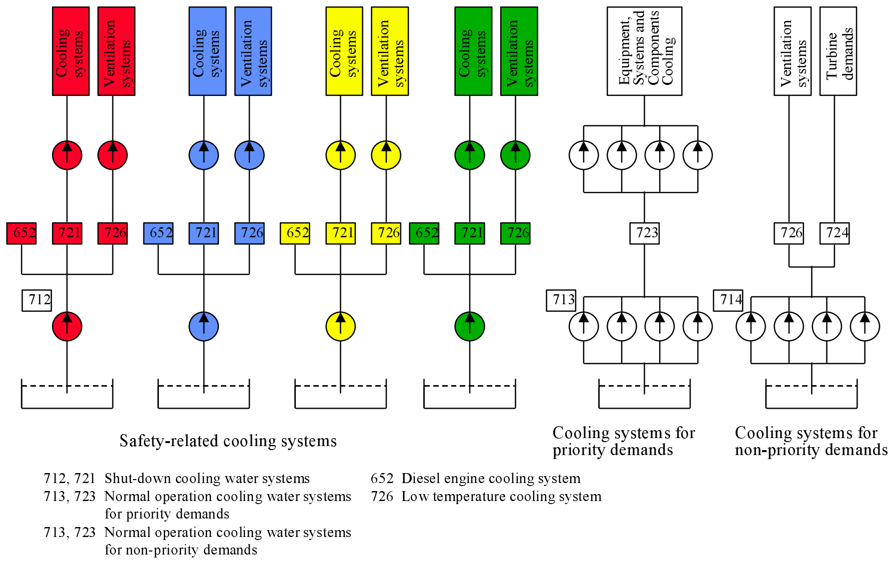


Figure 5. Simplified block diagram for cooling water systems in BWR 75

Anyhow, the main purpose of the functional redundancy in these systems is not to facilitate maintenance but to improve the functional reliability by making the power plant process less sensitive to component failures. The installed redundancy or over-capacity will involve an increase in the initial plant cost, but taking into account the potential gains in operational reliability, the effect on the total life-cycle cost will most likely be positive.

These ideas and principles can probably not be implemented fully in existing plants without significant costs and difficulties, but they deserve being taken into consideration.

3.3 Layout and installations

The requirements on physical separation of the subdivisions of safety systems strongly affect the layout and installation of components in the plant buildings. The design goal of high maintainability and low radiation exposure also has considerable influence.

In order to attain high maintainability the equipment should be installed in such a way that it is easily accessible for the maintenance personnel. Adequate communication routes should be provided throughout the plant - both for personnel and for transports of equipment. Space for performing hands-on operations, and for radiation shielding around components, is also important.

Some important ground rules and guidelines for planning of plant layout and equipment installation are:

- Non-radioactive systems and components should be separated from pipes and components containing radioactive substances, except when absolutely necessary for the function of the radioactive systems
- In order to reduce maintenance doses, large radioactive components like filters, heat exchangers, pumps, tanks, etc. should be either located in separate rooms or separated from each other by radiation shields.
- Each room containing radioactive system parts should have its own entrance from a corridor or communication area, where the radiation level is low. Corridors should be arranged so that they can be used by the maintenance workers for preparing operations in the radioactive environment and for use during short breaks or waiting periods.

- Each radioactive component must be easily accessible and have a suitable working space for maintenance and In-Service Inspection. The component orientation must as much as possible facilitate maintenance.
- Adequate space shall be provided for necessary handling equipment and in some cases for temporary radiation shields. Special tools, including remotely operated tools, may also require additional space.
- (From the point of view of radiology it is sometimes beneficial to remove the component and perform maintenance or repair in a better environment, e.g., in the active workshop. Therefore, it must be possible to remove and re-install the component rapidly.)
- Radioactive pipes should have as short lengths as possible and be located at a distance from other components. This is important, because in many rooms the main portion of the radiation emanates from the pipes.
- Electrical components should to the extent practically possible be located in separate rooms.

3.4 Materials selection and water chemistry

Proper materials selection is a key to the successful operation of a nuclear power plant. The best material is that which will be endurable, not crack, and not corrode significantly, i.e., that will suppress the need for maintenance and/or repair. Low content of metals that will become strong radiation sources, eg. cobalt, is of course also important.

3.5 Control equipment and power supply systems

Maintenance and operability aspects have also been taken into account in the design of the instrumentation and control and auxiliary power supply systems.

The design of the control equipment is governed by a Swedish safety requirement, in use since the early 1960s, that operator interventions shall not be needed within 30 minutes in the event of an accident with respect to ensuring nuclear safety. In addition, there are supplementary general design goals saying that a failure or malfunction of one component, or faulty calibration of a measuring channel should not cause a plant trip, and that the degree of automation for major process systems - with respect to keeping the plant in operation - should allow the operator some 10 minutes action time.

In addition to the measures intended to keep the plant in operation, there are also features aiming at reduced workload during maintenance and refuelling periods. Some specific features of the control equipment and power supply systems may illustrate these design efforts.

In the early plants, inputs to the reactor protection system (RPS) were taken from local limit switches, e.g., for temperature and pressure inside the containment. In the BWR 75, data are generated by transmitters located in instrumentation rooms outside the containment, which means that less manhours are needed for work inside the containment for checking and calibration of limit switches. The RPS input is generated by an electronic trip unit when the DC signal from the transmitter exceeds a set value; the calibration of the trip function is easily checked during plant operation.

Each of the parameters used for the RPS is supervised by four redundant measuring channels, and the process computer is routinely monitoring the proper function by comparing the signals of the four channels; a defective channel is detected rapidly and can then be disconnected and repaired at a convenient time. The RPS is built up with a general ”two-out-of-four” coincidence logic, and disconnecting one channel results in a transfer to ”two-out-of-three” which is a fully acceptable

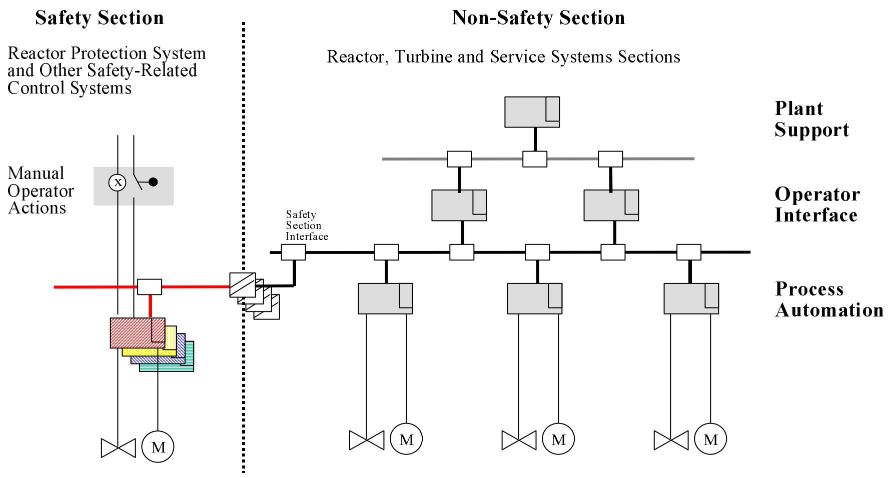


Figure 6. Integrated control system with separated safety and non-safety sections

situation from both safety and plant availability point of view. The RPS logic is built with electronic modules, without relays, which means that testing can be performed during operation, reducing the amount of work needed during the refuelling and maintenance periods.

In the new design, the control equipment for process communication, process monitoring and control, as well as for man-machine communication will largely be based on programmable equipment, including digital controllers for the major control systems. (cf. figure 6) Improved operator communication is one of the advantages offered by the new control equipment; human factors engineering can be incorporated in the design activities. Another advantage is the possibilities for introduction of new control functions without ”disturbing” the normal operation of the plant, and automatic test procedures for safety systems, etc. The new equipment is advantageous also in the longer term; when the ”computers” become ”obsolete” after some ten years, they can without too much impact on the plant be replaced by new generations, since most of the software can be maintained for the new equipment.

The division into four subdivisions (mainly of the safety-related portions) means that loss of one subdivision will have only limited effects on plant operation. Maintenance work can therefore to a great extent be performed at a convenient time during normal plant operation. Examples on such maintenance work are checking and tightening bolted joints in busbar systems, checking of relay protections, and testing of batteries.

With respect to the testing of batteries, it may be noted that a battery arrangement with two "50 %" halves, each provided with its own rectifier, was introduced in Forsmark 3/Oskarshamn 3. In this way, the batteries could be tested thoroughly during plant operation, including deep discharge tests, without affecting the plant operation. The only distinguishable effect is that the battery capacity, of the system and subdivision being tested, temporarily is reduced to about 50 %, to "1 hour" capacity. This arrangement was found well motivated for the Forsmark 3 and Oskarshamn 3 plants since they have DC distributions at 110 V, 48 V and 24 V, i.e., the number of batteries and distributions is large.

In the new design, there is distribution of battery-backed AC to local converters, instead of using central DC systems. The converters of the battery-backed AC system are provided with automatic switching devices for supply from the diesel-backed AC system instead which permits testing and maintenance to be carried out also during plant operation. This simplification and the reduced number of batteries, as well as DC distributions, yield a considerable reduction in maintenance work.

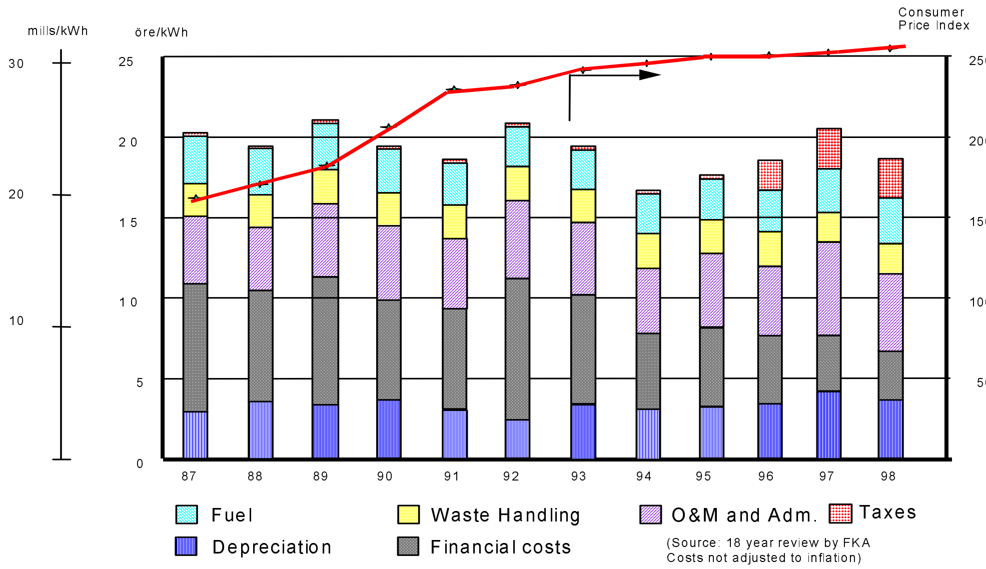


Figure 7. Power generation costs at Forsmark

4 ORGANISATION AND PROCEDURES OF PLANT OWNER/OPERATOR

Forsmark's Kraftgrupp operates three ABB Atom BWR units with a total electrical power of 3200 MWe. A competition-oriented culture has developed in the organisation from the start-up of the first unit in 1980; the listings in international media of capability factors have been a positive driver that has influenced the organisation's safety and economy. Capacity factors have in recent years been more than 90 %, reaching 93,3 % in 1998.

The need for efficient outages became obvious early, to improve both capacity factor and O&M economy. Outages have been under 20 days since 1983. There has been a full refuelling outage of only 10 days at Forsmark 1, and a less than 100 hour breaker-to-breaker outage at Forsmark 3 to change two fuel assemblies. This serves as indicators of the possibilities for reducing outages. The Forsmark achievements are illustrated by the diagram in Figure 7 over the power generation production costs during the last decade

The management of the three units is separated since 1982 and costs such as O&M are allocated directly, ie. it promotes competition between the units. The transition to deregulation has made it necessary to focus on the total production costs, and the company has developed into a "market-oriented economy". The electricity prices have dropped well below the predicted "bottom" level, and this obviously affects the owners and the power plant organisations.

Continuing to improve performance is a challenge, and Forsmark has identified five areas focusing on improvement and change. These are process performance improvement; balanced score card; plant renewal programme; competence development; and general cost control. Process performance improvement began in earnest in 1997 when a "strategic change focus" was defined. This required sustainable and competitive production; improved and measurable safety; world-class maintenance; and management focus on the main business. More than 100 people are now involved in process improvement teams. A "balanced score card" was developed for performance communication, with identification of critical figures setting of targets; one group of scores considers earning power; production; production cost; and net present value.

The technical condition of the plant is being developed in an R&D project in co-operation with Norwegian offshore industry and the University of Trondheim, Norway. Methods for “competence management” are under development; the programme begins by studying the work task based on the “business concept”, and explores the task in depth, down to “individual development talks”. As part of this programme, plant personnel is also given opportunities to visit other utilities and work with them for a certain period of time – to get ideas and inputs for the future and to improve the knowledge of foreign language and thinking. (Cf. the WANO programmes)

The Forsmark story illustrates the importance of management determination to have a well performing plant, and the importance of creating a competitive environment inside the organisation and “team” feelings. It also underlines the importance of long-term planning with respect to modernisation and renewals; the new year will see the start of a programme, which aims at establishing the needs for keeping the plant units in shape and operating economically for more than 40 years, to 2020 and beyond.

5 PREVENTIVE MAINTENANCE / REPAIR AND REPLACEMENTS

The merits of preventive maintenance are now well understood throughout the industry, but there is still need for more knowledge and information to enable proper “planning” of such activities, without too wide “safety margins”. Instrumentation systems for supervision of equipment status etc. may be an area for very cost-effective investments, when looking in a longer-term perspective.

Proper planning of maintenance activities should include assessment of real need, since “jobs not performed cause no doses and take no time.” This and other truisms were cited by David Miller from the University of Illinois in his presentation at the ANL meeting of an OECD study on work management practices that reduce dose and improve efficiency.

A specific concern in this context relates to the risk of failure of major components during an operation period; sometimes it could become very expensive to postpone replacement of essential components. The failure of some reactor internals in the middle of the operation season of a plant in Sweden resulted in an outage of several months. If the plant owner had decided to replace the components in a planned action earlier, the replacement could have been made at a marginal extension of a refuelling outage and at a much lower direct cost (possibly also at better quality).

Revitalisation of Quality Assurance (QA) is often needed for improving maintenance and repair activities. As noted by Frank Hawkins from U.S. DOE at the ANL meeting, we must remember that “QA is not perfection; it starts at the top but is everybody’s responsibility. It is not achieved by chance, and it can only be achieved by people, not programs.”

6 INTRODUCTION OF NEW TECHNOLOGIES AND NEW FUNCTIONALITIES

The introduction of programmable control equipment is quite natural for a new plant design, but it is of interest also for operating plants; significant modernisation programmes are under way at the nuclear power plants in both Finland and Sweden. It may be noted that these programmes do not involve an ”abrupt” change of all equipment on one occasion during a long plant shutdown, but are based on step-wise implementation during normal outages in accordance with a pre-determined plan - to achieve a desired final structure. This strategy is strongly recommended by ABB Atom, and it corresponds fully with the concluding recommendations of the IAEA-TECDOC-1016 on “Modernization of instrumentation and control” [2]. The utilities note that the increased computing capacities enable more efficient and flexible operation and saves money, e.g., by reducing start-up times after outages by some hours, and the impact of the investment on the life-cycle-cost is small and may even be positive.

The new technology is not applied only to I&C equipment proper; the improved computing capabilities have also resulted in a modified power supply for the internal recirculation pumps. The original design included a capacitor bank between rectifier and converter to provide an energy storage that could prevent too rapid runback of the pump speed. In the new design, a motor-driven flywheel provides energy storage for a pair of converters. The energy storage function is much improved and brings increased margins for the nuclear fuel in the core.

Adopting digital control equipment yields a number of advantages with respect to accuracy, flexibility and reliability. It also opens for introduction of new functionalities, in particular related to testing and supervision, and to recordings and reports.

One example of such applications was reported at the ANL meeting by Mark Bowman from Tennessee Valley Authority. He described a computer-based data acquisition system for diesel generators, which enables them to record and subsequently assess/verify the performance in an accurate and reliable manner with very little manpower needed. The Tech. Specs. for the TVA nuclear units requires that tests be performed at each start-up (after refuelling) to prove compliance with U.S. NRC Reg. Guide 1.9 [Susquehanna was built before 1.9 and has no reference to it], and verification was previously accomplished by evaluation of graphs from printers - with obvious limitations in accuracy. This testing and evaluation method was costly and time consuming; about 400 hours per outage. The new data acquisition system is a 2 MHz, 16 bits system with 40 times over-sampling, which uses ”raw” voltage and current waveforms; typical accuracy values are: ±0.15% in Vrms, ±0.25% in Irms, and ±0.045% in frequency. This way powers, speed, and recovery times can easily be deducted, and it has reduced work hours to less than 5 percent.

The new technologies also open for enhanced diagnostics systems, which can be simple or sophisticated. The management at Forsmark and other nuclear power plants in Sweden has indicated an interest in a “computer-based” start-up instructions “tool”, but not one limited to just an electronic presentation of the start-up procedures. Today, operators are relying on written “start-up procedures” and oral/written confirmation of established conditions. Start-up involves many steps and one point or two can easily be overlooked, but that is not too critical; if the operator jumps one or more pages, the situation may become quite critical. So, there is a wish for a system in which the start-up procedures are integrated with a data base incorporating component statuses, records of tests of safety systems etc; such a system would save a lot of time and man power for each plant start-up.

At the ANL meeting, Ralph Singer, ANL presented a model for power plant surveillance and fault detection, as well as applications to an LWR and a fast reactor. It is based on a non-linear state estimation technique coupled with a probabilistically-based statistical hypothesis test (checking in real time that all variables that in some way or other are ”connected” in the process, vary together in a reasonable way). Departures from correct operation can often be detected long before the deviation leads to a trip, i.e. the system can effectively help improve the availability of the plant.

There was also a paper by Dal Vernon Reising from University of Illinois, which described an approach to generating ”direct perception” displays to support operator actions during start-up etc. of complicated systems and processes. He claimed that such graphic representation of operating procedures and Tech. Specs. would make life easier for the operator while at the same time reducing the risk of erroneous actions, ie., it is a way to improve plant availability.

Michael DeVerno, AECL (from the small research centre in Fredericks) presented a model for equipment status monitoring - for plant configuration management during operation, an electronic unit that integrates operational flow sheets, equipment data bases, engineering and work management systems, and computerised procedures to assess, plan, execute, track and record changes to the plant's configuration. This system significantly simplifies the Operating Orders procedures, including controlling that erroneous actions are not taken, and that the final state reached is correct.

7 CONCLUSIONS

Energy production costs are becoming more and more important as the electricity markets are getting deregulated; all plants must get their costs under control, and improve as much as possible their operational flexibility and reliability.

In this context, experience from operating nuclear power plants represents important input to the design of new plants, but it also offers openings for re-evaluating system configurations and arrangements in existing plants, with respect to operation procedures, and to equipment installation and maintainability.

A number of ideas and principles used for new plant designs might also be applied, fully or partly, at existing plants. It seems that the plant owners must be prepared to look carefully at all options and not discard any alternative or idea without evaluating pros and cons.

If the nuclear industry can manage to develop mechanisms that take full advantage of all positive operation experience, this may well become the turning point for nuclear power and its future deployment.

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