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Advances in heavy water reactor technology

Proceedings of a Technical Committee meeting held in Mumbai, India, 29 January – 1 February 1996

INTERNATIONAL ATOMIC ENERGY AGENCY

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FOREWORD

Heavy water reactor (HWR) technology for electric power production has been developed by relatively few industrialized countries, notably Canada and Japan, but successfully implemented by developing countries, notably by Argentina, India, Pakistan, the Republic of Korea and Romania. In addition, construction of two HWRs is planned to begin in 1997 in China. Recent advances in HWRs include:

- development of new designs (for example the advanced CANDU-6 and CANDU-9 designs in Canada and the AHWR in India),
- improved pressure tube materials,
- improved in-service inspection techniques,
- improved fuelling machines and fuel handling equipment,
- validation of computer methods with experimental data,
- investigations of a variety of fuel options (for example slightly enriched uranium fuel and thorium based fuel).

Within the IAEA's nuclear power programme, activities in HWR technology development are conducted to foster information exchange among Member States with HWR power reactor programmes and to provide an objective source of information on HWR technology status and advances in HWRs that is available to all Member States. In order to document progress in HWR design and technology, the IAEA has published TECDOCs on advances in HWRs every few years. This is the third TECDOC in this series.

The technical officer responsible for the preparation of this TECDOC was J. Cleveland of the IAEA's Nuclear Power Technology Development Section, Division of Nuclear Power and the Fuel Cycle.

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SUMMARY

INTRODUCTION

At the beginning of 1996, worldwide there were 33 heavy water moderated reactors (HWRs)[1] with a total capacity of 18.6 GW(e) providing about 5.5% of the world's current nuclear electrical capacity, and 10 more HWRs were under construction[2]. HWR technology has proven to be economic, safe and reliable, and there is a mature infrastructure and regulatory base in several countries. The experience from development, licensing, construction, and, from the approximately 430 reactor-years of operation, forms a sound basis for the programmes underway in several countries to advance HWR technology. The very good neutron economy of HWRs allows fuelling with natural uranium thereby requiring no investment in uranium enrichment, while also providing flexibility for the use of different fuel options which would result in increased utilization of fissile materials.

The IAEA, following the recommendation of the International Working Group on Advanced Technologies for Water Cooled Reactors (IWGATWR), convened a Technical Committee meeting on Advances in Heavy Water Reactors, at Homi Bhabha Centre for Science Education, Mumbai, India, from 29 January to 1 February 1996. The meeting was organized in co-operation with the Bhabha Atomic Research Centre (BARC) and the Nuclear Power Corporation India Limited (NPCIL). The Technical Committee meeting focused on technologies aimed at improvements in currently operating HWRs, design and development for future plants, and development and evaluation of fuelling options.

This was the third Technical Committee meeting on Advances in Heavy Water Reactors. The previous meetings in this series were held in Toronto, Canada in 1993 [1] and in Montreal, Canada in 1988 [2].

This IAEA meeting addressed both the status of national programmes and technical topics including advances in plant and system design and new plant features, development of pressure tube technologies, fuel and fuel cycle options, computer code development and verification, and safety and accident analysis.

An important area of HWR technology which was not addressed at this Technical Committee meeting, but which was addressed in depth during an international conference in May 1996 organized by the Canadian Nuclear Society [3] is the topic of advances in fuelling machines and fuel handling equipment for heavy water reactors. This field is highly important because the high availability and capacity factors achieved by on-load refuelled HWRs are largely dependent on successful and consistent performance of on-power fuel handling systems. Therefore, this document includes an Appendix describing advances in design, manufacture, testing, operation and maintenance of fuelling machines and fuel handling equipment which is based in part on assimilation of presentations made at the 1st International Conference on CANDU Fuel Handling Systems. This Appendix is included with the permission of the Canadian Nuclear Society.

SUMMARY

Inaugural session

A. Sanatkumar, Director of Engineering, Nuclear Power Corporation of India Limited,

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1 The acronyms HWR and PHWR (pressurized heavy water reactor) are used interchangeably.

2 Subsequently, the Cernavoda Unit 1 plant achieved criticality in April 1996.

welcomed all participants to the meeting. A.N. Prasad, Director, Bhabha Atomic Research Centre, in his inaugural address, emphasized that HWRs have become a major line of reactors. He noted the high degree of safety of HWRs and the flexibility in regard to fuel options including the use of thorium which is planned within the Indian programme due to India's large indigenous natural resource of thorium. He impressed upon the delegates the importance of sharing experience to improve plant operation and to contribute to bringing about advancements in heavy water reactor technology.

In his introductory remarks J. Cleveland of the IAEA briefly described the activities of the IAEA in advanced technologies for water cooled reactors. He pointed out that a data base for advanced water cooled reactor materials has been established through a co-ordinated research programme, and that currently an activity is underway to establish a data base on thermohydraulic relationships. He also informed the delegates that actions are underway to form a new International Working Group (IWG) to advise the IAEA on co-operative activities on technology advances for heavy water reactors. Activities carried out within the frame of this new IWG would be initiated in 1997.

Y.S.R. Prasad, Managing Director, Nuclear Power Corporation of India Ltd, delivered the keynote address. He presented a summary of the Indian nuclear power programme briefly explaining its three stages starting with HWRs using natural uranium as fuel, building fast breeder reactors and advanced HWRs utilizing plutonium and thorium to convert it into ^{233}U and finally power reactors based on ^{233}U and thorium to establish a virtually unlimited source of power for generations to come. He said that the first stage has been achieved and then briefly discussed the challenges faced in establishing HWR technology in India. Some of these have been related to materials of construction for the end shield and coolant tubes, while the other challenges have arisen from emerging safety requirements including seismic qualification, diverse shut down systems, emergency core cooling, double containment and closed loop process water design to eliminate potential emissions to the environment. Challenges in the area of in-service inspection of coolant channels and issues arising due to hydriding and creep were also highlighted as were improvements which have been made in the areas of fuel performance and steam generator design for better ISI and maintenance.

Summary of national programmes

Argentina

Two HWRs (the Atucha-1 pressure vessel version and the Embalse pressure tube version) are operating in Argentina. An additional plant (Atucha-2) is under construction, but its completion as a NPP is in question due to lack of funding. NPP operation was segregated from CNEA in September 1994. Government funding through CNEA for R&D will continue. Development work includes slightly enriched uranium (SEU) fuel design, fabrication, and testing. In 1996 eighteen test SEU assemblies are undergoing irradiation in Atucha 1.

Canada

Twenty-one HWRs are operating in Canada. Worldwide five CANDU-6 units designed by Atomic Energy of Canada Ltd, are in operation and five more are under construction. The performance of CANDU-6 plants has been very good. In 1994 the Gentilly-2 (Canada) and Embalse (Argentina) CANDU-6 plants achieved the highest energy availability factors (98.3% and 97.8%, respectively) of the operating HWRs worldwide, and in 1993 the Wolsong 1 CANDU-6 plant in the Republic of Korea had the highest load factor (101.8%) of all power reactors worldwide.[3] Updating of 700 MW(e) CANDU-6 units is under way as a result of operational experience feedback and further R&D.

AECL's evolutionary advanced CANDU product is the 900 MW(e) class CANDU-9. The CANDU 9 design is based on the proven Darlington and Bruce B designs. The CANDU-9 improvements include improvements in plant layout and separation of safety from normal process systems (i.e. those systems used in the generation of electricity), a modern control room design employing state-of-the-art ergonomics, and distributed control systems. In 1996, the CANDU-9 design is under licensing review by Canadian regulatory authorities. Future advanced CANDU development include the 700 MW(e) advanced CANDU-6 and the larger 900-1300 MW(e) CANDU. The rated power level for the larger advanced CANDU is dependent on the number of fuel channels (480 channels in 900 MW(e) version; 640 channels in larger version) and the fuel used. Key areas in which technology development is under way to support these advanced designs are pressure tube technology, advanced fuel design and fuel options, heavy water and tritium management, control and instrumentation and construction techniques.

The developmental CANDU programme (called CANDU-X) is intended to be a large technological step to achieve competitiveness of nuclear electric power generation with electricity generation by natural gas turbine plants. This developmental CANDU should be ready for commercial service by the year 2020. Work on this reactor is mostly in the R&D stage and development efforts are examining potential new coolants for operation at high temperature and low pressure, construction of new materials for fuel channels which are compatible with the coolant and capable of withstanding the higher temperatures, and use of fuel with higher fissile content.

The CANDU Owners Group was formed in 1982 to include members from worldwide CANDU-owning utilities and Atomic Energy of Canada Ltd (AECL), the CANDU designer. COG provides a framework for co-operation in several programmes. COG is administered by Ontario Hydro. R&D funding for improvements in current plants comes mainly from COG. For advanced designs, funding comes from AECL. In 1992, COG organized a ten year strategic R&D plan divided into five major areas:

- safety and licensing,
- fuel channels,
- CANDU technology,
- waste management,
- radiological health & safety.

Canada, the Republic of Korea and the United States of America are collaborating on investigations of a fuel option involving the use of spent PWR fuel in CANDU, termed the DUPIC (direct use of PWR fuel In CANDU) fuel option.

China

China will need additional nuclear power capacity especially in its eastern coastal regions where energy demand is high but hydro potential and fossil fuels are not available in adequate quantities. While the PWR has been selected as the major type of nuclear power plant in China, construction of two CANDU type HWRs at the Qinshan site is planned.

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[3] Based on IAEA PRIS data.

India

In India eight HWRs are in operation, five under construction, and a few more are planned. The design of the larger 500 MW(e) HWR has been completed and construction of these units is also being planned. For current plants, India's specific development work focuses on improving fuel channel inspection techniques. For advanced designs, development is on-going with regard to containment systems, improved materials for pressure tubes, advanced I&C technologies, and passive safety features. India is also developing advanced fuel options for use in existing reactors with a view to exploiting its large, indigenous thorium resources.

Japan

In Japan the 165 MW(e) heavy water moderated, boiling light water cooled pressure tube Fugen HWR has successfully operated since 1979 as a part of its long term plutonium recycle strategy and to demonstrate technology in support of the Advanced Thermal Reactor Project (ATR), a demonstration 600 MW(e) plant. Considerable mixed U-Pu oxide (MOX) fuel assembly development and irradiation testing has been carried out. Fugen has achieved a cumulative load factor of 66.9% using more than 600 MOX fuel assemblies. Development activities in addition to fuel include pressure tube inspection devices, extensive thermal hydraulics tests on flow stability, CHF and heat transfer coefficient, severe accident code development, and core cooling during accidents using the moderator as a heat sink.

In 1995 the ATR project was cancelled due to concerns about its capital cost.

Republic of Korea

In the Republic of Korea, the Wolsong Unit 1 (CANDU-6) plant has been operated very successfully by the Korea Electric Power Corporation (KEPCO) since commercial operation began in 1983, achieving an average lifetime load factor of 85.9%. This outstanding performance is attributed to minimization of forced outages and short overhaul periods made possible by on-load refuelling. Wolsong 2, 3 and 4 (also CANDU-6 units) are under construction, and are scheduled to be connected to the grid in 1997, 1998 and 1999 respectively. While Wolsung-1 was a turn-key project supplied by AECL, the project management of the next 3 units is being done by KEPCO and about 65% of the equipment is being manufactured locally. KEPCO has a self reliant programme for PWRs but works in co-operation with AECL for HWR technology. AECL is the prime contractor for the NSSS for the Wolsong 2, 3 and 4 units.

According to the present long term power development plan, by 2006 one more HWR is scheduled to be operated. Further, KEPCO and AECL have signed a memorandum of agreement for CANDU export to third countries thereby establishing a framework for future commercial co-operation.

Pakistan

(While the national programme in Pakistan on HWRs was not presented at the TCM, some statements are included here for completeness.) The 137 MW(e) KANUPP HWR is operating in Pakistan and Pakistan's main interest in HWR technology derives from its objective to continue KANUPP operation. In its future nuclear power programme, Pakistan has no commitment to R&D on HWRs. However, the design improvements for future HWRs have been identified based on KANUPP experience and include improving accessibility to components, design of fuel channels for inspection/replacement, improved pressure tube material, and bi-directional fuelling.

Romania

The Romanian nuclear power programme started in 1980 with an ambitious plan for constructing 12 CANDU-6 units within a period of 15 years, and establishing an infrastructure to support the programme with maximum national participation. However, the programme was slowed down due to political changes and transition toward a free market economy, which resulted in a decrease in the industrial activities and power demand. Now the power demand is again showing an increasing trend. The present nuclear programme is mostly centred around the multi-unit station at Cernavoda. The first CANDU-6 unit at Cernavoda achieved criticality in April 1996. Due to economic and financial constraints the construction of three other CANDU-6 units at Cernavoda has been temporarily stopped. However, commissioning of Cernavoda unit 2 which is about 35% complete is predicted for around the year 2001. Domestic production of nuclear fuel bundles and heavy water, as well as research and engineering services will assure the support for operation of these units. A research and development plan is in place aiming at technology improvement, nuclear safety, radioactive wastes and spent fuel management.

Operating experience and service life improvements

The Fugen HWR was built for the purpose of developing and demonstrating the technology for the advanced thermal reactor (ATR) project with MOX fuel, D_2O moderator and boiling light water coolant. Fugen was commissioned in 1979 and most of the technological goals set for this prototype have been successfully achieved. Early in Fugen's operation, stress corrosion cracking occurred in some piping systems, and these were replaced resulting in lower capacity factors in the early years. Discounting this, the performance has been very satisfactory. A variety of fuels such as MOX, enriched UO_2, and MOX containing recovered spent MOX fuel have been successfully utilized by loading in the core in combination with other fuels. Significant core characteristics observed are: MOX fuel can be used almost in the same manner as UO_2; a flat power distribution can be obtained without insertion of control rods; and the void coefficient is nearly zero. High performance MOX fuel with burnup up to 40000 MW(d)/tonne has been tested. Tooling for in-service inspection (ISI) of pressure tubes has been developed and regularly used for periodic ISI. Post-irradiation examination of specimen pressure tube material has been carried out. Experience in radiation control, water chemistry control and D_2O upgrading has been very satisfactory.

The ATR project has been discontinued for economic reasons. Hence a new mission of the Fugen plant would be aimed towards development of more advanced MOX fuels, advanced technology for plant operation support systems, dose reduction, and techniques for diagnosis and predictions.

A programme to demonstrate use of SEU fuel in the Atucha-1 reactor is being carried out in Argentina. Various aspects such as economics of fuel fabrication, energy cost, spent fuel handling and storage cost, reactor physics and fuel management schemes, and safety related issues are being thoroughly studied. Eighteen fuel assemblies with 0.85 wt% ^{235}U enrichment were fabricated and introduced in three groups of six fuel assemblies each, but not exceeding twelve fuel assemblies at any time in the core. In 1995 the first group was taken out at an exit burn up of 10 Mwd/kg U. The reactivity gain, increase in channel power and other neutronic parameters were measured and were found to be in good agreement with calculated predictions. The performance of fuel was good with no failure occurring.

A new ferrographic technique for fault diagnosis through wear particle analysis has been developed and used in Indian HWRs. Wear particles generated in rotating and reciprocating machinery provide an indication of the wear process. A systematic examination of wear debris from oil samples gives clues about the condition of the machinery and an early warning of the impending failure. The mode of wear is revealed by size, shape, quantity, colour and composition of the particles. These characteristics can be measured by different instruments. The particle concentration for normally operated machines reaches a dynamic equilibrium after the initial break-in period is over. When the machine becomes aged, the pattern of small and large particles changes. An increase in both wear particle concentration and percentage of large particles is an indication of abnormal wear condition. Condition monitoring using wear particle analysis has been instituted for turbine generators, chillers, diesel generator sets, compressors, primary coolant pumps and boiler feed pumps in Indian NPPs, and data obtained from various instruments is being increasingly used for planning maintenance of these machines.

Technology development to increase the service life of CANDU pressure tubes is under way at AECL. The rate of elongation by creep and the growth induced by irradiation is slowed by about 30% by following a modified fabrication technique and heat treatment operation. Diametrical expansion is reduced by installing the tube with the back end (the end that exits the extrusion press last) at the inlet to match extrusion properties with the neutron flux profile. Crack initiation and propagation is controlled by limiting the total hydrogen concentration to absolute minimum values. The initial hydrogen concentration of the pressure tubes is reduced by identification of stages and locations where hydrogen enters zirconium during the pressure tube fabrication process and controlling it to a level of 5 ppm. This increases pressure tube lifetime by about 10 years. A layer of chromium at the rolled joint location would reduce deuterium ingress from end fitting to coolant tube. Use of hydrogen catchers like yttrium is also being studied. Higher fracture toughness has been obtained by minimizing trace elements (chlorine, and phosphorous) and by controlling carbon by quadruple melting and refining to prepare the material for fabrication.

A methodology has been developed to assess the service life of pressure tubes in Indian HWRs. The methodology consists of following steps:

1) Prediction of pressure tube–calandria tube contact due to creep deformation is done by a structural mechanics computer code SCAPCA which has been validated by measurements taken during ISI. A conservative assessment of creep contact time is made using this code.

2) Hydrogen pick-up is predicted by another code, HYCON-95, which is based on measured oxidation and hydrogen pick-up data of zircaloy-2 pressure tubes. This code has been validated by comparing predicted values with measured values from the Pickering and RAPS plants. This code will be further refined as more post irradiation examination data for Indian pressure tubes become available.

3) Growth of hydride blisters is modelled using computer codes (BLIST 1D and BLIST 2D) developed to predict the temperature fields in the neighborhood of calandria tube-pressure tube contact points and the diffusion of hydrogen towards the contact spots. These codes have been validated against published experimental results.

4) The limiting blister depth and the acceptable blister depth for Zircaloy-2 and Zr-Nb pressure tubes are determined on the basis of fracture toughness studies.

Fitness for service of a coolant channel with a pressure tube–calandria tube contact is assessed using data obtained during in-service inspection and the prediction of blister depth by the above mentioned codes. Assessment of leak before break capability is done by another code (CEAL) developed for this purpose.

Advances in plant and system design

Heavy water reactors (HWRs) have demonstrated their economic viability and very high level of safety in several countries. Evolutionary concepts for new power units (the advanced CANDU-6 and CANDU-9 plants in Canada and the AHWR in India) are based on well proven reactor technologies. Basic objective for these new designs include achieving a very high degree of safety and improved economics.

In all HWRs, safety is assured through the defence in depth approach (prevention–protection–mitigation), which builds on diversity and redundancy, taking advantage of the pressure tube system. Many of the new features in advanced HWR designs involve improvements in shut down systems, emergency core cooling systems, the containment system and heat removal systems. Two trends are noted in this direction: simplification of design and an increased use of passive safety systems. Both trends are expected to provide benefits of higher reliability and cost reduction. Other improvements are in the areas of operational simplicity, system and component reliability and longer lifetime, and use of advanced fuels.

In advanced CANDU designs, improved plant operation would be achieved in part through design of the control room to incorporate advancements in man–machine interface and human factors engineering with increased use of computerized information displays and automation of routine tasks. Safety will be enhanced by using the moderator and end shield cooling as reliable heat sinks in the event of a LOCA with loss of ECCS and by adding larger design margins in the pressurizer and emergency core cooling systems. Increased availability will be achieved by designing for planned replacement of certain components such as fuel channels. Better design quality and lower engineering costs will be achieved by using advanced engineering tools like 3 dimensional computer assisted design.

Design and development testing are proceeding on a calandria tube design that has a higher resistance to sag. One possible design uses a calandria tube with thick ends to optimize sag resistance without a severe burnup penalty.

A passive emergency heat sink concept for future CANDU plants involves an overhead pool of water to provide cooling water for various passive subsystems. This water pool is enclosed in a ring shaped tank placed at a high elevation in the containment. The pool is vented to the outside atmosphere. This pool is connected to the steam generator secondary side to the emergency core cooling system (ECCS) and to the moderator heat exchangers in such a way that water circulation will be established by thermo-syphoning without need of any pump. In case of a loss of coolant accident or a steam line break, the steam generators would be isolated from the secondary system and all the steam would be condensed in the water pool. The condensate would be returned to the steam generator by gravity. The containment is divided into an inner zone and an outer annulus zone and a natural air circulation flow would be established by which hot air will be cooled and heat rejected to the water pool. This flow would also improve hydrogen mitigation by appropriate location of catalytic recombiners. In the event of failure of ECCS the moderator will serve as a heat sink for the reactor and it will be in turn cooled by water from the water pool. This large pool of water can provide a passive heat sink for three days. Seismic isolation of the pool of water at high elevation will be addressed in the detailed design stage.

Simplifications have been made in the ECCS which would be incorporated into the advanced CANDU-6 and the CANDU-9 designs. The present CANDU-6 design employs a three stage system which includes:

1) light water injection by stored gas pressure at high pressure,

2) pumped injection of water at medium pressure and,

3) collection of water from the reactor building floor and pumping it back into core at low pressure.

This system requires a number of valve actions for successful operation and hence appropriate redundancy is provided to guard against failure of valve actuation. Regular on-power testing is also required to be carried out to ensure availability. In the ECCS for the advanced CANDU-9, isolation valves are replaced with a one-way rupture disc, and high pressure injection gas isolation is achieved by a floating ball seal in the water tank and medium pressure injection is eliminated.

A technology for removal of tritium from the moderator and from the heat transport system is being developed by AECL. This process is based on a combined electrolysis and catalytic exchange process. Design simplification and reduced operator dose are expected to result if tritium can be efficiently removed from the moderator and heat transport systems.

An advanced HWR is under development in India. This AHWR would use heavy water as moderator and light water as coolant. The main motivation for development of the AHWR is the plan to utilize thorium. Most of the energy would be produced from U^{233} bred in-site from thorium. Another key objective is to achieve a very high degree of safety.

Through elimination of the high pressure heavy water system in this AHWR, losses of D_2O will be significantly reduced. The engineering design will also achieve simplicity through use of conventional equipment in place of specialized nuclear qualified equipment, and minimization of site fabrication and installation efforts. The system will be designed to have a negative void coefficient. Also, emphasis will be given to use of passive safety systems relying on natural circulation, gravity driven water cooling of the core, and containment isolation by gravity driven water seal.

There is also research directed to give confidence that the improved systems will perform properly under all required conditions. For example, establishment of the optimal in-core configuration of the liquid poison shut-down system, and special testing programmes for key new components (e.g. one-way rupture disk, floating seal ball) for a simplified ECCS. There is also an effort directed at enhancement of the cold moderator system and calandria vault shield water system as a potential reliable heat sink for severe accidents.

Advances in Safety and Licensing

Because a nuclear power station involves large capital investment, the risk of delay due to licensing requirements is a key concern. Utilities would prefer to have an assurance that a reactor is capable of being licensed before entering into a financial commitment to purchase. In response to this, AECL has pioneered 'up front' licensing for new CANDU designs. In this approach the Atomic Energy Control Board (AECB) of Canada begins its review at an early stage of conceptual design. At the end of the process an accepted standard design is approved, providing an assurance to utilities and regulators of other countries that the design is licensable in Canada.

The Canadian licensing procedure places the fundamental responsibility for safety on the plant owner. AECB sets the safety objectives without prescribing design rules and leaves it to the designer to choose the means to achieve the safety objectives. The new licensing approach is a four step process. First, the documents to be submitted, reviewed and ultimately cleared are mutually discussed and agreed upon. Also in this first step, the licensing basis is defined by codes, standards, safety design guides and procedures to be adopted for the safety review. Next, a review of the technical description of the plant is conducted taking into consideration well established proven design features and concentrating on assessment of the new design features. Issues identified in the first two steps are analysed, and the safety report based on the conceptual design and the PSA report are reviewed and cleared. The final stage of acceptance of standard design is achieved after a thorough review of the detailed design and safety analysis. This final stage is equivalent to giving an operating license. The documents are thus generated and reviewed in a logical sequence of familiarization, requirement, design description and safety analysis.

In India, the licensing procedure by the Atomic Energy Regulatory Board (AERB) of India is initiated through preparation of a safety policy document defining required activities during each stage from site selection, design, manufacture, construction, commissioning, operation and decommissioning. Safety assessment criteria are set in these documents and the proof of their compliance is periodically recorded in a systematic manner. Furthermore, a system of assimilation of developments and experience elsewhere has been established. A beginning has been made in PSA/PRA studies. Operating procedures under emergency conditions have been compiled and are used by operators. Efforts are being made to establish symptom based emergency operating procedures. Regular site emergency drills are conducted and feedback sessions are held to identify and correct deficiencies. Quality assurance is conducted in an organized manner in all phases. The systematic efforts of ensuring operational safety have resulted in gradual reduction in safety related unusual occurrences and technical specification violations.

In the field of predictions of accident behaviour, analytical investigations of the response of Indian HWRs to prolonged deteriorated flow conditions have been carried out. Cases analysed covered:

1) Loss of coolant combined with unavailability of emergency cooling systems, with either
   - the pressure tube ballooning into contact with the calandria tube or
   - the pressure tube sagging into contact with the calandria tube;

2) Loss of coolant with residual coolant stratifying in the fuel channel.

The reactor is assumed to be tripped at the start of the transient, and the moderator cooling is assumed to be available. The calculations showed that for all cases, the heat rejection to the moderator was sufficient to keep the fuel from melting, even when the fuel cladding is completely oxidized by the exothermic zirconium–steam reaction. In the case of pressure tube sagging, large circumferencial temperature gradients in the tube are predicted, and this suggests a need for further analysis to determine whether the tube would survive the transient without failure. In all cases, the calandria tubes are predicted to remain cool and therefore are expected to maintain their integrity.

The influence of ^{239}Pu on void reactivity has been examined in Japan by both calculations and experiments. The experiments were performed in a heavy-water moderated critical assembly, with three different fuels, two different lattice pitches, and with either ordinary water or air as 'coolant'. The experiments showed that for all combinations of fuel and lattice pitch, the void reactivity is always positive. It is high at low enrichment (^{239}Pu + ^{235}U) and close to zero for the smaller lattice pitch and highest content of ^{239}Pu. The study showed that resonance absorption at 0.3 eV in ^{239}Pu is the main contributor to the reduction in void reactivity.

Advances in Fuel Options

AECL has studied a variety of fuel options for CANDU reactors to examine resource use and waste quantities. The resource utilization of 7 Mwd/kg NU of the present natural uranium (NU) fuel option can be improved to 11 Mwd/kg NU with the slightly enriched uranium (0.9 wt% ^{235}U) fuel option. Resource utilization can be further improved to 22 Mwd/kg NU by increasing the enrichment to 1.2 wt% ^{235}U. Use of slightly enriched uranium in HWRs in place of natural uranium can provide economic advantages in terms of higher burnup (almost double) rates and less handling of waste.

Thorium-based cycles in CANDU operate at near-breeder efficiency. They provide attractive options when used with natural uranium or separated plutonium as driver fuels.

Use of slightly enriched uranium in CANDU appears to be promising in the near future and thorium will provide an attractive alternative when uranium reserves become scarce in the more distant future. A number of other alternative fuel options have been studied. These options include use of spent fuel from LWRs in CANDU (DUPIC option), recycling of Pu and re-enriched uranium from LWRs, combination of uranium, plutonium and thorium in CANDU, and burning of actinides.

Argentina has studied different aspects related to the use of SEU fuel in its power plants. An improved fuel element design, keeping the general geometric configuration of the current one but using annular pellets in the outer ring of rods is shown to perform better in the extended burnup range. The startup of a new plant using the SEU cycle is analysed, showing that levelized costs as well as technical reasons favour a natural uranium initial core followed by a fast transition to the SEU fuel.

Improved utilization of uranium resources can be achieved by the re-use of spent fuel from LWRs in CANDU reactors. The LWR/CANDU tandem cycle leads to an additional 77% of energy through the use of reprocessed spent LWR fuel (1.6 wt% fissile content) in CANDU as compared with the energy obtained from the LWR. Dry reprocessing of LWR fuel followed by recycle in CANDU provides an additional 50% energy. Such recovered uranium (RU) from spent LWR fuel, which has a U^{235} content between 0.8 and 1.0 wt%, provides 15 Mwd/kg RU in a CANDU reactor. CANDU's relatively high neutron flux level, which is a result of its low fissile inventory, provides the possibility of converting the minor actinides and the long-lived fission products (such as ^{99}Tc and ^{129}I contained in spent fuel).

Advances in Computer Code Development and Validation

In Canada an activity is under way to validate the methodologies used for safety analyses for CANDU plants. The validation approach is broadly applicable to channel type HWRs. The disciplines treated are

- system thermalhydraulics,
- fuel and fuel channel thermal–mechanical behaviour,
- fission product release and transport,
- containment behaviour,
- physics (comprising reactor physics, shielding, and atmospheric dispersion), and
- moderator thermal–hydraulics.

Teams of specialists have been formed to develop validation matrices in the six disciplines. The validation methodology for system thermalhydraulics has been defined and is being reviewed by the Atomic Energy Control Board. The methodology is based on a validation matrix comprising two reference tables. The first identifies physical phenomena that might occur in design basis accidents, and the second identifies experimental and numerical tests that exhibit the physical phenomena. The validation programme is expected to span several years.

In India computer codes for predicting behaviour of the reactor containment under various accident scenarios have been developed. These codes have undergone thorough validation using data obtained from experimental test facilities, data from published literature and by participation in international standard problem exercises.

Conclusions and Recommendations

HWR technology for electric power generation has been developed by relatively few industrialized countries, notably Canada and Japan, but successfully implemented by developing countries, notably Argentina, India, the Republic of Korea, Pakistan and Romania. Further development of the technology is foreseen, especially in the following areas:

1) improved design (e.g. design for ease of replacement of equipment, design to achieve longer component lifetime, and design for improved access to equipment for inspection and maintenance);

2) improved safety and reliability;

3) improved economics (e.g. by the use of modular design and construction techniques to achieve a shorter construction schedule); and

4) development of alternative fuel options for improved use of fissile material.

Advances are being made in PHWR pressure tube technology. Current developments in in-service inspection techniques will allow more accurate evaluation of the "fit-for-service" condition of pressure tubes. Developments with regard to accurately placing the spacers between the pressure tube and the calandria tube will result in reducing the time required to replace pressure tubes. Improved fabrication techniques for Zr-2.5% Nb pressure tubes and methods for maintaining the hydrogen concentration of the pressure tubes at very low values are expected to increase their service life. This will allow the capacity factor benefit offered by on-line refuelling to be achieved to a greater extent.

Development efforts for advanced HWR plant designs mainly follow the evolutionary path. These include development of the advanced CANDU-6 and the CANDU-9 in Canada, which are incorporating improvements deriving from operating experience with existing plants, and the development in India of the AHWR which would utilize thorium based fuel and would also incorporate a number of passive safety systems. These design activities both in Canada and India focus on achieving improved economics and safety through improved construction methods, as well as improved operation and maintenance. Longer term development is carried out in Canada on the innovative CANDU-X design, with the goal of achieving considerably reduced capital cost through the higher thermodynamic efficiency attained by a direct cycle with higher core outlet temperatures (400–500°C) using low pressure coolants like organic or molten salt.

The high neutron efficiency of HWRs allows a wide variety of fuelling options including natural uranium, SEU, MOX, mixed plutonium-thorium oxide, U^{233}-thorium oxide and the DUPIC option. While by far the largest base of experience is with natural uranium fuel, significant experience has been gained with irradiation of MOX fuel bundles in the Fugen plant in Japan, and experience with SEU fuel is currently being accumulated through irradiation of SEU test assemblies in Atucha-1 in Argentina. Future development, test irradiations and implementation of the other options mentioned above could greatly extend fissile material resources.

A validation approach for safety analysis codes is being developed by the Canadians , and this approach is broadly applicable to channel-type HWRs. Thus the approach is of interest to designers/owners/operators/regulators of such reactors in other countries as well. This observation naturally led to the suggestion of participation by other countries in this work, perhaps including selection of benchmark problems that the international community could use for the validation of computer codes.

Passive safety systems have been proposed for future HWRs, but experimental demonstration of the phenomena and development of analytical simulation tools are required to thoroughly understand the system performance. This is a challenge that requires considerable investment in experimental facilities, expert manpower and in software development and validation. It may therefore be appropriate for countries with advanced HWR development programmes to pool their resources through international co-operation in these areas. In addition, international co-operation in the fields of pressure tube in-service inspection and maintenance, pressure tube replacement technology, life extension, standardization of neutronic data and standard problem exercises in reactor physics, thermohydraulics, and fuel channel thermo-mechanical effects could benefit all involved countries.

Finally, and of high importance, public acceptance of PHWR technology is of prime importance and public perception of safety based on accident free operation has to be maintained and strategies need to be evolved and standardized for high level waste management

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OPENING ADDRESSES

HEAVY WATER REACTOR TECHNOLOGY: THE NEED FOR INFORMATION EXCHANGE AND INCREASED INTERNATIONAL CO-OPERATION

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It gives me great pleasure to be here this morning for this inaugural function of the IAEA Technical Committee meeting on "Advances in Heavy Water Reactors".

Heavy water Reactors have established themselves as a viable technology. As you may be aware, worldwide, there are at present 33 heavy water reactors in operation with an installed capacity of 18.6 GWe accounting for nearly 5.5% of the world's current capacity of nuclear electricity generation and 14 more HWRs are under construction. While many other reactor systems have slowly given way to the dominant light water reactor technology, the Heavy Water system is marching on steadily.

There are strong techno-economic reasons for this. Heavy water reactors, besides being competitive electricity producers also offer significant flexibility for the use of different fuel cycles which can provide increased utilization of fissile material. They also possess considerable safety strengths superior to several other reactor systems. Perhaps, it is for this reason that the Heavy Water Reactor family has been relatively slow in bringing out safer advanced versions as has been the case with other reactor types. After all, necessity is the mother of all inventions. Nevertheless, we see that many technological advances at system, sub-system and component level have taken place in Heavy Water Reactors. Also advanced versions with considerable innovations to address current & future requirements are slowly emerging and I am certain that in the not too distant future Heavy Water Reactors would become dominant players in the arena of nuclear power by providing greater degree of safety and operational flexibility. The possibility of using the fissile materials released from dismantled nuclear war heads as well as the spent fuel discharged from light water reactors as fuel for the heavy water reactors would certainly enhance their value in the coming years apart from considerably easing the problems arising out of large stockpiles of these materials.

While PHWR technology has many favourable factors the reactor system is somewhat more complex with large number of pressure channels penetrating the reactor core. While many issues get resolved at the design stage itself, it is to be expected that some new issues always crop up at later stages. Expeditious resolution of such issues is important for the success of PHWR technology. It is rather unfortunate that information on PHWRs is rather limited in open literature as compared to the LWRs. We should therefore collectively work towards a regime where the technical information base is wider and such information can be exchanged freely to facilitate operation of PHWRs in an efficient and safe manner.

In India we have backed up our PHWR development with a strong R&D base provided by Bhabha Atomic Research Centre. As a result we have been able to find our own solutions for the problems that have cropped up in our reactors. We have developed sufficient insight into various PHWR issues to arrive at decisions and develop necessary technologies on our own.

We are also working towards the long term role of Heavy Water Technology in exploiting our vast thorium resources. I am glad to note that there are a number of papers in this meeting relating to enriched fuel cycles in PHWRs and also on ATR which has maximum experience on plutonium recycle. As many of you are probably aware, we have gone through several fuel cycle studies for our PHWRs and would be implementing plutonium recyle with thorium in our reactors at an appropriate time. In fact we have already successfully loaded thorium fuel for power flattening purposes in two of our PHWRs and would be working towards progressive increase in the use of thorium. Work on design and development of an Advanced Heavy Water Reactor is also progressing steadily. This reactor would generate large fraction of its energy from thorium fuel in addition to providing greater inherent safety.

We are interested in sharing our experience and expertise with other users of HWR technology. We are equally eager to learn from others' experience. We also view IAEA as an important platform to share such information. Recently representatives of countries with Heavy Water Reactor programmes unanimously recommended to the Agency to establish a new International Working Group on Heavy Water Reactors noting that several developing countries which have successfully established Heavy Water Reactor programmes can benefit from such an International cooperation. This could very well be justified on the grounds that while there are areas of commanality in technology with Light Water Reactors, there are significant differences in key areas including fuel channel technology, fuel cycle considerations, thermal hydraulics, severe accident sequences etc. The experience from design & development, construction and nearly 400 reactor-year operation of such systems could form the basis for such interaction. It is heartening to see that the Agency has responded positively in a relatively short time and is considering constituting an International Working Group for this purpose. Another useful mechanism to exchange experience in this area is through Coordinated Research Programmes of IAEA. We should together identify some useful CRPs which would benefit all of us. We are also open to other forms of mutually beneficial interactions.

Today, issues related to coolant channels and positive void coefficient are perhaps the most important items that need to be further addressed. While there are several technological fixes available, the relative cost benefits, particularly in the context of systems already operating, need to be fully understood. Then there are also issues relating to alternative fuel cycles. There are thus several opportunities to evolve common programmes.

In course of the deliberations of this TCM you will hear in great detail about the considerable amount of work done by this Department on passive double containment, vibration diagnostics, computerised I & C systems, in-service inspection, repair and life extension technologies and Advanced Heavy Water Reactor design. We are equally eager to learn about the advances made by the other participating countries in this field.

Technology is never static. Advances will continue to take place. Incorporating such advances in as many reactors as possible will decide the relative competitiveness of nuclear power technology. Developing world which is at present energy starved and is also faced with issues relating to environment protection as also financial resources, can benefit by adopting PHWR technology as it is better amenable to greater level of indigenisation in a developing economy as has been demonstrated by Indian and also some other examples. I hope this IAEA-TCM would contribute at least in a small measure towards this objective in addition to serving as a very useful information exchange meeting of benefit to all participants.

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THE INDIAN NUCLEAR POWER PROGRAMME: CHALLENGES IN PHWR TECHNOLOGY

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Indian nuclear power programme commenced with the construction of the Tarapur Atomic Power Station (TAPS-1&2) with 2X160 MWe [present capacity] Boiling Light Water reactors (BWRs), setup in 1969, on a turn-key basis, by General Electric Company, USA. These two Units were setup essentially to demonstrate the technical viability of operating them within the Indian regional electric grid system, which was at that time relatively small. These Units also helped us to gain valuable experience in operation and maintenance of nuclear power plants. After more than 25 years of safe and successful operation, these reactors are still in service, providing much needed electricity to the Western Grid.

The long-term strategy for development of nuclear power generation in India is based on a three-stage programme, formulated by Dr. H.J. Bhabha. This strategy takes into account and is optimally suited for

- achieving self reliance in nuclear technology,
- India's technological infrastructure,
- limited resources of Natural Uranium and abundant availability of Thorium within the country.

Stage 1 envisages construction of Natural Uranium, Heavy Water moderated, pressurized Heavy Water cooled reactors (PHWRs). Spent fuel from these reactors is reprocessed to obtain Plutonium.

Stage 2 envisages construction of Fast Breeder Reactors (FBRs) fuelled by Plutonium produced in Stage 1. These reactors would also breed U-233 from Thorium. It is also planned to develop an advanced heavy water thermal reactor (AHWR) as an extension of Stage 1 PHWR programme. The AHWR, using a Pu-239 enriched Uranium fuel in the driver (booster) zone and U-233 enriched Thorium fuel in the driven zone, would generate a large part of its energy output from Thorium through fission of in-insitu bred U-233.

Stage 3 would comprise power reactors using U-233/Th as fuel.

The first stage of India's nuclear power programme has matured to a great extent. The installed generating capacity of nuclear power stations in operation is 1940 Mwe, comprising of 320 MWe from the two BWRs in Tarapur (TAPS-1&2), and the rest from eight PHWRs [Table 1]. In addition, four PHWRs, each of 220 MWe capacity are under construction and are likely to become operational by 1998. Detailed designs for Advanced 500 MWe PHWRs, two Units of which are to be built in Tarapur (TAPP-3&4), have reached significant levels of completion; commencement of construction of these Units has been delayed due to financial constraints. Likewise, as show in Table 1, four more Units of 220 MWe each and six more Units of 500 MWe each, to be built at Kaiga and Rajasthan respectively are awaiting approval. Furthermore, eight PHWRs of the 500 MWe design are under planning stage. Keeping in view India's electricity demand, and various energy options, construction of two VVER Units of 1000 MWe capacity each is also in the planning stage. These will be built with Russian assistance.

A beginning has been made for the second stage of our nuclear power programme with the setting up of a 40 MWth Fast Breeder Test Reactor (FBTR) at Kalpakkam. This is being followed up with the proposal to set up the first 500 MWe Prototype Fast Breeder Reactor (PFBR).

Utilization of Thorium is the objective of the third stage; the first mile stones achieved in this regard are the development of chemical processing of irradiated Thorium to separate U-233, introduction of Thorium in PHWR for flux flattening, and setting up experimental critical facilities using U-233.

Nuclear Technology : Salient Issues

As noted earlier, the two BWR Units, TAPS-1&2, operated very successfully. At present, detailed plans for life extension of these reactors are being worked out. Innovative schemes are being evolved for the rehabilitation of Rajasthan Unit 1 (RAPS-1). Design and development of remotely operated tools are under way to rectify Heavy Water leak past a gasket in the Calandria Over-pressure Relief Device which is located in an inaccessible area. Considerable advances have been made in developing in-service inspection techniques for PHWRs, particularly for coolant channel inspection of RAPS-1&2 and MAPS-1&2. RAPS-2 is at present being prepared for en-masse coolant channel replacement. This opportunity is being utilized for incorporating several design upgrades to various systems with a view to improve performance. Both the Narora Units (NAPS-1&2) are back on line after extensive rehabilitation and upgrades carried out subsequent to the fire incident in NAPS 1 in March 1993. Kakrapar Units 1&2 (KAPS-1&2) are both connected to the grid and are performing well.

In running and developing our Nuclear Power Programme based on PHWRs, we have had to inevitably contend with a number of unique problems, some of which were a legacy inherited with the original design. Based on the extremely valuable experience gained by us in design, manufacture, construction, operation, maintenance, and safety regulation, we have been able to continually evolve, improve and refine the PHWR concept, and incorporate design changes in a progressive manner. All these have been achieved indigenously inspite of restrictions imposed on us on account of technology regime control. NAPS was our first opportunity to apply our operating experiences to the design keeping in view the evolving safety requirements, seismicity, ease of maintenance, in-service inspection needs, improved constructability, increased availability and standardization. Subsequent 220 MWe Units namely KAPS-1&2, Kaiga-1&2 and RAPS-3&4 are all similar to NAPS-1&2, but with certain modifications to reflect advances in technology and operation feedback from the previous reactors.

Some of these issues are discussed below, bringing out the problems, the challenges thrown by them and the solutions adopted.

1. End Shield Material

In the original design of the end shield (as adopted in RAPS-1&2 and MAPS-1, the material of construction was 3 1/2% Nickel Steel. At that time, based on available data, it was expected that the radiation embrittlement of this material due to the fast neutron irradiation would result in its nil ductility transition temperature (NDTT) to rise from its initial value of -101deg.C to 32 deg.C at the end of 30 years of reactor operation.

Subsequently, however, it was found that the NDTT of this material has crossed the operating temperature within a short period of operation. Thus these reactors are at present operating with calandria side tube sheet of the end shield in brittle condition. The suitability of these reactors to continue operation with these conditions of end shield, has been assessed in detail. It was concluded that the reactors can be operated safely considering that end shield is a low stress component and catastrophic failure of end shield during operation is unlikely since the end shield is a perforated structure and growing cracks will get arrested at the tube sheet holes. Suitable requirements have been introduced in the technical specifications for these reactors to ensure that during opration, the end shield temperatures are carefully controlled to reduce thermal stress and thermal shocks.

From MAPS-2 onwards, the end shield material has been changed to SS-304L which is immune to radiation embrittlement due to fast neutrons.

2. Earthquake Resistant Design

From NAPS onwards, the requirement to withstand seismic load necessitated certain design changes. In the earlier design, the main reactor components i.e. calandria and the two end shields were independently supported by means of support rods. This design was unsuitable for withstanding seismic conditions because of possibility of relative displacement during earthquake motion. In the modified design, these components are combined into a single integral assembly. The design was required to be optimized for conflicting requirements for earthquake forces and thermal loading.

3. Reactor Shutdown Systems

In the original design adopted in RAPS & MAPS, the reactor shutdown was achieved by fast dumping of moderator from calandria into the dump tank located underneath. The elimination of dump tank from NAPS onwards and the new safety requirement for having two independent fast acting shutdown systems led to the design of i) primary shutdown system (PSS) consisting of 14 mechanical shutoff rods & ii) Secondary Shutdown System (SSS) consisting of 12 liquid poison tubes. Because of the limitation in locating large number of devices in the space available in the calandria of 220 MWe, the reactivity worths achievable for each of these shutdown systems, though adequate for prompt shutdown for all postulated situations, requires further augmentation for long-term subcriticality to compensate for decay of xenon. For latter purpose, an automatic boron addition system has been adopted. In the newer reactors, (Kaiga, RAPP-3&4), this boron addition system has been further modified to be independent of the moderator system circulation.

The two shutdown systems adopted at NAPS, being the first of their kind, had initial troubles during the commissioning period. These troubles related to the drive mechanism of the primary shutdown system, and the solenoid valves in the secondary shutdown system, and were fully investigated and resolved before regular operation.

In the 500 MWe PHWR, there are two independent and diverse shutdown systems viz. shutoff rods (vertical) and Liquid Poison Injection System (horizontal). Both the systems have adequate reactivity depth and rate to shutdown the reactor rapidly and to keep the reactor in shutdown condition for a prolonged period. The incore assemblies of all the reactivity devices are designed to be easily replaceable.

4. Coolant Tubes and their Material

In the original design, the coolant tube material was cold worked Zircaloy-2. This material undergoes degradation due to hydrogen embrittlement, caused by pick up of hydrogen in reactor environment which precipitates as hydride. The life of the Zircaloy-2 tubes is severly restricted due to accelerated corrosion of this material after about 10 years of full power operation which leads to enhanced hydrogn pick up rates. Under these conditions the `leak-before break' criteria may not be met. This calls for en-masse replacement of the channels at this stage.

From KAPP-2 onwards, the coolant tube material was changed to Zr-2.5% Nb alloy which has low pick up rates of hydrogen. Manufacturing route for coolant tubes in this material has been optimised after an indepth evaluation of relevant fabrication dependent and performance related mechanical and metallurgical parameters and properties. This is a high strength material and required development of `Zero clearance rolled joints' to keep residual stresses to lower levels and prevent delayed hydride crackings. This material has been used in all reactors constructed later and will also be used in en-masse channel replacement of existing units after about 10 years of full power operations.

Design improvements have also been carried out to prevent harmful contact between pressure tubes & surrounding calandria tubes as discussed in next section.

An elaborate programme of preservice inspection, periodic inspection, inservice inspection (discussed later) and post irradiation examination has also been initiated and is now being implemented to monitor the health of coolant channels.

With above mentioned steps, coolant channels are expected to have a design life of more than 30 years and no prematuree en-masse replacement is envisaged during their design life in the case of our future projects.

One of the operational measures adopted with regard to pressure tubes in our PHWRs is prevention of their pressurization in cold condition. As the hydrides brittleness increases with decreasing temperature it becomes necessary to prevent application of high stresses in coolant tubes at lower temperatures. Thus a `hot pressurization' scheme has been evolved whereby, during the stage of reactor start-up from cold shutdown, the pressure and temprature are closely controlled in a systematic way so that higher stresses due to full primary system pressure are applied only at temperatures close to operating value.

5. Preventing & Detecting Contact between Pressure Tube & Calandria Tube

In PHWRs, the pressure tube (PT) containing fuel & hot pressurized coolant is separated from the calandria tubes (CT) (operating at ambient temperature), by garter spring (GS) spacers.

Contact between PT & CT needs to be prevented, as such contact leads to reduction in the local temperature in the contact region of the PT, making it susceptible to blister formation due to hydride precipitation and subsequent failure due to cracking. With GS's in place, as per design, contact between PT & CT is prevented throughout the life of the coolant channels (i.e. 20 years). However, if the GS's shift significantly from their design positions, the contact time can get reduced, thereby reducing the life of the coolant channel.

In the original design of GS's which is adopted upto KAPS-1, loose fitting GS's are used. During initial commissioning (hot conditioning) when the channels do not carry the fuel load, some of these GS's were found to move. Techniques were developed to detect the location of the GS's as well as to relocate them back to their design positions. This exercise was carried out for NAPS & KAPP-1 reactors before initial fuel loading. In subsequent reactors, i.e. KAPS-2 onwards, the design is changed to have tight fitting garter springs. In KAPS-2, it was found that these GS's did not move during commissioning.

Recently, based on inspections, it has been discovered that in RAPS/MAPS, the GS's can shift even during normal operations (perhaps due to vibrations caused during moderator dumping). This makes it difficult to accurately predict the PT-CT contact time. It is therefore necessary to have reliable means to identify contacting channels. For this purpose, two methods have been developed in BARC. The first is a non-intrusive screening technique, based on vibration measurement. The second method, used for channels identified as likely contacting ones by the first method, involves physical inspection through the use of a system developed in BARC called BARCIS. The non-intrusive vibration technique is under further development. In-service inspection procedures for coolant channels (discussed below) include monitoring the location of GS's. Where significant shifting of GS is confirmed after ISI, a tool (INGRESS) has been developed by BARC to relocate them, and thus extending the safe life of the channel. The relocation technique is based on flexing of the PT in order to unpinch the GS, and them moving it by a device using principle of linear induction motor.

6. In-service Inspection of Coolant Channels

The pressure tubes seen in the reactor environment are subject to irradiation enhanced creep, growth and corrosion leading to deuterium pick up (hydriding) and changes in mechanical (fracture resistance) properties.

A systematic periodic, in-service inspection (ISI) programme is executed for assessment of balance life of the pressure tubes. The ISI programme includes:

a) Measurement of dimensional changes of coolant channels i.e. axial elongation, sag & diameter.
b) Gap between PT/CT.

c) Eddy current/UT examination.
d) Scrap sampling on ID of tubes for hydrogen determination.
e) Non-intrusive vibration response technique for pre-assessment of contacting channels.
f) Garter springs positions.

Based on the above data and the data generated by post irradiation examination of the removed PTs for mechanical & metallurgical properties and theoretical assessment, a detailed evaluation of the balance life is done.

The conclusions of the findings/observations of above analysis programme for life extension of PTs are in following steps:-

a. Creep adjustment
b. Garter springs repositioning
c. Quarantining/Removal of contacting channels.
d. Batch replacement of contacting/predicted to be contacting
e. channels before next ISI campaign

7. Calandria Vault Design

In the original design (RAPS & MAPS) the calandria vault atmosphere is air. The vault is lined with a complex system of thermal shields which are cooled by air and water and the vault concrete is further cooled by embedded cooling coils. Great simplification in design has been achieved by incorporating a water filled calandria vault in NAPS and subsequent reactors. This has eliminated the thermal shield & embedded cooling coils; the production and routine release of radio active Argon-41 from the vault & thermal shield air has also been eliminated.

8. Emergency Core Cooling System

The Emergency Core Cooling System (ECCS) required for cooling the core in the postulated event of a Loss Of Coolant Accident (LOCA) as provided in the original design of RAPS & MAPS, is based on low pressure injection of water through moderator pumps. In this design the injection can start when PHT system pressure has fallen to around 5 Kg/cm2 (from the operating pressure of 87 Kg/cm2.

In line with the current practice of having a high pressure ECCS, in NAPS onwards, an improved ECCS, using accumulators is incorporated which provides a high-pressure injection of heavy water at 55 Kg/cm2 system pressure, followed by injection of light water initiated at 32 Kg/cm2 pressure, before low pressure long-term recirculation system comes into operation. The injection pressures and accumulator inventories are such that the PHT system continuously keeps receiving cold water till it switches on to the long term recirculation mode.

The ECCS of RAPS & MAPS is also being strengthened by retrofitting a high pressure accumulator injection system, in addition to the existing low pressure injection scheme.

9. Steam Generator Design

In RAPP/MAPP, there are eight steam generators each consisting of a number of small hair-pin type heat exchangers connected to a common steam drum. In NAPP, these are replaced with four steam generators of mushroom type with integral steam drum. These steam generators allow better in-service inspection & maintenance & repair and also use better corrosion resistant tube material (incoloy 800 instead of monel).

10. Containment System - Elimination of Dousing Feature

The RAPS containment system incorporates a dousing feature consisting of a large tank of water located in the dome region and a set of valves so arranged that on sensing of a LOCA or steamline break situation, a curtain of water is established in the path of escaping steam and thereby achieving pressure suppression in the containment. This feature has the risk of spurious actuation of dousing; such situation actually occured once at RAPS. Besides, this being an active system, involves considerable effort by way of maintenance and surveillance requirement to ensure continued level of required reliability. This feature of dousing has been replaced by a passive suppression pool system in subsequent plants starting with MAPS.

The existing dousing system at RAPS has a modulating feature whereby the dousing flow varies in proportion to the velocity of steam air mixture flowing during the accident. However, problem with this feature is, once the dousing is initiated, it increases air velocity in the vicinity which in turn gives signal for increased dousing flow. Thus leading to a divergent situation ending with the maximum dousing flow. The system is being now modified to have a optimized constant dousing flow which would cater for all situation requiring pressure suppression.

11. Fuel Design & Performance

The design of fuel bundle used in Indian PHWRs has undergone several advances over the years, one of them being introduction of graphite coating on the inner surface of fuel tubes. With the introduction of this feature, the capability of the fuel to withstand power ramping has increased significantly. The design improvements together with improvements in manufacturing quality control as well as operational practices to limit power ramping, have led to a dramatic improvement in fuel performance. This is demonstrated by current levels of Iodine in the primary coolant system which is typically in the region of 2-3 micro curie/lt compared with an order of magnitude higher values in earlier years.

12. Closed loop Process Water System

The design of process water system which was once through in Rajasthan Atomic Power Project, has been changed to closed loop. This change has fecilitated maintenance of appropriate water chemistry. Tritiated heavy water also will not go to cooling body (lake) in the event of failure of tubes.

Performance of Nuclear Power Stations in Operation in India

An analysis of the performance of the nuclear power stations in India and measures taken and also planned, to improve the performance are given below.

Performance Analysis

Units using mid 1960 design

TAPS, RAPS and MAPS fall under this category. While most of the components of TAPS were imported, a sizable percentage of equipment for RAPS and MAPS were manufactured in India between late sixties and mid seventies. The performance of these units in the past few years are as follows :

Unit	Construction Period		Capacity factor (%)		1994-95
	Start	Commercial	1985-93	1993-94	
TAPS-1	1964	October 1969	66	64	39#
TAPS-2	1964	October 1969	64	66	69
RAPS-2	Dec '67	April 1981	66	63	70 (upto31.7.94)
MAPS-1	Dec '67	January 1984	49*	19**	66**
MAPS-2	May '71	March 1986	47*	53**	65**

Extension of the refuelling outage for major repairs

* The availability factors of MAPS-1&2 during 1985-93 were 66% and 65%. The maximum operating power levels of these units are restricted to 175 MWe/unit since November 1989. MAPS units encountered repeated failures of the turbine blades.

** MAPS-1 was shutdown for a long period during 1993-94 for inspection and modification of turbine, considered essential based on Narora-1 turbine failure. In-service inspection of coolant channels and chemical decontamination were also carried out. In 1994-95 MAPS-1 and MAPS-2 achieved availability factors of 89% and 83%, respectively.

Units built to the designs of late 1970s to mid 1980s

NAPS and KAPS fall under this category. These unit are built to improved safety standards of fairly current international levels. However, equipment are mostly indigenous. While some improvements in the quality and reliability of Indian equipment have been made, the quality of turbine generators deteriorated as compared to MAPS. Also, even after some amount of derating, the design margins available are not adequate to operate at the maximum rated power level on a sustained basis. The performance of the power grid also continues to remain bad for the BHEL made turbines to operate in these grids. The performance of these units in the last few years are as follows :

Unit	Commercial	Capacity factor (%)			
		1991-92	1992-93	1993-94	1994-95
NAPS-1	Jan'91	26	54	Fire Incident	10
NAPS-2	July'92	-	45	-do-	40
KAPS-1	May'93	-	-	36	19
KAPS-2	Sept.95	-	-	-	-

Major improvements in design were made at NAPS compared with earlier stations. Narora is the first design developed indigenously for siting in moderately seismic sites. Considering the initial stabilisation period, performance in 1992-93 is fair.

KAPS-1 unit achieved a capacity factor of 36% in the first year. This unit commenced commercial operation in May 1993 at 75% full power level. Authorisation for full power was accorded in November 1993. The unit took an outage from 6th February 1994 for inspection and modification of the turbine. KAPS-2 has achieved first criticality in January 8, 1995 and was synchronised to the grid on March 4, 1995. After operating at power levels of 50%, 75% and 90% as approved by AERB, it has been declared commercial and has started commercial generation on 1.9.95.

Performance in the years 1993-94 and 1994-95 were largely affected due to Narora fire incident and consequent inspections of the turbine and modifications in MAPS & KAPS. The period of operation of NAPS & KAPS is too small to assess their performance. However, proper technical analysis and measures being taken should assure performance well above normative levels in future.

Performance Targets

Performance of the operating plants during the year 1995-96, and projections for 1996-97 & 1997-98, are as indicated below :

Unit	Target Capacity Factors (%) 1995-96(RE)	Actual Capacity Factors (%) during 1995-96 (Upto Sept.'95)	Target Capacity Factors (%) 1996-97	Target Capacity Factors (%) 1997-98
TAPS	58	77	58	63
MAPS	57	40	65	65
NAPS	63	72	63	65
KAPS	58	61	57	65
Overall	60	63	61	64

Measures taken / being taken to improve performance

a) Strengthening condition monitoring of equipment, outage management, preventive and predictive maintenance and spare parts planning.
b) Root-cause analysis and corrective action.
c) Inspection and modifications of BHEL supplied turbines to improve the performance.
d) Selection of improved version of turbo-generators for the future power stations, namely TAPP-3&4 and Kudankulam projects.
e) Efforts being made with CEA and Regional Electricity Boards, on improvement of performance of the power grid.
f) Strengthening training programme of operating and maintenance personnel, particularly with use of simulators.
g) Improvement in the organisational structure - Decentralisation of decision making to enable prompt action at operating stations.
h) Motivation, more authority and simplification of procedures.
i) Improving interaction with AERB.
j) With the above measures being implemented, NPCIL is planning to achieve progressively capacity factors of 60-65% in older units, 65-70% in NAPS, KAPS, KAIGA-1&2, RAPP-3&4 and 70% in 500 MWe units after initial stabilisation.

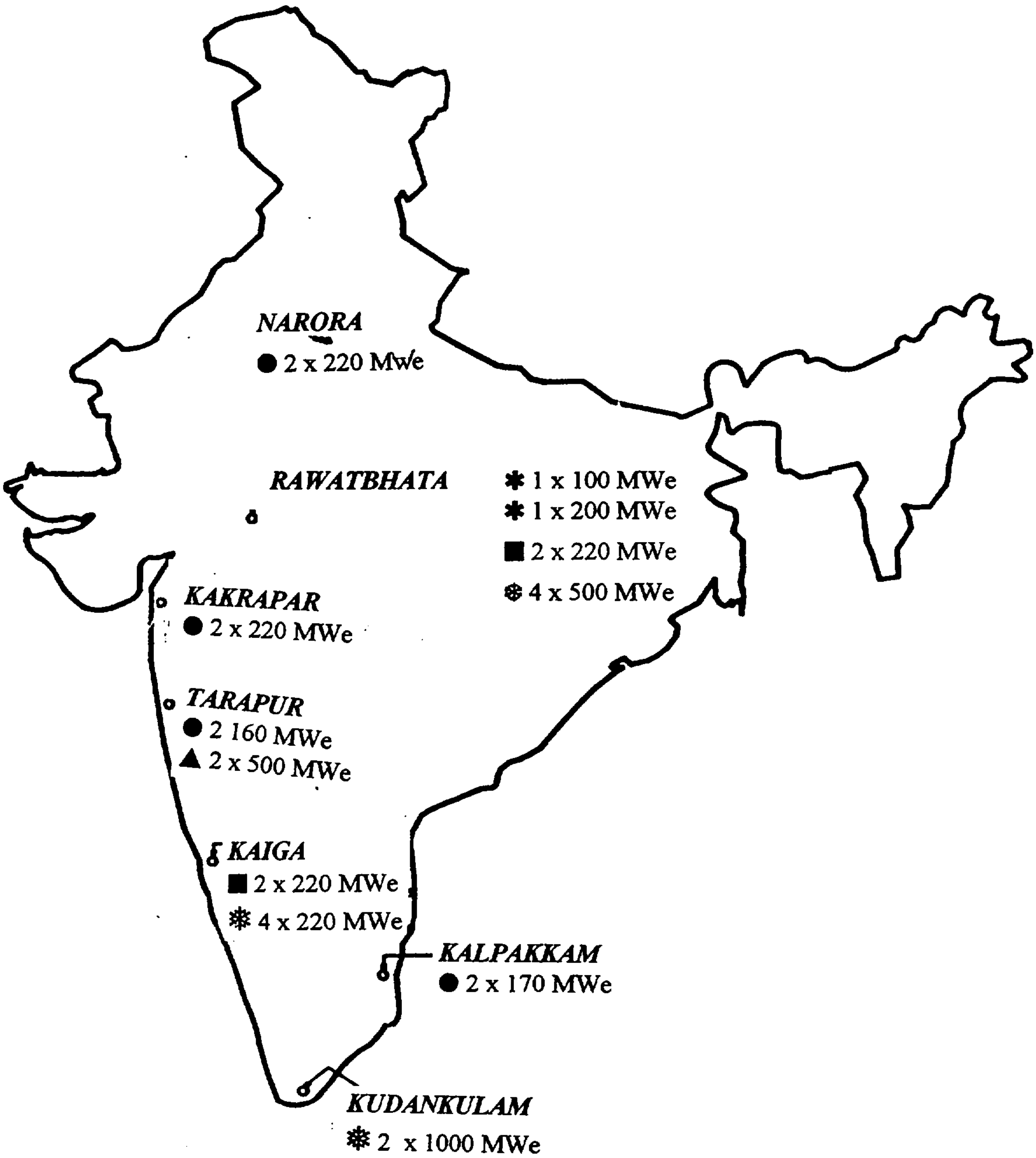
Based on the above analysis, performance of the NPCIL units in the past years can be considered as fair and with the measures being taken and planned, there is adequate confidence that all the units should be able to perform above normative level or better in the coming years.

INDIA's NUCLEAR POWER PROGRMME

•	STAGE - I 10,000 MWe through natural uranium fuelled PHWRs
•	STAGE - II Fas breeder reactors with plutonium as fuel and thorium as blanket for breeding U-233
•	STAGE - III Breeder reactors using U-233 as fuel and thorium as blanket
•	Short term goal is to complement generation of electricity at locations away from coal mines
•	India's long term nuclear energy policy is based on recycling nuclear fuel and harnessing the available thorium resource
•	India has sizable thorium reserves (about 360,000 Te)
•	Energy potential of 3 stage programme is placed more than 3 times the energy potential of coal reserves

NUCLEAR POWER PLANAS

● IN OPERATION
■ UNDER CONSTRUCTION
▲ SANCTIONED
❄ AWAITING SANCTION
✱ UNDER REFURBISHMENT



SUMMARY OF FIRST STAGE INDIAN NUCLEAR POWER PROGRAMME

DETAILS REACTOR UNITS	TOTAL CAPACITY (MWe)
OPERATING REACTORS 2 BWRs OF 160 MWe EACH, 8 PHWRs (1x100, 1x200, 2x170, 4X220 MWe)	1540 *
REACTORS UNDER CONSTRUCTION (4 PHWRs OF 220 MWe EACH)	880
REACTORS SANCTIONED & WORK TO COMMENCE (2 PHWRs OF 500 MWe EACH)	1000
REACTOR AWITING SANCTION 6 PHWRs, (2 x 500, 4 x 220 MWe)	1880
SUB TOTAL	5600
REACTORS IN PLANNING STAGE 8 PHWRs OF 500 MWe EACH, 2 x 1000 VVER	6000
TOTAL	11,600

* REACTOR 1 x 100 AND 1 x 200 ARE UNDER REFURBISHMENT

a:skj-A.DP MMP-4

NUCLEAR POWER CORPORATION OF INDIA LIMITED (NPCIL)
STATUS OF UNITS ON "GRID" AS ON 30/09/95

REACTOR CODE	REACTOR NAME	TYPE	CAPACITY MW(e) NET	CAPACITY MW(e) GROSS	OPERATOR	NSSS SUPPL.	CONSTRUNCTION START	FIRST CRITICALITY	GRID CONNECTION	COMMERCIAL OPERATION
IN-1	TARAPUR-1	BWR	150	160	NPCIL	GE	1OCT64	1FEB69	1APR69	28OCT69
IN-2	TARAPUR-2	BWR	150	160	NPCIL	GE	1OCT64	28FEB69	5MAY69	28OCT69
IN-3	RAJASTHAN-1	PHWR	90	100	NPCIL	AECL	1AUG65	11AUG72	30NOV72	16DEC73
IN-4	RAJASTHAN-2	PHWR	187	200	NPCIL	AECL/DAE	1APR68	8OCT80	1NOV80	1APR81
IN-5	KALPAKKAM-1	PHWR	155	170	NPCIL	DAE	1JAN71	2JUL83	23JUL83	27JAN84
IN-6	KALPAKKAM-2	PHWR	155	170	NPCIL	DAE	1OCT72	12AUG85	20SEP85	21MAR86
IN-7	NARORA-1	PHWR	202	220	NPCIL	DAE/NPCIL	1DEC76	12MAR89	29JUL89	1JAN91
IN-8	NARORA-2	PHWR	202	220	NPCIL	DAE/NPCIL	1NOV77	24OCT91	5JAN92	1JUL92
IN-9	KAKRAPAR-1	PHWR	202	220	NPCIL	DAE/NPCIL	1DEC84	3SEP92	24NOV92	6MAY93
IN-10	KAKRAPAR-2	PHWR	202	220	NPCIL	DAE/NPCIL	1APR85	1JAN95	1MAR95	1SEP95

RAJASTHAN 1 &2 ARE UNDER REFURBISHMENT

NUCLEAR POWER CORPORATION OF INDIA LIMITED (NPCIL)

STATUS OF UNITS "**PLANNED**" AS ON 30 SEPT.95

REACTOR		CODE	CAPACITY MW(e)		OPERATOR	NSSS SUPPL.
CODE	NAME		NET	GROSS		
IN-15	KAIGA-3	PHWR	202	220	NPCIL	NPCIL
IN-16	KAIGA-4	PHWR	202	220	NPCIL	NPCIL
IN-17	KAIGA-5	PHWR	202	220	NPCIL	NPCIL
IN-18	KAIGA-6	PHWR	202	220	NPCIL	NPCIL
IN-19	RAJASTHAN-5	PHWR	450	500	NPCIL	NPCIL
IN-20	RAJASTHAN-6	PHWR	450	500	NPCIL	NPCIL
IN-21	RAJASTHAN-7	PHWR	450	500	NPCIL	NPCIL
IN-22	RAJASTHAN-8	PHWR	450	500	NPCIL	NPCIL
IN-23	TARAPUR-3	PHWR	450	500	NPCIL	NPCIL
IN-24	TARAPUR-4	PHWR	450	500	NPCIL	NPCIL
IN-25	KUDANKULAM-1	VVER	920	1000	NPCIL	AEE
IN-26	KUDANKULAM-2	VVER	920	1000	NPCIL	AEE

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NATIONAL HWR PROGRAMMES

(Session 1)

Chairman

J. CLEVELAND
IAEA

XA9846592

THE AECL REACTOR DEVELOPMENT PROGRAMME

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Abstract

The modern CANDU-PHWR power reactor is the result of more than 50 years of evolutionary design development in Canada. It is one of only three commercially successful designs in the world to this date. The basis for future development is the CANDU 6 and CANDU 9 models. Four of the first type are operating and four more will go on line before the end of this decade. The CANDU 9 is a modernized single-unit version of the twelve large multi-unit plants operated by Ontario Hydro. All of these plants use proven technology which resulted from research, development, design, construction, and operating experience over the past 25 years. Looking forward another 25 years, AECL plans to retain all of the essential features that distinguish today's CANDU reactors (heavy water moderation, on-power fueling, simple bundle design, horizontal fuel channels, etc.). The end product of the planned 25-year development program is more than a specific design - it is a concept which embodies advanced features expected from ongoing R&D programs. To carry out this evolutionary work we have selected seven main areas for development: Safety Technology, Fuel & Fuel Cycles, Fuel Channels, Systems & Components, Heavy Water & Tritium, Information Technology, and Construction. There are three strategic measures of success for each of these work areas: improved economics, advanced fuel cycle utilization, and enhanced safety/plant robustness. The paper describes these work programs and the overall goals of each of them.

1. INTRODUCTION

The Canadian-originated CANDU® reactor system is now a mature and substantial contributor to the world's nuclear-electric energy system. It is fully competitive with other available reactor types with regard to economics, safety, and fuel utilization. The world, however, is constantly changing in its expectations of electricity generating systems. New competitors have eroded the small-reactor end of the market. The public is wary of any nuclear power system regardless of its technical merits. AECL is deeply involved in meeting these new challenges posed by changing economic, social, and regulatory conditions. Fortunately, the CANDU system offers a high degree of flexibility which allows the required design adaptations.

A competitive market in which the individual dollar commitment by a power utility is very large leads to a conservative buying policy. The owners are understandably reluctant to commit large-scale funds on unproved products. As a result the nuclear plant market is dominated by designs which are somewhat obsolescent; little opportunity is offered for technology advancement as the reactor vendors move from one project to the next.

It is obvious that AECL must evolve the CANDU plant design in full accordance with the customers' needs; this is a simple business criterion. Additional design flexibility sometimes will be accepted by a customer, especially if that customer and the local industry of the nation can take an active part in some aspect of the project. Thus, development opportunities, economics and social policy become entwined with the technological factors in successful major projects such as these. CANDU plants are competitive with others' nuclear plant offerings from all these points of view.

Recently, the rapidly-evolving technology of natural-gas turbines arrived on the scene and upset the carefully-crafted competitive schemes of nuclear vendors. At the present time it appears imperative to achieve a major reduction in nuclear plant costs. AECL is looking at this issue in an evolutionary manner in both the short and medium term. The result in the long term likely will seem revolutionary in the light of today's knowledge; nonetheless it will be achieved.

2. EVOLUTIONARY DEVELOPMENT

First, we can fix the starting and end points for the 25 year development program illustrated in Figure 1.

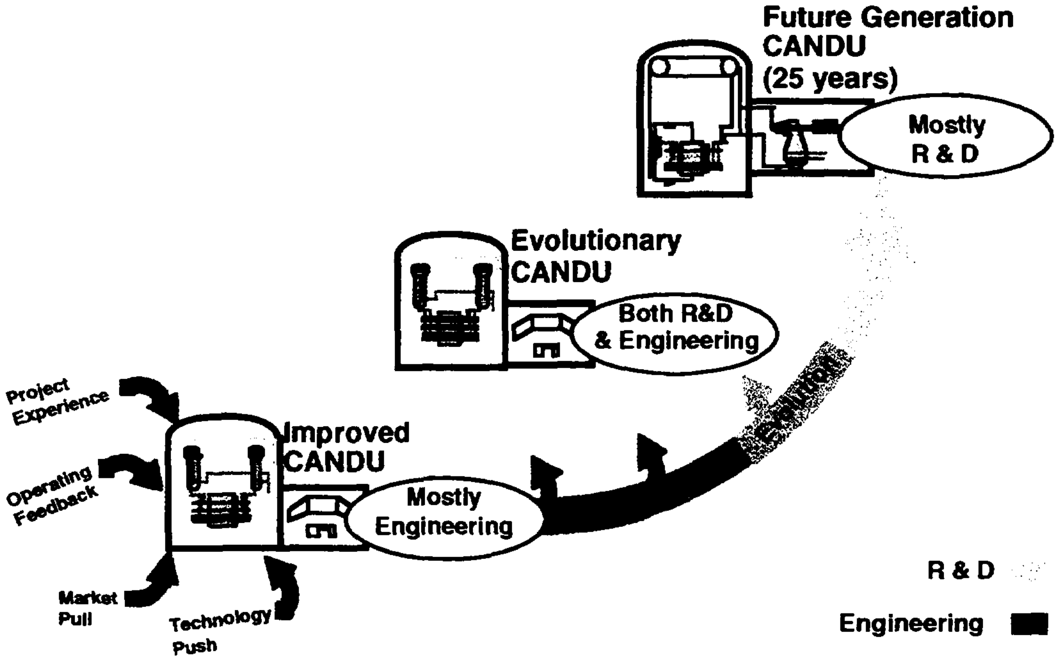

Figure 1: The Evolution of CANDU Reactors

Currently, AECL offers two products: the CANDU 6, of which there are four operating plants and four additional plants that will go critical before the end of the decade, and the CANDU 9, which is based on the twelve large multi-unit plants being operated by Ontario Hydro. These plants all use proven technology that was developed from R&D programs carried out in the past, mainly in Canada.

The 25 year development program retains all the essential features of CANDU reactors (heavy water moderation, on-power fueling, simple fuel bundle design, horizontal fuel channels, etc.). These are the features that make CANDU unique and enhance flexibility with respect to fuel cycles, safety, and cost. Within this framework, there are several opportunities for advancing the technology.

The end product of the 25 year program is not a specific design - it is a concept that includes the advanced features arising from our R&D programs. Moreover, since the approach to CANDU development is evolutionary, the final product will not suddenly just appear; rather, new features will be incorporated into our designs as the knowledge base is advanced. Also, as a result of the modular nature of the CANDU design, it will be possible to backfit many of these advances into existing reactors if there are sound economic or strategic reasons for doing so.

To carry out this evolutionary work, we have divided CANDU technology into 7 main areas: Safety Technology, Fuel & Fuel Cycles, Fuel Channels, Systems &

Components, Heavy Water & Tritium, Control & Instrumentation, and Constructability. There are three key strategic thrusts for these areas over the next 25 years: improved economics, fuel cycle flexibility, and enhanced safety/plant protection. Each of the 7 technology areas has specific goals associated with the three key strategic thrusts. In what follows, some aspects of these initiatives will be described.

3. ECONOMICS

Nuclear power plant economics depend on several factors, including initial capital cost, capacity factors, operating and maintenance costs, and output power.

While the overall unit energy costs averaged over the life of a nuclear plant are very competitive (owing to inexpensive fuel), nuclear plants have relatively higher initial capital costs when compared to, for example, natural gas plants. However, capital costs can be reduced significantly by improving components and by assembling the components faster. This, in fact, occurs for all high-technology products - at least, for those that are successful. For example, the first electronic devices used vacuum tubes, which required relatively large amounts of material to manufacture and high energy to operate. Vacuum tubes were replaced by discrete transistors that used much less material and energy. Discrete transistors were then replaced by integrated circuits that used even less material and energy. This was accomplished by constantly adding knowledge to the product - that is, knowledge replaced material and cost. The same process can be applied to nuclear plants, and every system and component in CANDU reactors is being assessed to do everything "faster, better, cheaper" using advanced knowledge.

Capacity factors, and operating and maintenance costs are affected by such determinants as operability of the plant, component lifetimes, and ease of maintenance. CANDU reactors already enjoy relatively high capacity factors, and occupy 7 of the top 25 lifetime capacity factor positions for operating plants above 300 MWe. Nevertheless, improvements are possible using more advanced information technology systems, reducing the complexity of systems, and incorporating long life components. CANDU fuel channels are a good example of how knowledge evolution has led to improvements. By understanding in some detail the various mechanisms affecting such phenomena as fracture toughness, creep and growth, and corrosion we have been able to make substantial improvements to fuel channels over the past decade. Current fuel channels will last much longer than those used in the earliest CANDU plants. They are defect free, are less affected by radiation, and contain smaller amounts of initial hydrogen that could adversely affect performance. By continuing to understand the basic phenomena affecting component lifetime and operability, and by developing advanced materials, we are confident that even greater gains can be made for most CANDU components.

Power increases can have a large beneficial effect on the unit cost of electricity, especially if they can be accomplished with relatively small changes in plant costs. One approach to increasing the power of CANDU reactors is to introduce Slightly Enriched Uranium (SEU) fuel containing 0.9 to 1.2% U-235. The SEU can be used to flatten the radial power distribution to produce about 15% more power, without changing the core design. Alternatively, owing to the modular nature of the core, it is possible to add more fuel channels. For example, the CANDU 9, a 925 MWe plant based on the large multi-unit stations used by Ontario Hydro, contains 480 fuel channels. The number of channels could be increased to 640 in a calandria vessel of essentially the same size, with an increase in power to 1275 MWe. In the longer term, it may be possible to operate the primary heat transport system at much higher temperatures (perhaps as high as 400-500° C, compared with the current 312° C) using organic coolants or molten salts, thereby substantially increasing the thermodynamic efficiency. Such a change would require considerable advances in our understanding of materials at these elevated temperatures under reactor conditions, but the efficiency gains could have a significant impact on unit energy costs.

4. ALTERNATE COOLANTS

Many designers have attempted to increase the primary coolant temperature in order to improve the overall system economics. With the possible exception of the LMR these attempts have been unsuccessful for one reason or the other. In the light of modern materials and chemistry knowledge, AECL intends to search over this field due to its very high potential payoff in plant capital cost. Using water as a coolant becomes less and less attractive as temperature increases due to the high pressures involved as well as the corrosive properties of high-temperature water.

The objective is to increase temperature and decrease pressure. At the same time the essential features of the CANDU concept will be retained. It is very likely that the fuel will be enriched slightly to compensate for additional absorption introduced by in-core materials required to tolerate high temperatures.

This is a research project and, as such, is not easy to quantify in hard schedules or plant data. It is uncertain in another sense -- it will, almost certainly, require some form of prototype prior to commercial introduction.

5. SUSTAINABILITY - LONG-TERM FUTURES

The high neutron efficiency, simple fuel bundle design, and on-line fueling, make these reactors particularly well suited to burning a variety of fuels, as illustrated in Figure 2. The use of natural uranium (0.7% U-235) fuel is a major attribute of CANDU reactors and has allowed countries adopting CANDU technology to manufacture fuel without dependence on a source of enriched uranium.

However, the use of slightly enriched uranium (SEU) has the potential to increase uranium utilization and to reduce fueling costs by about 30%, if 1.2% enriched SEU is used. Indeed, power increases and/or cost savings could also be effected by using a waste product from LWR fuel reprocessing - reprocessed or recovered uranium (RU), which contains 0.9% U-235. Currently, most RU is stored at the reprocessing plants awaiting final disposition by utilities that recycle the Pu from spent LWR fuel. While in principle the RU can be re-enriched and recycled in LWRs, almost twice the energy can be extracted by recycling RU in CANDU reactors.

All nuclear fuel ultimately comes from U-235. As uranium resources become more expensive (perhaps some time in the latter part of the next century), it will be necessary to ensure that we make optimal use of existing fuel resources. One way of doing this is to extract the maximum possible energy from spent LWR fuel. Spent LWR fuel contains about 1.5% fissile material (0.6% Pu-239 and 0.9% U-235),

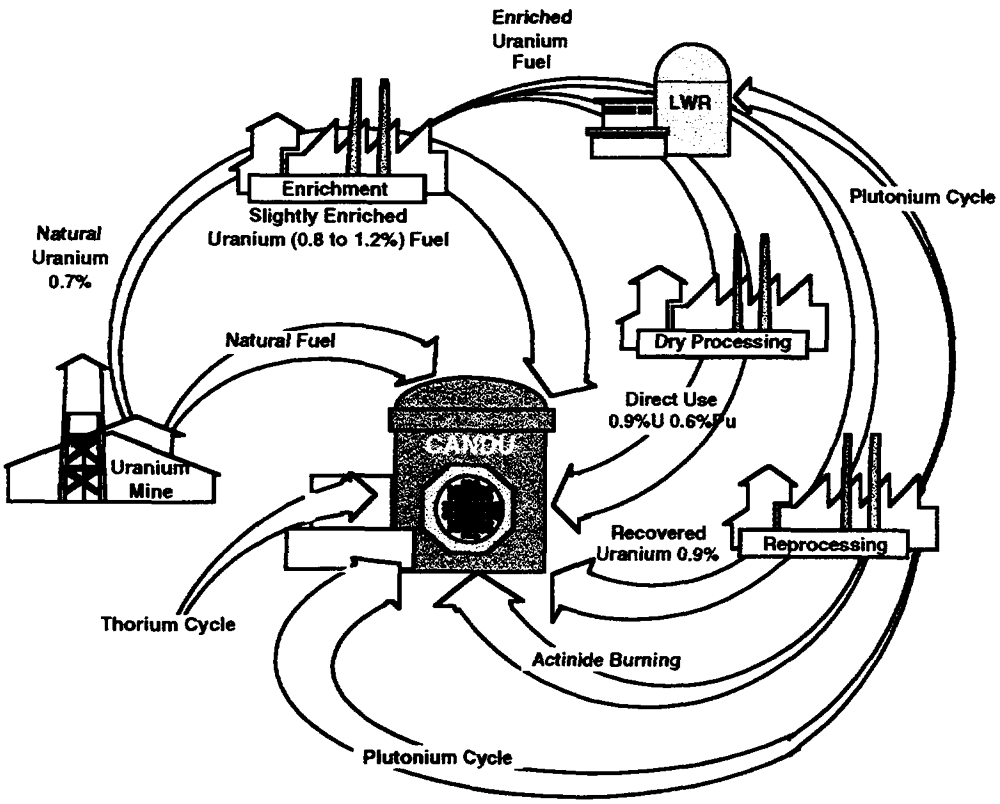

Figure 2 -- CANDU Fuel Cycles

compared with 0.7% fissile content in natural uranium fuel. Reprocessing plants can extract the Pu from the spent fuel, and the Pu is recycled back into LWRs. As discussed above, the RU with 0.9% U-235 is suitable for use in CANDU plants. However, both the Pu and U could be used as CANDU fuel by simply recycling LWR fuel without separating the Pu and U. Such processes are considered to be more proliferation resistant, since Pu is never produced in a pure state.

AECL and the Korean Atomic Energy Research Institute are currently working together to develop a viable process for achieving this, called DUPIC - Direct Use of Pressurized water reactor fuel In CANDU (note that there are two types of LWR: pressurized water reactors, PWR, and boiling water reactors, BWR; DUPIC contains the name "PWR" since these are the types of LWR operated by Korea). The process refabricates spent LWR fuel elements into CANDU fuel bundles without going through a wet chemistry separation process.

Even the actinide wastes with long half lives from reprocessing plants can be used as fuel in CANDU reactors. For example, if the actinides wastes were placed in an inert fuel matrix (i.e., a matrix without fertile or fissile materials), then the actinide waste would only need to provide about 5% of the fissile content of natural uranium fuel to produce energy. Therefore, some of the wastes from spent fuel are actually resources when LWR/CANDU synergism is considered.

In summary, CANDU reactors can be used to extend current supplies of fuel by adopting one or more of the above options. Figure 3 shows the improvement in uranium utilization using some of these options, relative to a reference pressurized water reactor.

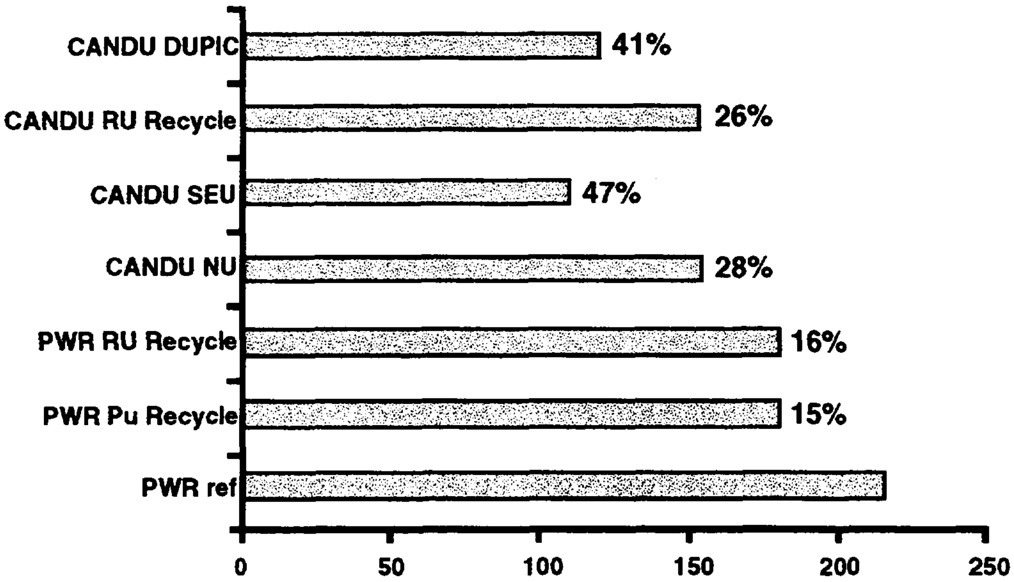

Figure 3: Specific Uranium Use (Mg/GW·a) (% Improvement over PWR ref.)

In the longer term, uranium resources could be extended by adopting thorium fuels for CANDU reactors. The world's resources of thorium are very large - perhaps 3 times those of uranium. India is a country rich in thorium. While thorium does not contain any fissionable material, thorium is a fertile material in which U-233 is produced in a reactor. There are a number of schemes for burning the U-233, including once-through cycles and reprocessing cycles. Using these cycles, CANDU reactors are assured of a fuel supply far into the future.

Another long-term possibility is the synergism between CANDU and Fast Breeder Reactors (FBR). FBRs are being considered by some countries as the long-term solution to dwindling uranium resources. However, FBRs are likely to be very expensive. If FBRs were used to fuel CANDU reactors, and the thorium cycle were adopted to extend fissile resources, then about 8-9 CANDU reactors could be supported for each FBR built. This would be the most economical approach to the introduction of FBR technology.

As the above discussion illustrates, CANDU reactors can not only be used to ensure that the maximum amount of energy is extracted from current fissile resources, but can also ensure security of fuel supply well into the future, even if uranium resources become scarce and/or expensive.

6. SAFETY AND PLANT ENHANCEMENTS

In CANDU reactors, the presence of the heavy water moderator surrounding the fuel channels effectively reduces the consequences of postulated severe accidents.

The reason for this is that if primary coolant is lost from the system, heat is transferred out of the fuel channel and into the moderator water. From the moderator, heat can be transferred to the environment via the normal moderator water cooling system. Even if the moderator were not available, the shield tank surrounding the calandria is capable of containing and maintaining a collapsed core in a cooled state. Therefore, in addition to the usual engineered safety systems in plants that meet international safety standards, CANDU reactors contain a number of passive safety systems that result from the inherent design of the reactor.

In the future, we can make even better use of the moderator system as a passive heat sink, so that in addition to safety considerations, CANDU reactors can be protected against severe damage that would impact on plant economics. New fuel channels are being developed that can transfer even larger amounts of heat to the moderator under loss of coolant conditions. New fuels are being developed that have high thermal conductivity, and that operate at lower temperatures. New moderator cooling systems are being developed for highly efficient heat removal based on natural circulation as opposed to forced convection using pumps. At the same time, such improvements could lead to ease of operation and lower capital costs due to the simplification/elimination of components. All these initiatives will build in an even higher degree of plant protection and safety based on both engineered and natural systems.

7. CONCLUSIONS

CANDU reactors will continue to evolve over the next 25 years, with an emphasis on cost efficiencies, fuel cycle flexibility, and enhanced safety/plant protection features. This will be done by building on the inherent characteristics of existing CANDU technology and by introducing well-proven new design features in a planned and orderly manner.

MAIN OBJECTIVES OF THE ROMANIAN HWR PROGRAMME

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Abstract

The strategy of the Romanian Electrical Power Authority (RENEL) is to finalize the nuclear power production capacities concentrated on Cernavoda Nuclear Power Plant complex and to assure the support services of associated nuclear fuel manufacturing facility (FCN) in Pitesti and heavy water production plant (ROMAG) in Drobeta-T.Severin. During 1995 an important progress was achieved in meeting short-term objective-to commission Cernavoda NPP Unit No.1. There is a comprehensive action plan involving specialized companies supporting engineering, research and development activities following three main lines: CANDU technology improvement, Nuclear safety and environmental protection and Radioactive wastes and spent fuel management.

1. INTRODUCTION

The decision to implement nuclear power in Romania was adopted basing it on overall optimisation of predicted power programme including power units and associated manufacturing units to assure special materials-nuclear fuel and heavy water [1,2].

The decision to build first Nuclear Power Plant (NPP) at Cernavoda site was taken in 1978 after more than 10 years of analysis, studies and negotiations with potential foreign partners. The Cernavoda NPP was designed to consist of five units with an installed capacity of 700 MWe each.

Its technology is based on the "CANDU-6" type nuclear reactor developed by Atomic Energy of Canada Ltd (AECL).

The construction works at Unit No.1 started in 1980. For the following units the work started in a period of about one year interval for each next unit related to the previous one. Before 1989 a large National Nuclear Program (NNP) was planned to be implemented in Romania, in the centralised economic system. More than 12 CANDU-6 units had to be installed within 15 years period and a tremendous effort was spent to build a national infrastructure to maximize the national participation for this program. In RENEL there is also an industrial support structure for the NNP represented by the Nuclear Fuel Plant (FCN) located in Pitesti and Heavy Water Plant (ROMAG) located near Drobeta-T.Severin. Both manufacturing units are based on technologies developed in Romania.

Moreover, the Romanian industry for the supply of equipment and special materials to nuclear power units was the beneficiary of a transfer of know-how for components employed in nuclear power plants, such as calandria vessel, steam generators, fuel channel elements, pumps, valves, electrical equipment etc.

There are also specialized institutes: The Nuclear Research Institute (ICN) in Pitesti dedicated to specific research and development activities and Center of Technology and Engineering for Nuclear Projects (CITON) in Bucharest-Magurele for design and engineering activities. Both these institutes in tight cooperation assure the implementation of RENEL development plan supporting nuclear power programme.

The December 1989 revolution in Romania has brought fundamental changes in the political, economical and social life of the country. The difficulties of this transition period toward free market economy have led to the decrease of the industrial activities and, as a consequence, to a reduction of the electrical power demand.

The national power demand decreased from 83.59 TWh in 1989 to 55.77 Twh in 1994, representing a 33.3% decrease [3]. At the same time the internal power supply capability has decreased due to: existence of a high wear thermal power plants park, low performance of lignite-fire power plants, and lack of conventional fuel sources (especially oil and natural gases). The difference between power demand and internal power supply, as imported power, was about 1.3% of the demand during 1994 year.

The forecasts predict an increase of the power demand in 1995/1996 and the demanded level of 1989 is going to be reached again in 2000-2005.

Within this context it was necessary to review the future national power development programme and to introduce accordingly the necessary adjustments in the nuclear power programme.

2. RECONFIRMATION OF NUCLEAR POWER POSITION

During 1993-1995 special studies have been performed in order to determine the possible evolutions of the energy and electrical power demand for the coming years under different socioeconomic and technical development scenarios for the country [4,5]. The main objective of these studies was the economic optimization of the Romanian power generating system expansion and the role that nuclear power units might play in meeting the future energy requirements. In keeping with the forecast electricity demand, beside the actions imposed by short-term strategy guidelines, such as energy preservation and rehabilitation of the present conventional power plants in operation, new power capabilities are requested for generating electrical power [6].

It is worth to mention that the first additional power capacities preferable in 1995-2000 year interval in all optimal expansion plans of national power system are the nuclear power units No.1 and No.2 from Cernavoda NPP. In this way the opportunity to complete the works and to commission these two units has been demonstrated. Implementation of the last three units in Cernavoda NPP is provided for a much later period of 2005-2015 year interval.

According to these studies the updated nuclear power programme has the following phases:

- short-term 1995-1996 period: the main milestones include Cernavoda NPP-unit No 1 commissioning, feasibility studies to resume the construction and installation work for Unit No 2, finalization of retrofitting work and qualification as CANDU nuclear fuel supplier for fuel plant FCN in Pitesti and extension and commissioning of two additional modules at heavy water plant ROMAG;

- medium-term 1996-2000 period: the main milestones include finalization of construction work and commissioning of Cernavoda NPP Unit No 2, production of fuel bundles and heavy water for the first two units in Cernavoda NPP, provision of support engineering services for optimum operation of nuclear units;
- long-term forecast (after 2000) is focused on research, development, engineering and construction works for Cernavoda NPP Units No 3,4 and 5 implementation, correlated with technological improvements. These units can be regarded as power units which designs are still under development and constitute long-term option with potential important improvements.

3. STATUS OF CERNAVODA NPP AND SUPPORTING SPECIAL PRODUCTION UNITS

3.1. Cernavoda NPP.

After December 1989 essential changes in the concept of the project progress management have been done. Starting with 1990 the application of an exhaustive inspection, verification and repair programme on the works carried out up to that moment has begun. The Romanian government has issued the decision (July 1990) regarding the approval of updating the technical-economical study on the Cernavoda NPP. In October 1990 the IAEA was asked by the Romanian authorities to review the project. The IAEA PRE-OSART report done for Cernavoda refers to the need for cultural changes to facilitate the safe operation of the station. The key recommendations were to give more responsibility and financial control to the owner of the plant, to reduce interference from various government ministries, to implement a proper project management and to enhance expert assistance from outside Romania. An important moment was in August 1991 when RENEL signed a Project Management Contract (PMC) with AECL (Canada) - Ansaldo (Italy) Consortium (AAC) stipulating that the latter Consortium takes over the management activities which span completion, commissioning and connecting to the national power grid of Unit No 1, as well as its initial operation for up to 18 months. Staff training also entered under the responsibility of AAC. For Cernavoda NPP Unit No 1 the construction work for the buildings and main systems has been practically completed.

The revised level 1 schedule shows the following key dates [7]: physical completion of systems - Dec, 1995, criticality-begining of 1996, first synchronization to the grid - March, 1996.

For the last quarter of 1995 it was estimated that the critical path passes through a completion of commissioning checks and tests for a group of some twenty systems (e.g. moderator cover gas, emergency water supply, liquid zone control, failed fuel location, fixed area gamma monitoring, liquid injection shut-down, raw service water, etc). The productions of Commissioning Completion Assurance Reports are also on this critical path.

During 1995 the first inventory of heavy water was supplied on site, partly from domestic source (ROMAG) and party leased from

Canada. The first nuclear fuel load was supplied from the canadian company Zircatec Production Industries.

A special attention was given to timely resolution of licensing issues not to become a major constraint affecting the schedule. In this respect the licensing schedule and practice is following the status of the design modifications implementation and their impact on the plant safety, as a prerequisite for various licensing milestones [8]. The current regulatory practice revealed no major impediments in the Cernavoda NPP-Unit 1 licensing process [9].

The reactor building leak rate test was completed successfully. A measured leak rate of 0.3% of building free volume in 24 hours at 124 KPa maximum pressure was achieved. A final report is being produced for submission to National Commission for Nuclear Activities Control (CNCAN) for approval. The first group of expatriates shift supervisors and control room operators were authorized by the licensing authority-CNCAN.

The staff training and simulation center built on Cernavoda site is close to being commissioned. The center is outfitted with a full-scope simulator for the training of operators.

During all phases of Unit No 1 commissioning a separate plan was settled to follow the fulfilment of IAEA PRE-OSART mission recommendations, which took place in April/May 1993.

RENEL is already taking part in international information exchange programs run by CANDU Owner Group (COG) and World Association of Nuclear Operators (WANO). After starting operation of Unit 1 RENEL will become an active participant in the other COG and WANO programs as well.

For the other Cernavoda NPP units the construction work has been stopped and they all are now in preservation state. Today the status of execution of project for these units is as follows: U_2 = 35% (24% construction and 47% procurment); U_3 = 12%; U_4 = 5% and U_5 =3%. As it was mentioned earlier the completion of all power units in Cernavoda NPP is provided as future investment in long-term development strategy. A joint AAC/RENEL implementation team has been set-up to develop the necessary studies required for the implementation of a Project Financing Scheme for the completion of Cernavoda Unit No 2. This team has prepared a viable financing proposal which is to be completed at the beginning of 1996.

We consider that the successful completion and commissioning of the first unit has to be an impulse for the second one. A recent economic efficiency study emphasised the opportunity to produce additional 700MWe in nuclear power Unit No 2 comparing with either rehabilitation or completion works of other fossil fuelled power plants, independently of the load demand rate of growth [10]. To finalize Cernavoda NPP-Unit No 2 a capital investment about 725 million USD is needed.The cost of delivered electricity by this nuclear unit will be less than 45 USD/MWh. Estimated duration to complete this unit is 58 months.

3.2 Heavy water Plant

ROMAG, the Romanian heavy water plant consists of four production modules, 90 tonnes per year each. The manufacturing technology was developed by Romanian specialists based on the process of isotopic exchange between water and hydrogen sulfide, followed by a final stage of vacuum distillation.

The first two modules are now in operation. For Cernavoda Unit No.1 more than 165 tonnes of heavy water were produced by this plant, the balance up to 500 tonnes was leased from Canada. The last two production modules under construction will be commissioned in 1996-1997 correlated with Cernavoda NPP- Unit 2 completion schedule in order to assure the heavy water inventory of this unit, the return of leased heavy water and the compensation of Unit 1 annual losses. A special programme for retrofitting and upgrading the plant is in execution.

3.3 Nuclear Fuel Plant

The Nuclear Fuel Plant (FCN) in Pitesti was commissioned in 1980 producing CANDU type nuclear fuel bundles using a domestic developed technology following AECL specifications. In order to fulfil the initial contracts requirements and to maintain the burn-up warranties stipulating that for the fuel this has to be approved by AECL and to be produced by a traditional CANDU supplier, the first load for Unit 1 in Cernavoda NPP was imported from Canada. In 1993 a contract was concluded with AECL and Zircatec from Canada which provided the backfitting of FCN, ascertaining the qualification of the latter as fuel supplier for the CANDU-6 type nuclear reactors. The qualification process was completed in 1994/1995.

The overall manufacturing capacity of the plant is 110 tonnes of fuel yearly, covering the annual consumption of the first unit. This capacity could be extended in correlation with the schedule to resume the work for Cernavoda NPP-Unit 2.

Another task in the future for this facility is the reevaluation of the fuel bundles (about 500 tonnes) produced before 1990 in order to qualify them for use in CANDU-6 reactor.

3.4. Main objective of the nuclear power programme

The basic strategical objective in a medium term (1995-2000) for the nuclear programme consists in consolidation of technical and economical competitive position of RENEL producers for electrical power in nuclear units, for nuclear fuel and heavy water.

In this respect the following actions will be taken:

<u>Cernavoda NPP</u>

- implementation of a control system of both the investment and production costs, in order to place them within the limits which have been already accepted at international level;
- completion of the Personnel Training Center and implementation of training programs;
- implementation of a modern basis for maintenance, inspections and repair programs;
- defining of a proper supply policy of the nuclear fuel, in close connection with its evolution at market level;
- assurance of technical and scientific support by engineering and research and development institutes in order to obtain an average long-term availability factor for nuclear units exceeding 80%;
- maintenance of a high nuclear safety level of installations, meeting all requirements of the international standards;
- keeping the radiation doses and the radioactive releases at the lowest possible levels below the allowed limits, by optimizing both the operation and maintenance procedures;

FCN - Pitesti

- increase of labour productivity in order to range the production costs at the international market level;
- the intensive utilization of output capacity so that the domestic demand should be completely covered.

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- completion of the project as well as up-grading the output capacities in order to perform the necessary manufacturing program up to 2000;
- implementation of a preventive maintenance program, in order to preserve the installations correspondingly;
- completion of the measures for raising the safety degree of facilities for protection of the operating personnel and the population;
- provision of a special control of the environmental factors in order to minimize the impact of the technological installations both upon population and environment.

4. ENGINEERING, RESEARCH AND DEVELOPMENT WORK

The outstanding design works and development studies drawn up during the past years have had a key contribution in the implementation of all power objectives, including nuclear units. There is a research and development plan for HWR programme following three main lines: CANDU technology improvement, nuclear safety and environmental protection and radioactive wastes and spent nuclear fuel management. Two RENEL institutes CITON and ICN, in cooperation with other institutes implied in nuclear field, assure the technical and research support asked for this plan.

4.1 Candu Technology Improvement

The programme is aimed to analyze and solve some issues related to upgrading of the NPP equipment operation, repair and inspections.

There has already been solved or under solutioning some items regarding the plant maintenance (maintenance standards, maintenance processing cards), equipment repair (fuel channel, feeders, radioactivity mechanisms), in-service inspections of NPP mechanical equipment (fuel channel, generator, pressurizer), guarantee inspections and tests, updated guides and standards, technical studies and technical solutions for the NPP equipment and system updating, analyses regarding the implementation of some design modifications and application of updated code versions, reliability studies, etc.

Among the accomplished results on the programme issues the following can be mentioned:
- finalization of NPP equipment reliability studies,
- technologies and designs for fuel channel replacement equipment
- computer programs to determine: pressure tubes axial displacement, behaviour of perforated plates subjected to elastic and elasto-plastic stresses; residual stresses in pipe flanged areas, stability analysis of the nuclear vessel tube elements etc.

For the next period (1996-1997), the following issues shall be approached:
- technologies and designs for special tools employed in the reactivity control mechanism repairing and interventions;
- technology and design for the fuel channel in-service control equipment;
- assessment of technical solutions for the in-service inspections of the main equipment associated to primary heat transport system: steam generators, pressurizer, piping, etc.

Qualitative, constructive and operational improvements of NPP equipment and systems shall also be considered, on the basis of the operation experience and our own technological studies and solutions.

4.2 Nuclear safety and environmental protection

This programme was elaborated to solve the following main issues:
- the fulfilment of a capacity to perform the nuclear safety analysis required for NPP Cernavoda-U1 commissioning and operation;
- the fulfilment of a capacity to analyze the U1 operation in order to improve the design solutions;
- the fulfilment of the possibility to analyze and accommodate to the new requirements in the field of nuclear safety;
- solutioning of other nuclear safety issues determined by the existence and operation of Cernavoda NPP (e.g.irradiated fuel and radioactive wastes disposal, radioprotection, evaluation of component aging effects on nuclear safety)

During the period 1992-1995, the following main items in the programme were fulfiled:
- Revision 1 of Nuclear Safety Probabilistic Evaluation Level 1 and IPERS - IAEA mission expertise;
- Mathematical models, methodologies and PC versions for Source Term Code Package (STCP) computer programmes;
- Fuel analysis codes (ELES, CAREB) similar to the Canadian ones and the completion of all the experimental facilities for in pile testing of the CANDU type fuel elements;
- Performance of safety analysis employing a large package of computer programs (RFSP, ORIGEN, NUCIRC, ELESIM/ELESTRES - for initial conditions, FIREBIRD III - for the heat transport circuit thermo-hydraulics, ELOCA and COREFPR for the nuclear fuel behaviour, PRESCON, for the containment and the radionuclides behaviour and PEAR for the environmental impact).

In order to fulfil the programme objectives, in the years to come, the activity shall develop on the following directions:

- further probabilistic evaluations of nuclear safety, considering the "as-built" design, PSA application in the plant operation and maintenance and further elaboration of PSA Level 2 and Level 3 both for Cernavoda NPP Units 1 and 2 (CNCAN licensing requirement for Unit2);
- further elaboration and finalization of technical limits and conditions, both for U1 and U2;
- implementation and operation of the programme for the evaluation of aging effects.
- STCP programe package adaptation for CANDU type reactor and Cernavoda NPP-U1 characteristics and the analysis of severe accidents;

- analysis of other accidents at Cernavoda NPP that were not approached in the documentation received upon AECL agreement and which are presently analysed with other plants of this type (CNCAN requirement);
- fulfilment of an integrated design, operation and nuclear safety analysis data base.

4.3 Radioactive wastes and spent nuclear fuel management

The main purpose of this program is the achievement of some storage facilities for all types of wastes resulted from the operation of NPP and from other nuclear units as well as for spent nuclear fuel elements.

For storage of both the average and low active wastes, within the years 1992-94, a series of studies were elaborated, which could determine the possibility of building a Final Storage Facility, close to Cernavoda NPP.

Along 1995, a pre-feasibility study for the Final Storage Facility of both low and medium active wastes has been issued, this building following to be put into operation in 2002, as the latest date.

Referring to the management of the spent nuclear fuel, the studies elaborated at present by CITON have shown the necessity of performing a Spent Fuel Intermediate Store which has to be put into operation during 2002 ÷ 2003, considering that the optimum storage period of the spent fuel at NPP is about 7 ÷ 10 years since NPP commissioning, depending also on the depositing type which can be either dry or wet. Under such circumstances elaboration of a pre-feasibility study should be advisable starting with 1996, when the optimum technical solution should be settled down.

Another major project of this programme represents the building of the Final Store for Spent Fuel, having its commissioning somewhere towards 2025 years.

For 1996 it is provided a continuation of the works which had been already performed in the framework of pre-feasibility and feasibility studies for the main projects, having their commissioning data in 2002 ÷ 2003.

All these studies will finalize the first stage of radioactive wastes and spent fuel management, obtaining thus, all the design data for opening up, in 1997, the investments of the following projects:
- Final Store for Low and Medium Active Wastes (DFDSMA), having its commissioning date: 2002;
- Spent Fuel Intermmediate Store (DICA) having its commissioning date 2002-2003.

In the next stage, the period 1997-2002, the programme will be in progress by supporting the performance of all the proposed investments in the field of developing the techniques, methodologies, testing procedures of radioactive wastes storage containers and casks, radioprotection calculations, evaluations of performances etc.

Finnally, we mention a set of nuclear research projects, sponsored by the National Agency for Atomic Energy, such as : evaluation of nuclear fuel cycles for the CANDU units, alternate possible options for reactors to be installed in the next NPPs, estimation of domestic industrial capability to supply competitive components, detritiation of heavy water by cryogenic technology.

We would like to underline here the permanent support which was granted by IAEA for our research and development works.

5 CONCLUSIONS

Considering the present status of economic development in our country, as well as the situation of natural power resources existing in Romania, the technical and moral wearing-out of a part of existing power capacities, the necessary rehabilitation of operating conventional power plants and the introduction of new investments, the massive funds invested already in nuclear power which, besides the Cernavoda NPP works, should also include the assembly of the necessary technologies, as well as the minimum impact of NPP upon the environment, we can conclude that the development of the nuclear power in Romania represents a priority both on average and long term.

Completion of Cernavoda NPP is considered as a viable alternative within RENEL strategy regarding the development of the national power system. Due to limited financing capacity, as well as estimation of the future evaluation of the power demand, Cernavoda NPP, which has been designed with 5 CANDU-PHW units of 700 MWe each, is provided to be performed in two stages:

- stage 1 - units 1 and 2, up to 2000
- stage 2 - unit 3,4 and 5 after 2000

Depending on the evolution of the economic conditions in the country this scenario of Units 3,4 and 5 implementation could be included in a more accelerated performance programme. This could depend also on future electrical power demand in South-Eastern European area in connection with decommissioning of old nuclear power units (Kozlodui NPP, Chernobyl NPP). It is recognised the fact that the only realistic way of developing the National Nuclear Programme needs to be further on supported by the state budget. Of course, lack of funding could direct us to other options such as attracting foreign finance and implementing a certain project financing scheme. In this respect all developing activities should be properly coordinated in order to assure on optimum correlation between all involved fields (research and development, implementation of design, development of raw material basis, manufacture of material and equipment, technical support and services for maintenance, operation, environmental control, etc). This should be performed by authorized bodies such as RENEL, ANEA, CNCAN, having the support of the speciality institutes.

Our own efforts should also take into account the exchange of information and the cooperation works with the other partners belonging to CANDU owners group family.

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PROGRESS OF CHINA'S NUCLEAR POWER PROGRAMME

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Abstract

From a long-term point of view, nuclear power is the only solution for the shortage of energy resource. Nuclear power development strategy has been specified in China according to national condition. The electricity development of nuclear power optimizes the national energy structure and ensure the power supply, particularly in east China. China's first self-designed and self-constructed nuclear power plant--Qinshan Nuclear Power Plant (300MWe PWR) is now well under commercial operation. China is willing to cooperate with IAEA, other countries and regions in the field of nuclear energy for peaceful use on basis of mutual benefit.

1. Status of Nuclear Power Development

The idea of developing nuclear power in China has long been conceived. It was pointed out in the 12-Year-Outline for Atomic Energy Development Program put forward by the central government in 1955 that "Nuclear power which ushers in a new era in the history of power development, has brilliant prospects. An integrated power supply system shall be taken shape in China within the coming 10 years, which includes primarily comprehensive exploitation of rivers and development of thermal and hydroelectric power as well as use of nuclear power depending on the practical conditions". On February 8, 1970, hearing the report on a shortage of electricity in Shanghai given by the municipality, the late Premier ZHOU Enlai said:"From a long-term point of view, nuclear power is the only solution for the shortage of electricity in Shanghai and East China". China's first nuclear power plant project was therefore named"728 Project". From then on, the prelude to develop nuclear power was drawn in China.

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China is a large and developing country. In recent years, the gross national product is increased by about 9% per year. Lack of electricity arose in many regions, specially in southeast China. Because 80% of coal resource is in north China, 70% of hydroelectric resource is in Southwest China, and the developed east, northeast & south China only possess about 15% of nation's energy resource. However 50% of train and one third of water freight transport capability have been used for transport of coal.' So in response to economic growing demand, perhaps the only way is to build a group of nuclear power plants in east China. (See Fig.1)

The first concrete was poured on March 20, 1985 for China's first self-designed and self-constructed nuclear power plant--Qinshan Nuclear Power Plant (300MWe PWR). And it was connected to the grid and started to generate electricity on December 15, 1991 under the joint efforts of all the builders, and reached full power operation in July 1992. It had entered the period of high-power trial operation then. On the First of April 1994, Qinhsan 300MW NPP got into commercial operation. First cycle operation was finished in Oct. 1994. After refueling and necessary maintenance, the Qinshan plant restarted and connected to grid for second cycle operation in Jan. 1995. It is expected that the plant load factor would reach about 84% in 1995.

The successful construction of Qinshan NPP, which results from both making full use of our achievements in science and technology over the past 30 years and absorbing the advanced experience of other countries thanks to the opening policy, is an another significant breakthrough in peaceful use of nuclear energy and nuclear technology. It marks an end of no nuclear power in China's mainland and initiation of a new stage in peaceful use of nuclear energy and nuclear technology.

Daya Bay NPP (2 x 900MWe PWRs) has been one of the largest joint venture projects since China started the reform and opening policy. For the Unit 1, the first concrete was

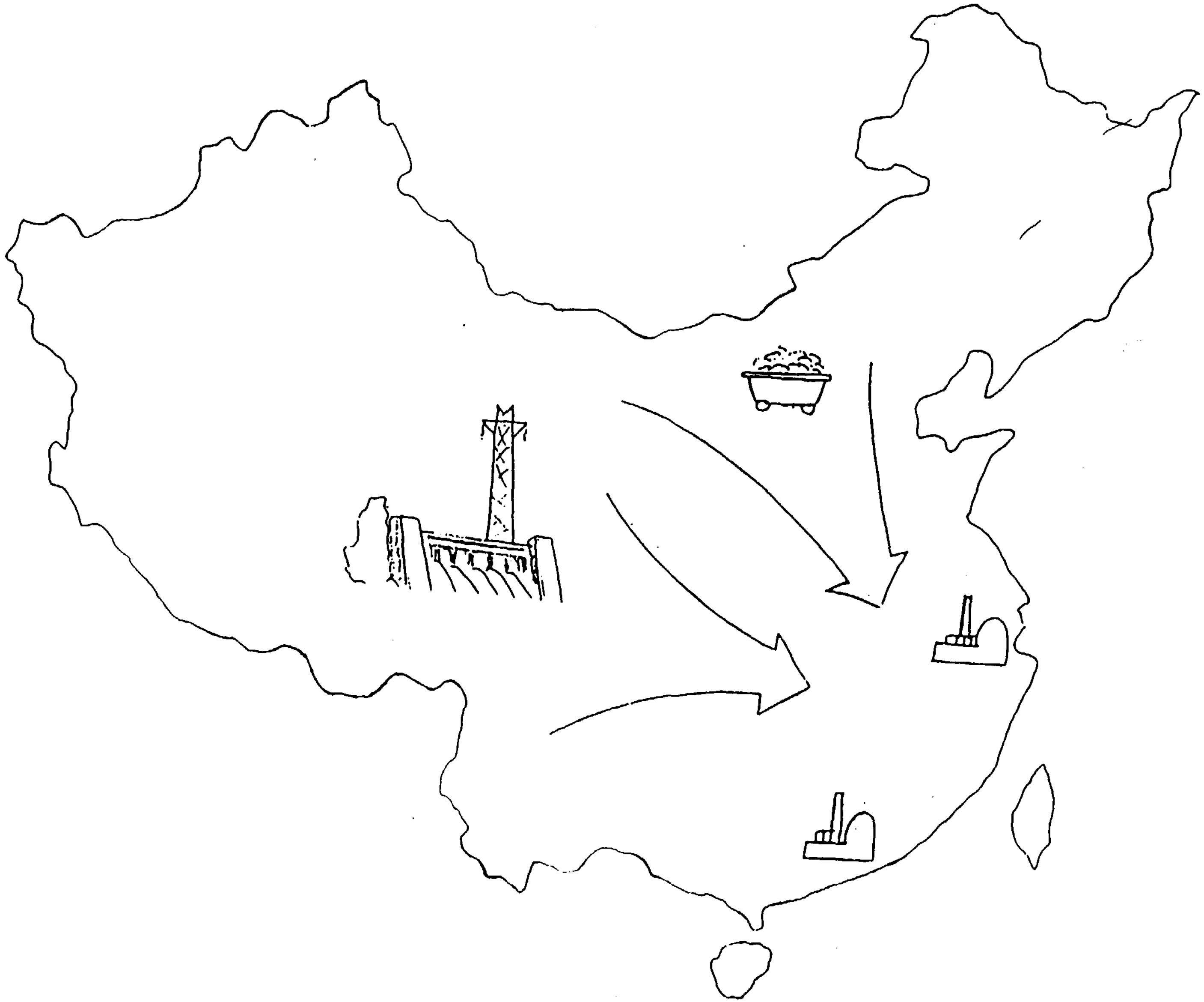

Fig 1 Energy Resource in China

poured on August 7, 1987, and connection to the grid and generation of electricity, initiated on August 31, 1993, and commercial operation reached in Feb. 1994; For the Unit 2, the first concrete was poured on April 7, 1988, and commercial operation reached on May 6, 1995. Both unit had finished first cycle and refueling and is now on second cycle operation.

The construction and operation of the Daya Bay NPP plays a major role in learning the foreign advanced techniques and management experience and in improving China's construction and management of nuclear power plant.

The second phase of Qinshan NPP Project (2 x 600MWe NPPs) has been placed on the National Plan as one of the major items. The review of the preliminary design was fulfilled in November 1992. It is scheduled that the connection to the grid and generation of electricity shall be initiated by the end of 2000, and for the Unit 2, one year after.

The recommendation for the second phase of the Guangdong NPP Project (2 x 900 MWe PWR Units) was approved and preparations are underway.

The feasibility study on construction of nuclear power plant is actively carried out in Liaoning Province and in the southeast coastal provinces such as Shandong, Jiangsu, Zhejiang, which are well developed in economy but short of coal and power resources.

2. Nuclear Power Program and Policy

In January 1984, China joined in the IAEA and acts as a designated member of the Board of Governors, taking an active part in activities of international nuclear cooperation. In November 1988, China concluded an agreement with the IAEA to voluntarily place some of its civilian nuclear facilities under the Agency's safeguards; and has accepted Agency's inspectors to inspect the nuclear facilities in China. In January 1989, China acceded to Convention on the Physical Protection of Nuclear Material. In March 1992, China acceded to the Treaty of the Non-Proliferation of Nuclear Weapons. In February 1993, China gave the Agency a pledge to report on its import and export of nuclear material, which means China's support to the efforts made by the international community in non-proliferation of nuclear weapons.

Since 1990s, China has entered a period of rapid development of national economy, thus presenting a new challenge to the existing energy industry which can hardly meet the needs of the economic development. As estimated, there is a shortage of conventional energy resources amount to 0.1 - 0.2 billion tons of standard coal by 2000 and 1.2 billion tons of standard coal by 2050. The situation of energy shortage is more serious in southeast coastal areas with fairly developed economy, therefore, alternative energy resources must be developed as early as possible so as to mitigate this gap.

30 years of experience in developing nuclear power in the world can fully witness that nuclear power possesses safe, clean, and economical features. The electricity development of nuclear power is the only way to optimize the national energy structure and ensure the power supply. In order to fit in with the objective needs of economic development, the general principle of developing nuclear power is "Developing mainly the thermal and hydroelectric power while the nuclear power as and auxiliary"; for the southeast coastal areas, the principle is "Developing the thermal, hydroelectric, and nuclear power in the light of local conditions". At the same time, nuclear power development strategy has been specified, taking into account the national conditions.

In technology, we shall conscientiously sum up the experience in construction and operation of Qinshan NPP and Daya Bay NPP. Through the construction of Qinshan NPP, we have mastered the techniques in design and construction of nuclear power plant with 300 MWe scale. 70 percent or more of the equipment used in the Plant were manufactured in China. Now China's nuclear power construction is climbing to a new level.

The construction and operation of the Qinshan NPP has heightened our confidence in developing nuclear power and demonstrated the superiority of nuclear power as new energy resources. As a result, several provinces and municipalities have made applications for constructing nuclear power plant. The safety and reliability of nuclear power are much concerned by the public. The principle of "Safety First and Quality First" is consistently followed throughout the construction of nuclear power in China and specific measures are adopted accordingly:

- An authoritative organization for independent implementation of safety supervision to civilian nuclear facilities - China National Nuclear Safety Administration (NNSA) was established in 1984. on the basis of the IAEA NUSS documents, the NNSA has enacted and promulgated four nuclear safety codes relating to design, siting, operations, and quality assurance, and a series of standards and guidelines. As a competent body of nuclear safety, the CNNC has also promulgated a series of administrative rules relating to operation and radiological safety control, etc. and at the same time, it possesses a complete quality assurance system and special organizations and personnel to deal with research work and management for nuclear safety, and international cooperation in this field.

- From the very beginning of development of nuclear power, all the constructors are always instructed to have a strong sense in safety and quality, and also to have high professional morality to the State and the people.

- During design, construction, and operation of the nuclear power plant, the IAEA experts are invited to carry out safety review. And our work will be improved according to their comments.

- The operators of nuclear power plant shall be strictly trained and further trained. They are allowed to take their posts, only when they have obtained their licenses.

These measures will basically ensure the safety and reliability of nuclear power plants in China.

3. International Cooperation and Peaceful Use of Nuclear Energy

Promoting the peaceful use of nuclear energy throughout the world for the benefit of mankind is one of the objectives set forth in the Statute of the IAEA. One of the major tasks in China's nuclear industry is extensively developing international cooperation to promote peaceful use of nuclear energy and nuclear techniques.

Since 1980s, under the guidance of the reforming and opening policy, China has developed extensive cooperation with many countries. In the course of design and construction of the Qinshan NPP, advisory opinions were acquired from foreign experts and some of the major equipment, were imported. Daya Bay NPP is constructed by joint venture with HongKong, importing foreign equipment and technology. The principle of "Relying mainly on our own efforts while cooperating with foreign countries" shall be followed in construction of the second phase of Qinshan Project. In manufacture of nuclear fuel assemblies, foreign technology shall be introduced to backfit the existing production line, etc. The cooperation inevitably benefits us to fully control of nuclear power technology and further promote the development of nuclear power construction.

In 1990s, the width and depth in international cooperation shall be greatly increased. Some of the untouched cooperative scope and forms before are now open for discussion, including construction of nuclear power plant by joint investment and joint development of new-type reactors and spent fuel reprocessing technology for power reactors. Though PWR is selected as major type of NPP, the other type of NPP are to be considered for sake of increasing nuclear power supply capacity. A plan to build a CANDU type NPP is under consideration. The active cooperation between CNNC and AECL is being taken for building PHWR NPP on Qinshan site in China. Under the circumstances the nuclear power industry will be developed steadily, which shall undoubtedly provide a broad prospect for cooperation in the field of nuclear technology.

While introducing foreign investment, equipment, and technology, improving its technique level and construction capability in nuclear power development, China is ready to play an active role in the world market for the sake of promoting peaceful use of nuclear energy and nuclear technology in the world. Now China's nuclear industry, developed on the basis of self-reliance, is in a position to export nuclear power plant and research reactors. China's technology and experience fairly meet the needs of the developing countries; following the principle of "Equality, mutual benefit and common development", China is willing to cooperate with other countries and regions in the field of nuclear energy and nuclear technology for peaceful purposes.

China has concluded inter-governmental gareements with 12 countries on the cooperation of peaceful use of nuclear energy, and has established relations of exchange and cooperation with many countries and regions. On the basis of independence, self-reliance, equality, and mutual benefit, China wishes to carry out active cooperation with foreign countries in science and technology, economy and trade, and continue to make its due contributions to the prosperity and development of nuclear undertakings in the world.

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THE PRESSURIZED HEAVY WATER REACTOR PROGRAMME IN THE REPUBLIC OF KOREA

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Abstract

KEPCO(Korea Electric Power Corporation) opened its HWR program in 1977 by starting the construction of Wolsong#1(CANDU-6). Wolsong#1 completed its construction in 1983 and has been operated successfully achieving an average lifetime capacity factor of 85%. It has recorded three times world's top ranked nuclear unit during last twelve years of operation. It is attributed to the outstanding operation, minimization of forced outages and short overhaul periods by KEPCO operation team.

KEPCO restarted HWR program in the year of 1990s signing contracts with AECL for Wolsong #2 on Dec. 28, 1990 and for Wolsong 3&4 on Sep. 18, 1992. Wolsong#2,3&4 are 700 MW class PHWR which are identical to Wolsong#1 and being constructed at the same site. The construction of Wolsong#2,3&4 is well progressed and they are scheduled to be connected to the grid in 1997, 1998 and 1999 respectively.

While Wolsong#1 was turn-key project, Wolsong #2,3&4 is component approach project. KEPCO performs overall integrated project management, AECL is the prime contractor to provide A/E services and NSSS system design and equipment supply, in each case Korean industy is participating as subcontractors, for T/G supply HANJUNG is responsible with a foreign partner GE and construction is performed by HYUNDAI and DAEWOO. Korean industry could expand its area in PHWR using PWR technology achieved through technology self reliance program. As a result, equipment localization of Wolsong#3&4 is 51% in NSSS, 80% in BOP and 90% in T/G and total localization of Wolsong 2,3&4 is 65%.

As far as PHWR technology concerned, KEPCO didn't have self reliance program like PWR but for the PHWR technology development of Korea, KEPCO and AECL made technology transfer agreement for the CANDU-6 NSSS design at the time of Wolsong#3&4 contracts. Under this agreement, AECL has provided the CANDU-6 design technology and it is contributing to the development of Korean PHWR design technology. In the field of fuel, KAERI(Korea Atomic Energy Research Institute) is carring out the feasibility study of CANFLEX and DUPIC.

Based on experience, KEPCO is trying to export the nuclear technology. KEPCO signed an agreement with AECL for the cooperation of CANDU export to third countries. It is expected the participation of KEPCO in the Chinese Qinshan project which AECL is negotiaing China Nuclear Corporation for the supply of two CANDU-6.

Planning of future CANDU projects in Korea as well as type and capacity if consructed is under study by KEPCO. The economic aspects of CANDU-6, the future prospects of

CANDU-9 technology development and availability of construction site will be the important points of study and it will be finished next year. The result of study will be reflected to the new Korean electric power development plan, which will extend Korean power plant construction schedule until 2010.

I. Introduction

The Pressurized Heavy Water Reactor (PHWR) program in Korea consists of three principal segments. The first focused on the turn-key based construction and the operation of Wolsong Unit 1, a typical PHWR of 700 MWe class designed by Atomic Energy of Canada Limited (AECL), while the second is directed of constructing three CANDU units employing the identical design to Wolsong Unit 1 with technology transfer agreement between Korea Electric Power Corporation (KEPCO) and AECL. The third is the future development of PHWR technology and fuel cycle.

This paper will outline the status of operation, construction, development direction of PHWR in Korea and international cooperation.

II. Turn-key based construction and the operation of PHWR

Generally, the CANDU PHWR offers principal advantages over Pressurized Water Reactor (PWR) ;

- on-power refueling and flexible power management
- extremely high neutron economy provided by heavy water moderator enabling the use of natural uranium fuel.

Accordingly, KEPCO selected CANDU PHWR as the complementary reactor type to the PWR in its nuclear reactor strategy, considering the diversity of nuclear technology and the nuclear fuel supply. KEPCO started the construction of Wolsong Unit 1 (CANDU-6) on turn-key based project in 1977. AECL and NEI Parsons were the Nuclear Steam Supply System (NSSS) manufacturer and the Turbine Generator manufacturer, respectively. A domestic industry, Dong-Ah Construction and Industry Co. was only participated in the project as the civil constructor.

Wolsong Unit 1 was completed in April 1983 and has been operated successfully by achieving an average lifetime capacity factor of 85% which is 6 % higher than that of the other Korean nuclear units since then. It has achieved the world's best annual performance in 3 of its 12 years of operation as shown in Table 1 (World's Top Record in Capacity Factor by Wolsong Unit 1).

This could be achieved through the minimization of forced outages by KEPCO operation team and the short overhaul periods due to the on-power refueling, which is a CANDU's unique characteristics. Annual operation record of Wolsong Unit 1 is presented in the Table 2 (Annual Operation Record of Wolsong Unit 1 '83 - '94).

Duration	Capacity Factor (%)	Reported By
Apr. 1, '85 ~ Mar. 31, '86	98.4	NEI Statistics
Oct. 1, '91 ~ Sep. 30, '92	98.0	"
Jan. 1, '93 ~ Dec. 31, '93	100.8	Nucleonics Week

Table 1 : World's Top Record in Capacity Factor by Wolsong Unit 1

Year	Generation (GWh)	Capacity Factor (%)
1983	2,562	61.90
1984	3,985	66.80
1985	5,610	94.40
1986	4,736	79.70
1987	5,521	92.87
1988	4,732	79.38
1989	5,411	91.01
1990	5,106	85.89
1991	5,417	91.12
1992	5,177	86.84
1993	5,993	100.81
1994	4,912	82.62
Cumulative	59,163	85.02

Table 2 : Annual Operation Record of Wolsong Unit 1 '83 - '94

Unit : Won/KWh

Average Cost of Generation	'87	'88	'89	'90	'91	'92	'93	'94
Overall	36.50	32.15	29.36	29.81	27.82	29.48	30.36	31.76
Nuclear	27.41	26.62	23.62	23.75	22.62	25.31	24.57	22.70
Wolsong #1	26.55	28.85	23.22	26.16	23.74	22.47	21.13	29.49

Table 3 : Comparison of Power Generation Cost from 1987

KEPCO consistantly finds its nuclear units to be the lowest cost source of power. Statistics have shown that Wolsong Unit 1 has frequently produced power at an annual average cost below than that of the average nuclear units since start of its commercial operation as shown Table 3 (Comparison of Power Generation Cost from 1987).

III. Construction of Wolsong Units 2, 3&4

A. Project Structure

KEPCO resumed the PHWR program in the early 1990s upon signing the contracts with AECL to build 3 CANDU PHWR units - one for Wolsong Unit 2 on December 28, 1990 and the others for Wolsong Units 3&4 on September 18, 1992. These units are 700 MWe class PHWRs identical to the Unit 1 and are being constructed at the same site. While the first unit was constructed by AECL, as the prime contractor on turn-key project, these units are under construction on component approach by KEPCO.

Responsibilities of participating organizations are as follows ;

- ○ KEPCO as the final owner, performs overall integrated project management, construction management, procurement of BOP equipment and commissioning for the project

- ○ AECL as the prime contractor, provides Architect Engineering (A/E) services and supplies equipment and system design of NSSS. In order to upgrade Korean nuclear industry's engineering capacity through this project, domestic companies are participating as subcontractors ; KOPEC for A/E services, HANJUNG for NSSS equipment supply, and KAERI for NSSS system design.

- ○ HANJUNG is the prime contractor for T/G supply with a foreign partner, General Electric as subcontractor.

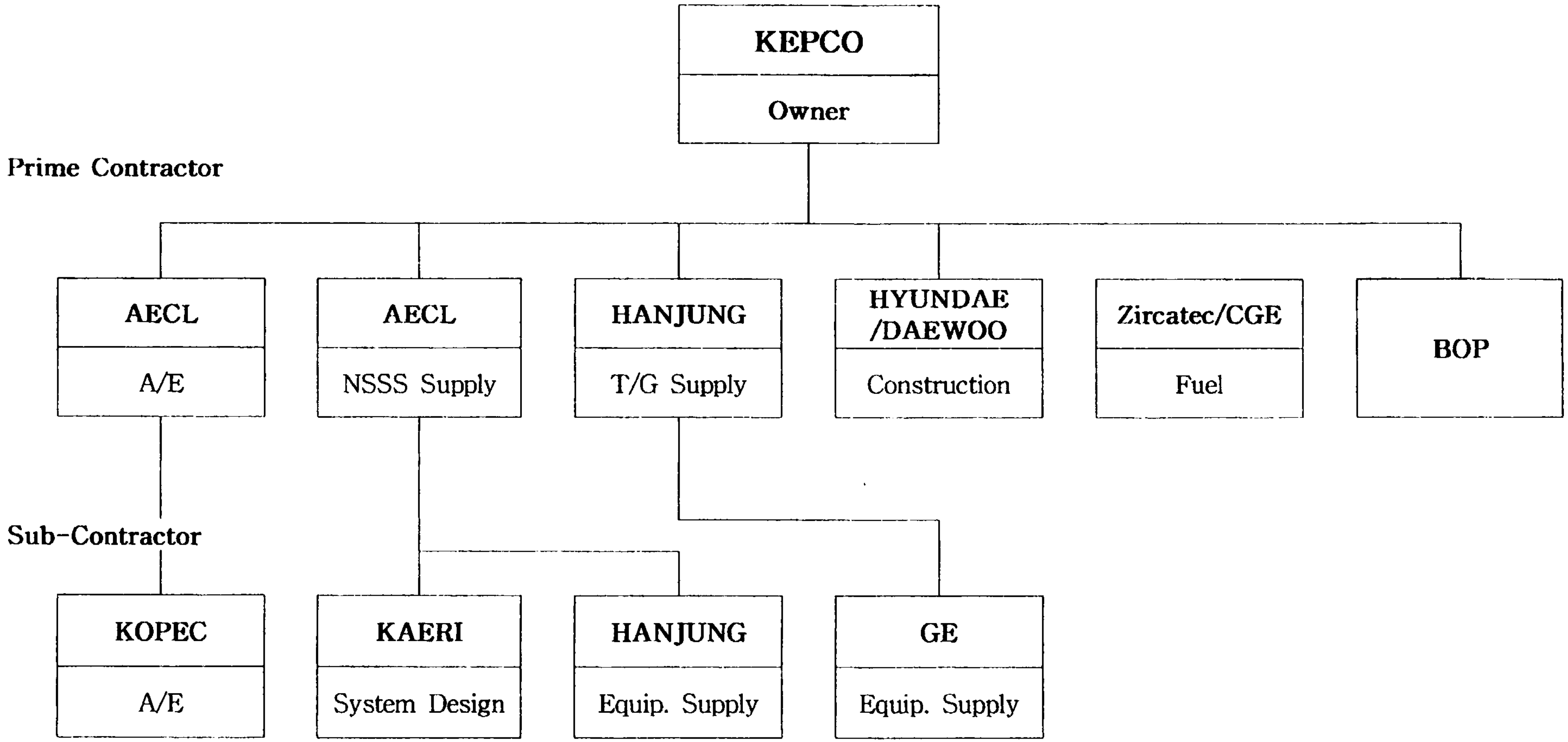
- ○ HYUNDAI and DAEWOO are responsible for the construction of Wolsong Unit 2 and Wolsong Units 3&4, respectively.

Organization chart for this project are attached in Figure 1 (Organization Chart for Wolsong Units 2, 3&4).

B. Project Summary

Even though contracted at different times, Wolsong Unit 2 and Units 3&4 are managed as one project. As a result of this integrated approach to project

Figure 1 : Organization Chart for Wolsong Units 2, 3&4



management, AECL and Korean contractors are able to retain their key personnel for all three units. This enables KEPCO to benefit from the increased efficiency of Korean engineers who have experience specific to the CANDU project.

Status of the project is summarized in Table 4 (Project Summary of Wolsong Units 2, 3&4). As shown in Table 4, KEPCO enables to shorten the construction period of Unit 3 six months less than that of Unit 2 due to the previous experience. The construction period from first concrete pouring to completion is 58 months for Unit 2 and 52 months for Unit 3. Also the localization ratio is increased from 58 % for Unit 2 to 69 % for Units 3&4. These figures show a great improvement, considering that the ratio of Unit 1 was only 14 %. Total project cost is estimated at 3,989 million US dollars.

As the end of December 1995, the progress of Unit 2 and Units 3&4 is 84.15 % and 54.38 % of total progress, respectively. Currently, these units are expected to be connected to the grid for three successive years from 1997 to 1999, adding approximately 2,780 MWe capacity accounting for a considerable portion of KEPCO's nuclear capacity

Item	Wolsong #2	Wolsong #3, 4
Generator Output	700 MW Class	700 MW Class
Construction Period (First concrete ~ C.O.D.)	58 Months (Sep. '92~Jun. '97)	64 Months (Mar. '94~Jun. '99) (#3 : 52 Months)
Project Cost - Domestic Capital - Foreign Capital - Total	 \$ 960 Million \$ 414 Million \$ 1,374 Million	 \$ 1,899 Million \$ 716 Million \$ 2,615 Million
Construction Unit Cost	\$ 1,945 / KW	\$ 1,850 / KW
Localization Ratio	58 %	69 %
Overall Progress (Dec. '95)	84.15 %	54.38 %

Table 4 : Project Summary of Wolsong Units 2, 3&4

IV. Development Direction

A. Technology Development

As far as PHWR technology concerned, KEPCO didn't have an extensive technology self-reliance program like the program executed for PWR technology self-reliance in 1986. However, for the PHWR technology development of Korea, KEPCO and AECL made technology transfer agreement for the CANDU-6 NSSS design at the time of Wolsong Units 3&4 contracts.

Under this agreement, AECL has provided the CANDU-6 design technology and it is contributing to the development of Korean PHWR design technology. Since then AECL has supplied KEPCO with the documents, patents and computer codes. The documents are engineering procedure, plant engineering and system design documents, design support documents, generic CANDU design documents and R&D report. KAERI, KOPEC and HANJUNG are entitled to use these documents in Korea.

In parallel, KAERI partially performs a joint study with AECL for CANDU improvement project and KAERI and AECL are operating a cooperative program for DUPIC cycle, i.e., the direct use of spent PWR fuel in CANDU.

IAE (Institute for Advanced Engineering) sponsored by DAEWOO carries out Development of Plant-Control Simulation Analyzer Supporting Design and Operation of CANDU NPP with KAERI.

DUPIC long-term plan is as follows ;

- Phase I (1992~1993) : feasibility study
- Phase II (1994~2000) : experimental verification
- Phase III (2001~2010) : pilot scale verification
- Phase IV (2011~) : commercialization

B. Future plan for the CANDU plant

Feasibility on subsequent CANDU projects in Korea is being performed by KEPCO. This study includes the capacity of CANDU if it is constructed in future. The economic aspects of CANDU-6, the future prospects of CANDU-9 technology development and availability of construction site will be the important point of study and it will be finished next year. The result of study will be reflected to the long term power development plan to be revised, extending Korean power plant construction schedule until 2010.

It is also essential for KEPCO to maintain the current excellent operating record of Wolsong Unit 1 by continuously updating operating experience and research on O&M technology which will be applied to operate Wolsong Units 2, 3&4 to the same level in future. In an effort to achieve this goal, Korea Electric Power Research Institute (KEPRI), former Research Center of KEPCO are performing the R&D activities as follows ;

- Development of tritium removal technology
- Study on the optimal operating conditions of vapor recovery system for Wolsong Unit 1
- Development of C-14 analysis technology
- Probabilistic safety analysis Level II for Wolsong Units 2, 3&4

V. International Cooperation

As mentioned in the above section, while Korean research entity such as KAERI and IAE are participating the international cooperation program with AECL, KEPCO made the technology transfer agreement for the CANDU-6 NSSS design upon Wolsong Units 3&4 contract. Furthermore KEPCO has joined the CANDU Owner's Group (COG) to exchange technical information with the other members since November 1986. The highlight of the international cooperation between KEPCO and AECL is to sign the Memorandum of Agreement for CANDU Export to Third Countries, establishing the framework for future cooperation in November 1994. Through this agreement, both parties shall cooperate to jointly explore third country markets for CANDU export and to jointly participate in CANDU projects in such countries.

For marketing in third countries, particularly in the Asia Pacific region, cooperation will include, but not be limited to,

- Exchange of information
- Developing a marketing plan requiring project joint action
- Developing a proposal package
- Holding CNADU export strategic review meetings
- Dispatching a joint resident team, where necessary or appropriate.

Based on the agreement spirit, KEPCO is trying to export the nuclear technology. It is expected the participation of KEPCO in the Chinese Qinshan project which AECL is negotiating China Nuclear Corporation for the supply of two CANDU-6.

Ⅵ. Conclusion

KEPCO continuously update its operation and maintenance technology and perform the research and development activities to keep the excellent operation records for the PHWR. In parallel, KEPCO are performing the feasibility study for the building the subsequent PHWR units in Korea. On the other hand, international cooperation for technology development, market development and the advanced fuel cycle development will be maintained between Korean nuclear industry and AECL. As a conclusion, although we do not have an extensive PHWR program, by incorporating the above mentioned efforts, we expect a long term beneficial impact on our reactor strategy through this program.

ADVANCES IN PLANT DESIGN

(Session 2)

Chairman

D. MENELEY
Canada

DEVELOPMENT OF THE ADVANCED HEAVY WATER REACTOR

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XA9846596

Abstract

The design of the Advanced Heavy Water Reactor (AHWR) is influenced by the objectives of India's Nuclear Power programme. The AHWR utilizes heavy water moderator and light water coolant. The safety approach is based on incorporation of passive safety systems, along with some active systems to attain a high degree of safety. The core design will accommodate both mixed oxide fuel and ThO_2 fertile material in order to achieve the goals of India's programme. Maximum advantage is taken from experience with India's current HWRs.

1.0 Introduction

Thorium utilisation has been an important aspect of Indian Nuclear Power Programme. The strategy of realising large scale power generation from thorium has been clearly laid out right since the inception of Indian Nuclear Power Programme. As a part of this strategy an Advanced Heavy Water Reactor (AHWR) is being developed. This reactor would generate most of its energy from U-233, bred in-situ from thorium. The reactor is based on well proven water reactor technologies and would enable a number of passive systems to be incorporated in its design. The evolving design of Advanced Heavy Water Reactor discussed here aims at thorium utilisation in a thermal reactor with a special emphasis on passive safety features and economic competitiveness.

2.0 Genesis of AHWR

It is reasonably well known that thorium works almost as well in thermal neutron spectrum as in fast neutron spectrum. This is due to the eta value (no. of neutrons liberated for every neutron absorbed in fuel) of U-233, which is very nearly constant over a wide neutron energy range extending from ev right up to Mev. Eta value of U-233 compares very favourably with that of other fissile nuclei. For example, in thermal neutron environment the eta value for U-233 is the highest among all the fissile nuclides. Usually, a fissile specie yields higher eta value in fast neutron spectrum than in thermal neutron environment. That is how a fast reactor can generate more fissile material than its consumption. In this connection it is interesting to note that eta value of U-233 in thermal spectrum is marginally larger than the eta displayed by U-235 in fast neutron spectrum. Heavy water reactors are therefore good candidates for thorium utilisation. Nuclear power development worldwide is passing through a critical phase. New design schemes are being proposed which have the promise of greater safety. There are several debates related to the merit of a totally new revolutionary design vis-à-vis the benefit of vast experience already accumulated with existing designs. Without going into these issues, it is quite clear that a greater level of confidence in safety is required to be built into the

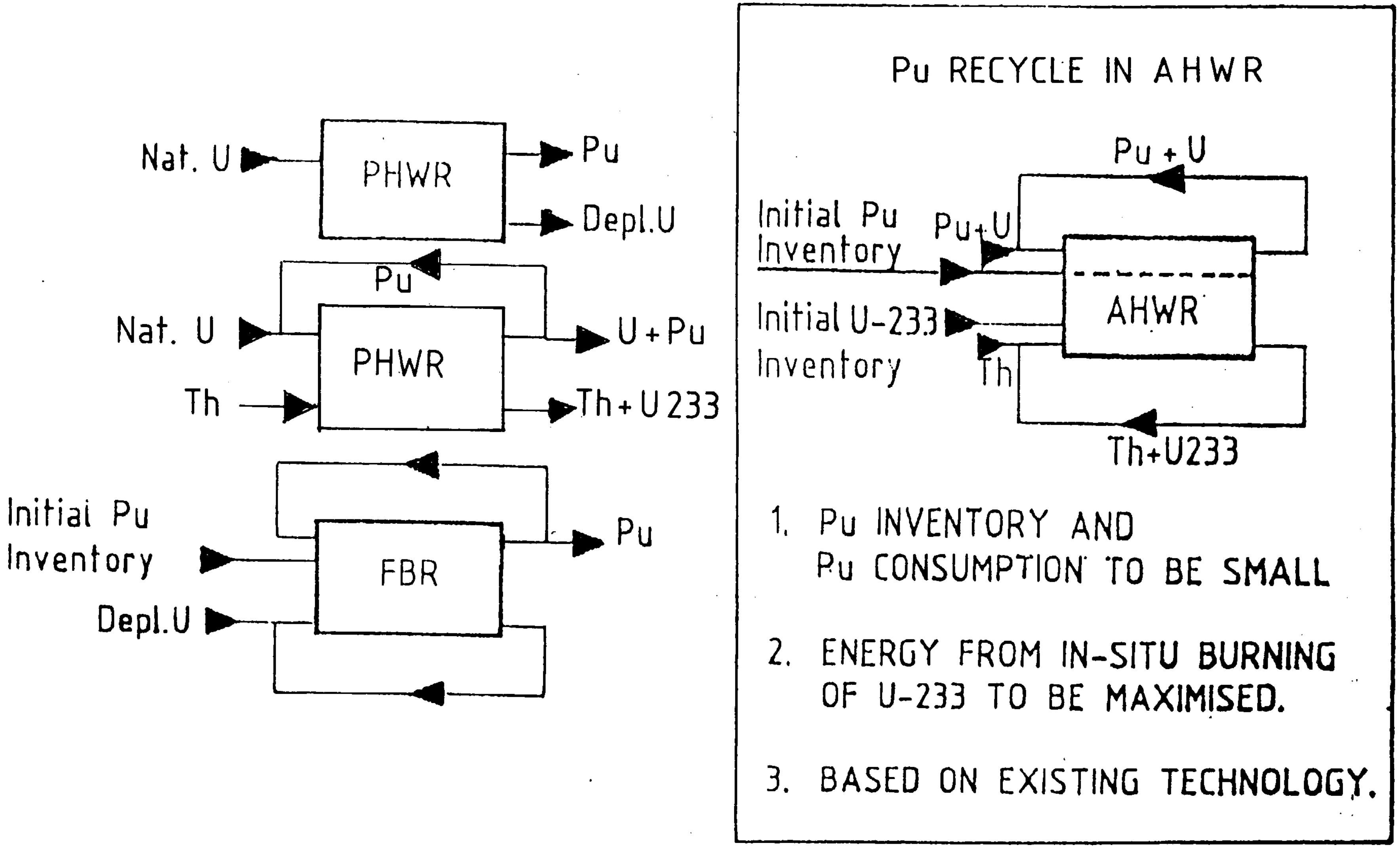

FIG. 1 FUEL CYCLE OPTIONS FOR INDIAN NUCLEAR POWER PROGRAMME.

design of newer reactors. The trend is towards incorporating greater degree of passive safety systems for carrying out different safety functions.

The motivation for the design of Advanced Heavy Water Reactor arises from above background. One of the important characteristics of thorium is its capability to produce large amounts of energy through burning of in-situ bred uranium-233. Thorium-U233 fueled lattices can achieve this characteristic without a very large change in reactivity unlike other fissile- fertile fuel combinations not involving thorium. Thus, while the burn-up realisable in Self Sustained Equilibrium Thorium (SSET) mode is moderate, with a small reactivity boost, one can augment the burn-up of thorium-U-233 fuel quite considerably. This however, requires a net fissile input into the system on a sustained basis. Plutonium is a logical choice for this purpose under our conditions. Plutonium however, has a relatively high capture-to-fission ratio in soft neutron spectrum. Therefore it is necessary to locate plutonium in regions where spectrum is somewhat harder. This would keep to a low level the consumption of plutonium as also the power generated from plutonium while the power derived from thorium is maximised. This indeed is the genesis of Advanced Heavy Water Reactor (AHWR). Figure-1 presents various fuel cycle options and indicates the role AHWR can play in Indian Nuclear Power Programme

3.0 Core Design Specifications

The main core performance requirements to be satisfied in the design of AHWR are given below:

Most of the power generation should come from thorium [2].The objective is to achieve more than 75% power from thorium. This is a very basic core physics objective for AHWR.

Another important feature specified for AHWR core is a negative void coefficient of reactivity under all operating conditions. A negative void coefficient will bring down the reactor power safely whenever the coolant supply becomes inadequate irrespective of the availability of the power supply, instrumentation or control systems.

The core configuration evolved for AHWR [2] meets the conflicting requirements of negative void coefficient, acceptable power distribution and reasonably low fraction of power output from plutonium.

4.0 Reactor Configuration

The reactor has a vertical tube type construction. While heavy water has been retained for moderator, coolant will be boiling light water. Boiling light water enables heat removal from the core through natural circulation which makes circulating pumps unnecessary.

All safety systems are configured in four identical loops. Each of these loops is rated at 50% of overall plant requirement. This 4X50% philosophy will ensure 50% stand-by capacity even when one of the four loops is under maintenance.

The reactor vessel is housed in a water filled reactor vault which acts as an effective radiation shield and passes most of the radiation energy to the feed water going to the reactor.

Reactor building will provide a double containment for the reactor (Figure 2). The top of primary containment shell will be closed by a composite cellular concrete slab. The hollow of this cellular slab will store a large water inventory called Gravity Driven Water Pool (GDWP). GDWP inventory will be sufficient to cool the reactor for three full days following an accident.

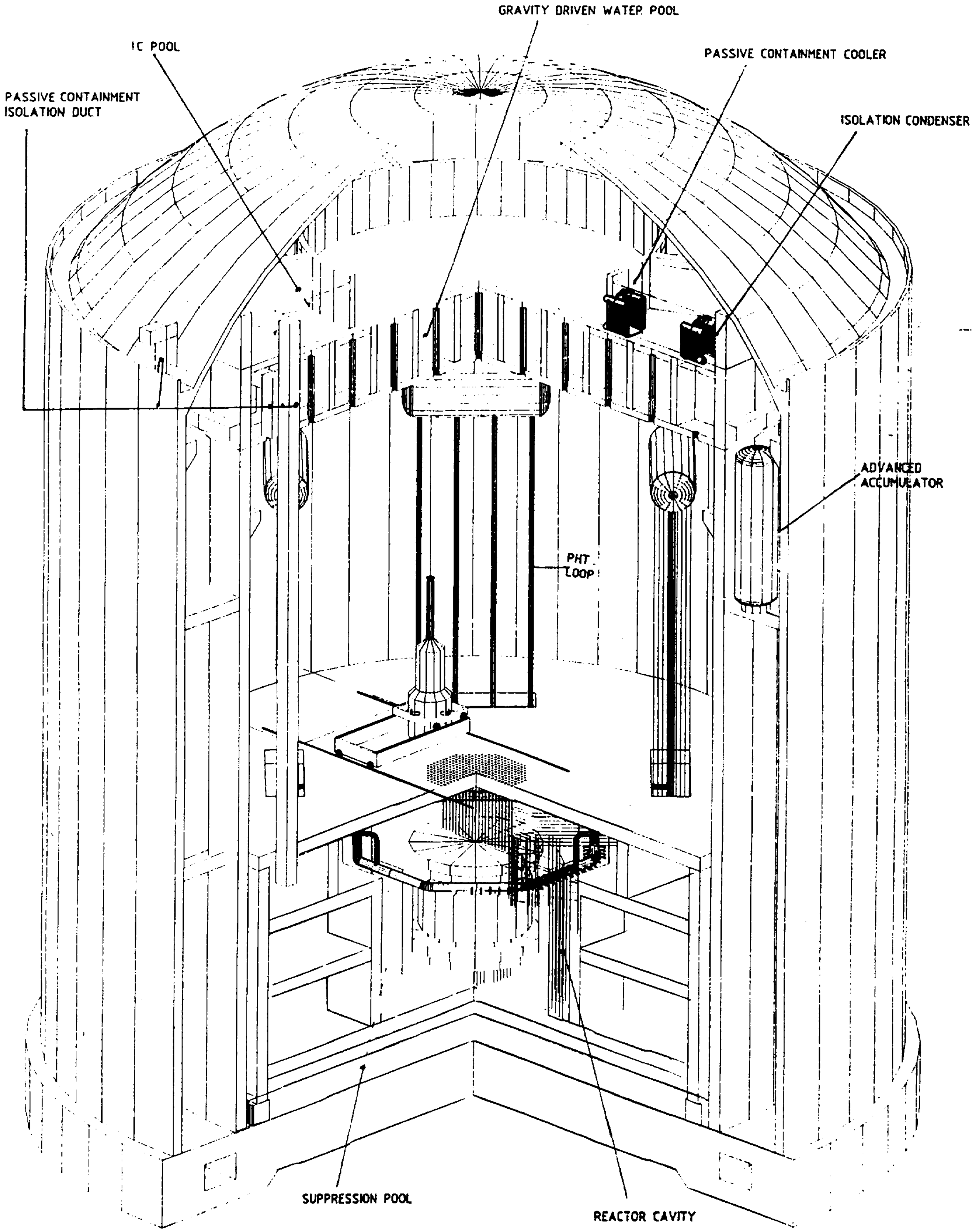

FIG. 2 PASSIVE SAFETY FEATURES OF AHWR

An annular water storage tank called IC Pool is provided on the primary containment roof. So called Isolation Condensers (IC) positioned in the IC Pool will transfer reactor decay heat to the IC Pool during short reactor shut downs. Another set of similar condensers located in the IC Pool will cool the primary containment during LOCA. Like the GDWP, the IC Pool will also be divided into smaller compartments for ease of maintenance. Seismic analysis is being done to establish feasibility of the IC Pool and GDWP locations.

Primary containment will be divided in two volumes viz. V1 and V2. Volume V1 will house the high enthalpy primary heat transfer (PHT) system and it will be connected to V2 through the suppression pool. In the event of a LOCA, the water flowing out of PHT system will collect in the reactor cavity (space around the reactor), thus ensuring core submergence. The water in excess of core submergence requirement will be made to overflow into the suppression pool so that the fueling machine area on top of the reactor block will remain dry and functional. The lowest floor of the primary containment is occupied by the suppression pool which will condense the vapour releasing into V1 in the event of a LOCA, thus mitigating the pressure rise in the primary containment.

The space in the annulus between the primary and secondary containment shells will be utilized to house the equipment required for cooling and/or chemistry control of the IC pool, end shields, GDWP, suppression pool, etc.

A single on-power fueling machine will approach reactor core from top. A corridor in V1 area will enable this fueling machine to traverse between top of reactor core and temporary spent fuel storage provided in the primary containment. The primary and secondary containments will be connected to the outside through a set of personnel and equipment access air locks.

5.0 Engineering Objectives

Several engineering objectives are[3] identified in line with the current trends in nuclear reactor design. The emphasis is on simple, economic, reliable and safe system design. Some important engineering design features of the AHWR are discussed in subsequent paragraphs.

5.1 Simple Design

Every system, each system component and individual feature of a component must justify its existence to be a part of the AHWR. All the components shown in the accompanying figures have undergone and passed this rigorous scrutiny. Many major components like reactor outlet header, secondary steam generators, etc. have been eliminated as a result of this scrutiny. The primary coolant circulation pumps too have been eliminated. Also, the moderator-process water heat exchangers have been eliminated and the entire 48 MW of moderator heat will be utilized to preheat the feed water.

In another effort to make AHWR a simpler and safer design, the length of large diameter pipes in primary heat transfer system has been kept minimum to minimize the probability of a loss of coolant accident (LOCA) caused by large pipe rupture. These large pipes are located well above the core, so that even in case of a large pipe rupture the core submergence is ensured.

5.2 High Pressure Heavy Water Absence

Perhaps the most important gain of the above scrutiny is elimination of high pressure heavy water (D2O) system. Although the AHWR design could be viewed as an extension of the Pressurized Heavy Water Reactor (PHWR) technology, a far-reaching deviation from the PHWR design is the substitution of light water in place of heavy water as reactor coolant.

That leaves heavy water only in the moderator system; drastically cutting down the heavy water inventory in the plant. Moderator's is a low pressure system without any interface with fueling machine. Because of all these reasons, loss of precious heavy water due to system leakage will come down at least by an order of magnitude from PHWR figures, significantly bringing down the plant operating cost.

5.3 Conventional Equipment

An important advantage of assigning the safety function to the passive systems and adopting the 4X50% policy is the relaxation of performance demand on many system components including power supply, pumping and heat transfer equipment. These equipment no longer need be guaranteed of uninterrupted operation or be ever-ready to cut into operation. Many equipments would be moved away from high radiation field and will be easily accessible for routine break down or preventive maintenance without affecting reactor operation. In other words, for many of these nuclear applications it will be possible to employ conventional equipment routinely used and easily available in industry. This will bring down the equipment cost as well as the procurement time.

5.4 Maximum Shop Work

AHWR systems will be subdivided into suitable units/ components to minimize site fabrication and installation effort. This will result in speedier construction schedules and shorter maintenance outages. The overall product quality will greatly improve due to better quality control possible in the shop environment. For instance, nearly shop assembled coolant channels will dramatically cut down time needed for their installation and replacement. Instrumentation and control system will be designed to comprise of modules containing maximum details and needing minimum interfacing effort with their counterparts at site.

5.5 Prudent Energy Management

AHWR will utilise the heat generated in reactor shielding and moderator to preheat the feed water returning to the reactor. Heat from these sources will add up to about 50 MW and will contribute significantly to the station electrical output instead of being wasted as in most nuclear power plants. The feed water (FW) will flow through the reactor vault picking up a small gain in temperature. It will then pick-up moderator heat while passing through D2O/FW heat exchangers. Simplified primary heat transfer (PHT) system flow sheet (Figure-3) schematically illustrates how this will be done.

5.6 Passive Safety Features

All important safety functions have been assigned to passive systems. These systems don't depend on external power for their operation and have minimum number of moving parts, if at all. As far as possible, they are directly actuated by an event or a process and do not depend on any instrumentation or control interface. Passive safety approach forms a very dominant undercurrent of AHWR design philosophy.

Provision of thermosiphoning for the primary coolant circulation and a negative void coefficient of core reactivity form the foundation of inherently safe nature of AHWR. In addition, the vapour suppression pool has been retained for energy management in containment during a LOCA. These and a number of other passive safety features incorporated in the AHWR design are described briefly in the subsequent paragraphs.

5.6.1 Pumpless Primary Circulation

The full power primary circulation flow will be obtained by natural circulation caused by thermosiphoning phenomenon. This will eliminate the need for primary circulation pumps as

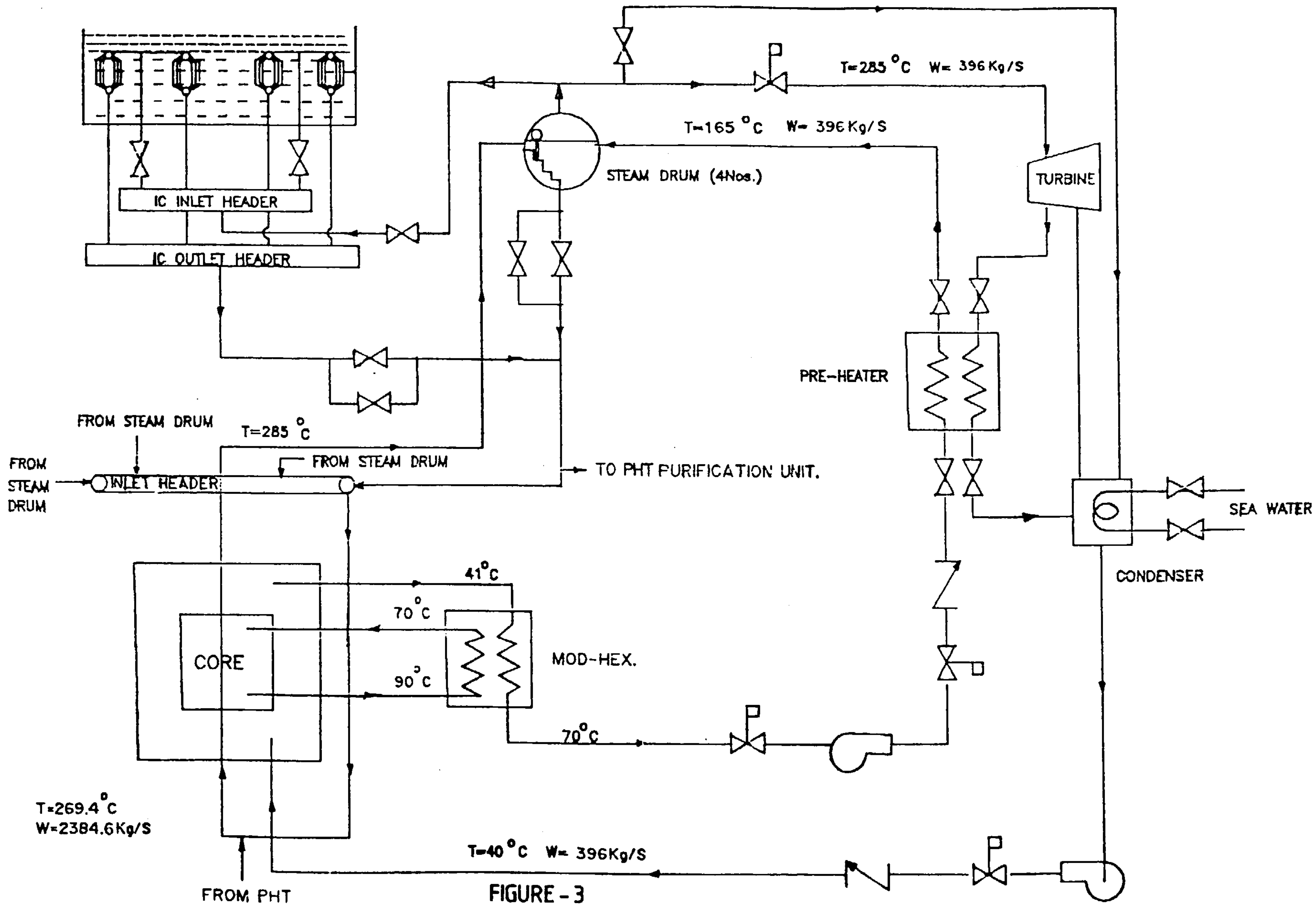

FIGURE - 3

SIMPLIFIED PHT SYSTEM FLOW SHEET

well as a few valves. What is more important is that the core cooling would not have to depend on the external pumping power, moving parts or instrumentation, leading to considerable enhancement of the system safety and reliability. To generate necessary thermosiphon induced driving head, the height of the primary circulation loop (Figure-4) has been kept close to 40 M [3], taking advantage of the reactor building height. An experimental programme has been launched to confirm the analysis leading to the loop height and to study the thermal hydraulic stability of primary heat transfer (PHT) system.

5.6.2 Decay Heat Removal System

The reactor will be cooled by means of thermosiphoning also during a planned shut down and in the event of a scram. In either case, the steam flow from steam drum to steam turbine will be diverted (Figure-3) to a bank of isolation condensers by means of suitable valve operation. The isolation condensers will dissipate the reactor heat to the IC pool water. The heat capacity of the IC pool will be large enough to guarantee steam condensation inside the isolation condensers even in case of several reactor shut downs occurring in quick succession. The height of this thermosiphon loop is about 1.5 times the primary circulation loop. To cool down the coolant system to well below 100 degrees Celsius, a pumped circulation system has been provided. Water will be forced through the reactor core and a set of heat exchangers in such a situation.

5.6.3 Containment Cooling System

Passive containment coolers (similar to isolation condensers in working principle) located in the IC Pool are provided for long term cooling of the containment atmosphere in case of a LOCA. Like the suppression pool, these passive containment coolers (PCC) will help mitigate the pressure build up in the primary containment, especially the volume V1. Simple experiments [4] have demonstrated feasibility of passive containment cooling system and more detailed experiments are under assembly. While this route would enable heat removal in a passive way, use of regular containment coolers is retained.

The high enthalpy steam released from PHT system into V1 will rise to the primary containment ceiling along with some non- condensable gases and enter the primary side of the passive containment coolers. From here the condensed steam will flow down to the GDWP in the form of water while the non-condensables will be directed to V2 through the suppression pool. It may be noted that this entire circuit including PCC primary side internals is an integral part of the primary containment and will be designed accordingly. Heat capacity and evaporation will keep the temperature of the IC Pool water down for a long time but eventually the operator can switch on the IC Pool cooling system consisting of industrial grade heat exchangers, pumps, valves and interconnecting piping.

5.6.4 Emergency Core Cooling System

Accumulators storing water under high pressure nitrogen will cool reactor core shortly after a break in the PHT system. A series of rupture discs act as isolating interface between the accumulators and the reactor core. These discs rupture when post LOCA depressurisation of PHT system reaches a pre-set level. This water gets sprayed on the individual fuel pins as well as the coolant tube through eight perforated tubes running along the of periphery of fuel cluster coolant tube to provide efficient emergency cooling.

Accumulators and connected piping will permit large initial flow to reactor core. The flow will diminish with time in accordance with core decay heat rate. Suitability of vortex based fluidic flow control device is being evaluated for this application. Alternately, branched feed pipes with strategically located flow control orifices will be designed to achieve desired flow

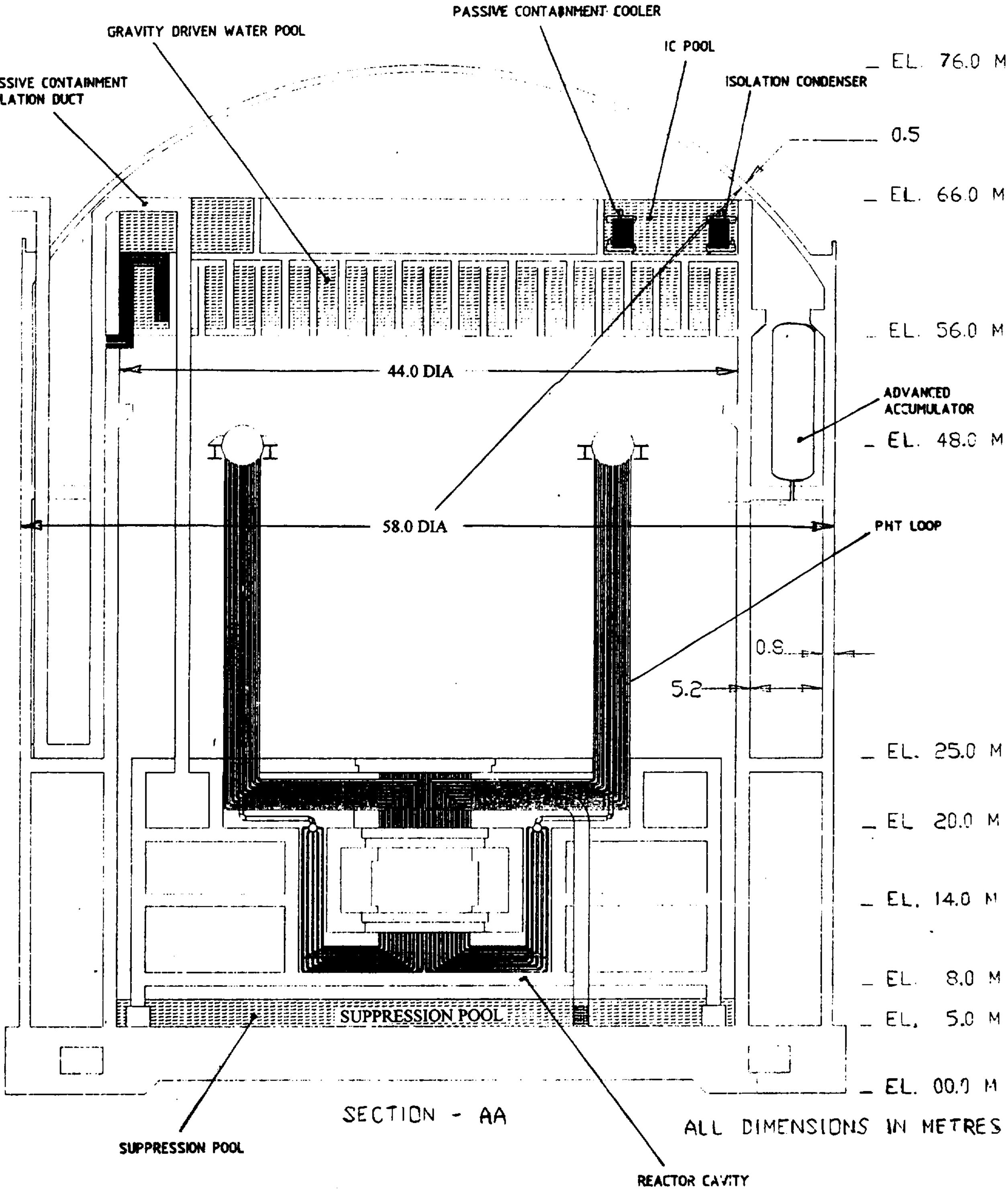

FIG. - 4 AHWR REACTOR BUILDING ELEVATION variation over time. This feature will make the accumulator inventory last for long while ensuring adequate core cooling.

5.6.5 Gravity Driven Water Pool

The GDWP is used for core injection by gravity when PHT system pressure is reduced close to the containment pressure in the aftermath of a LOCA. The GDWP inventory will be adequate to provide core cooling for three days following a LOCA after which the pumped circulation from suppression pool to the core can take over. The GDWP inventory too will be connected to the core through a series of rupture discs and will not involve use of external power, moving parts or instrumentation.

The GDWP supports yet another passive safety function. It provides water inventory required for passive isolation of the containment building.

5.6.6 Core Submergence

The water spilling out from the PHT system, accumulators and the GDWP, as a sequel to a LOCA, will be guided to collect in the space called reactor cavity to submerge the core. After submerging the core, any excess of this spillage will be directed to the suppression pool to prevent unwanted flooding of other areas of the reactor building. To facilitate good communication between the water accumulated in the reactor cavity and the reactor core, a number of vent valves could be provided on the submerged portion of the PHT system e.g. on the inlet header.

5.6.7 Suppression Pool

The suppression pool is located on the lowest floor of the primary containment building. In the event of a LOCA the steam releasing from PHT system into V1 volume will be directed into V2 volume through the water inventory of the suppression pool through a large annulus of more than 100 square metres area. Suppression pool condenses the steam and contains the pressure rise in the reactor building. The temperature of the suppression pool inventory would rise to store in heat form the energy releasing from the PHT system. Special mixing arrangements would be provided to ensure participation of maximum fraction of suppression pool water inventory in heat absorption.

5.6.8 Passive Containment Isolation

In a post LOCA scenario, considerable radioactivity might be expected to be released in the containment building. To protect the population at large from exposure to this radioactivity, the containment must be isolated within specified time of initiation of a LOCA.

To achieve this containment isolation function in a passive manner, the reactor building air supply and exhaust ducts will be shaped in the form of U-bends of sufficient height (Figures 2 & 4). In the event of a LOCA, the V1 gets pressurised. This pressure acts on the GDWP inventory and pours water by swift establishment of a siphon, into the ventilation duct U-bends. When appropriately filled, these act as water seals between the containment and the environs, providing necessary isolation between the two. Drain connections provided to the U-bends will permit to re-establish the containment ventilation manually when desired. A feed line running from the GDWP to the U-bends will also permit manual establishment of containment isolation. Our experiments have demonstrated that this system can be designed to respond to small containment pressure rise and that it can respond fast in case of a major accident.

Passive containment isolation devices have been incorporated in the design in addition to regular isolation features like motor operated isolation dampers.

6.0 Coolant Channels and Fuel Elements

The current AHWR design envisages uniform coolant channel design for all lattice locations. The coolant channels will be largely shop fabricated for better quality control, speedier installation and faster replacement. The coolant channel design provides for possible replacement of pressure tubes, or the entire coolant channel including the calendria tube. This will extend the reactor life with minimum outages and man-rem cost.

AHWR fuel cluster (Figure 5) will have 52 fuel pins arranged in a square array. To be consistent with the basic design objectives of thorium utilisation and negative void coefficient of reactivity, the cluster shall incorporate MOX as well as ThO2 fuel pins. The MOX pins in this cluster will be surrounded by ThO2 pins to place the former in a harder neutron spectrum with a view to obtaining reactivity enhancement. Both types of the fuel pins will be clad with zircaloy-2. The fuel pins will be supported at regular intervals along their length by suitable spacers. The fuel elements will be designed to give a burn up in the vicinity of 20000 MWD/Te of heavy metal to begin with.

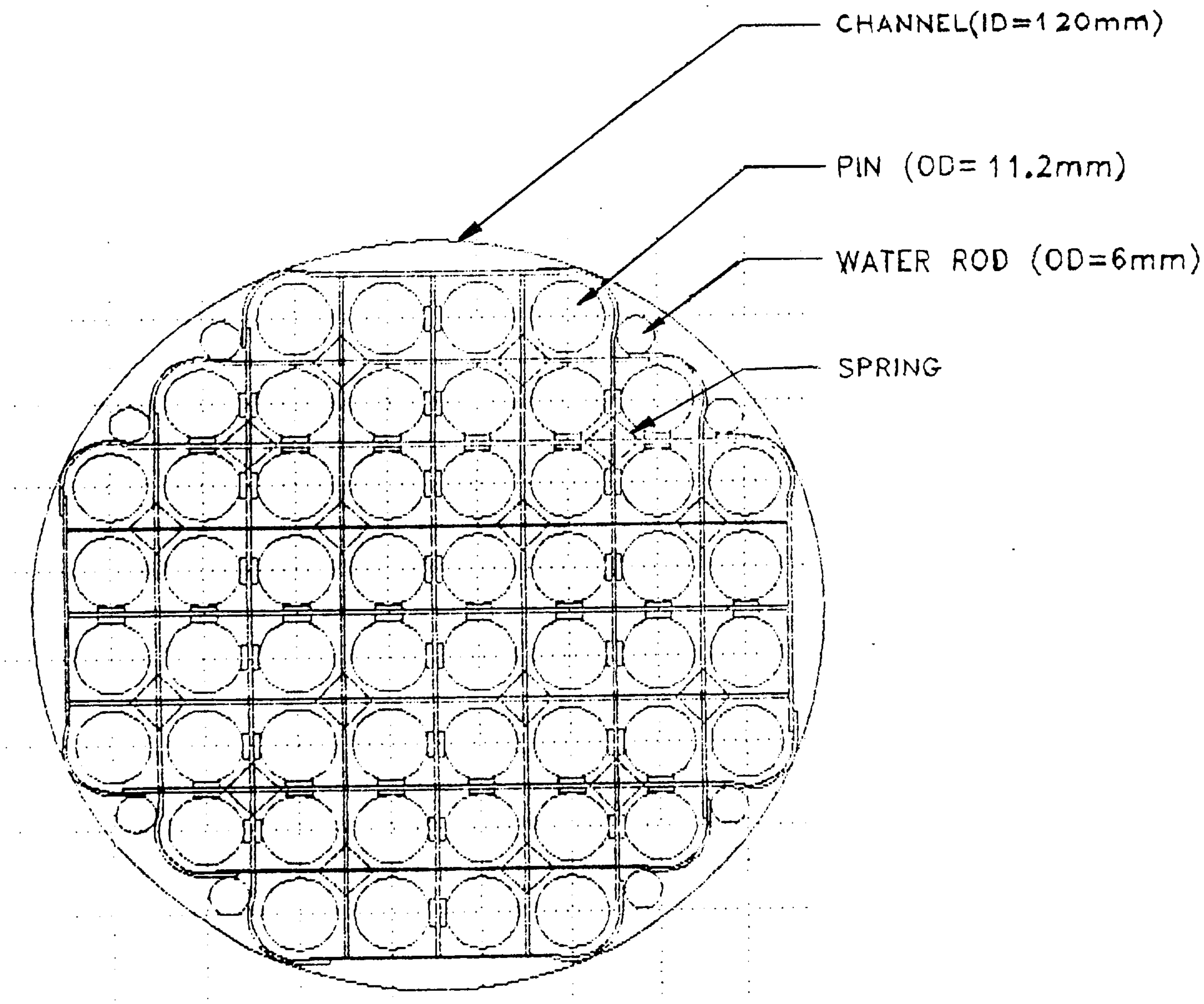

FIGURE - 5

AHWR FUEL ELEMENT

A special feature of the fuel cluster is a provision to direct the emergency core cooling flow to the individual fuel pins and to the coolant tube. Eight perforated tubes border the fuel pins and run along the entire length of the cluster so as to direct the water jets at different elevations and from different directions.

7.0 Concluding Remarks

The design of the reactor is being worked out in such a manner that the electrical output is similar to the present day 220 MWe PHWR plants so that maximum advantage can be taken from the existing design.

Much more development work and analytical studies are required before fully evolving any reactor design. However, it is clear that a basic heavy water reactor offers many possibilities which can be exploited to meet our evolving needs and at the same time satisfy concerns that advanced reactor designs may have to satisfy in years to come. The key to success lies in our ability to do this without losing economic competitiveness of the system or if possible, even improving it. The contention is that this can very well be done in view of the design simplifications leading to elimination and simplification of several systems and components as also the reduction of heavy water inventory and leakage.

Acknowledgement

The authors wish to express their sincere gratitude towards Mr. Anil Kakodkar, Director, Reactor Design and Development Group for continuous inspiration and guidance in the design of AHWR.

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CANDU IMPROVEMENTS

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Abstract

The evolution of the CANDU family of nuclear power plants is based on a continuous product development approach. Proven equipment and system concepts from operating stations are standardized and used in new products.. Due to the modular nature of the CANDU reactor concept, product features developed for CANDU 9 can easily be incorporated in other CANDU products such as CANDU 6. Design concepts are being developed for advanced CANDU 6 or larger advanced CANDU, depending on the number of fuel channels and the fuel cycle selected.

This paper provides a description of the design improvements being incorporated in CANDU 9 and further design enhancements being studied for future incorporation in CANDU 6 or larger advanced CANDU meeting the requirements of future CANDU owners. The design enhancement objectives are:

- To improve operational simplicity by applying modern information technology.
- To improve safety in a cost effective way.
- To improve system and component reliability and to increase plant life.
- To improve economics and to reduce owners' risks during all phases of a project using up-front licensing, an improved engineering process and project tools during design, construction and operation.
- To continue to exploit the neutron economy of CANDU with the development of advanced fuels and fuel cycles.

1.0 Introduction

The evolution of the CANDU family of nuclear power plants is based on a continuous product development approach. Proven equipment and system concepts from operating stations are standardized and used in new products. The three 700MWe CANDU units being built in South Korea are the updated versions of CANDU 6 that are operating in three different countries; the CANDU 9 design that is currently under licensing review in Canada is the single unit adaptation of the 900MWe class CANDU with 12 units currently operating in Canada. The operating 900MWe units are integrated 4 unit stations and the latest of these are located at Darlington with all 4 units in operation in 1994. Due to the modular nature of the CANDU reactor concept, product features developed for CANDU 9 can easily be incorporated in other CANDU products such as CANDU 6. Design concepts are being developed for advanced CANDU 6 or larger advanced CANDU, depending on the number of fuel channels and the fuel cycle selected.

The design of future CANDU stations will continue to evolve, building upon the success of current operating units. Progressive improvements and enhancements will continue to be made to the CANDU system with heavy water moderated, pressure tube reactor technology of high neutron efficiency, based on the results of advanced technology R&D and operational experience from operating CANDU stations. The development objectives for advanced CANDU design features are guided by the requirements of the operating utilities. The timely incorporation of advanced design features represents an evolutionary approach to the development of the next generation of CANDU.

Most of the requirements [1] for design improvements are based on systematic reviews of current operating CANDU stations in the areas of design and operational reliability, operability, maintainability, and licensability. In addition, there is a continuing pressure to improve safety, partly in recognition of the greater risk associated with the increasing number of plants; continuing improvements will also help to reduce the public concern of risk associated with nuclear power plants.

The next section describes the design improvements that have been developed for incorporation in the current CANDU products. The CANDU 9 design has incorporated most of these improvement features during the design integration engineering. These features can all be adapted on CANDU 6, if the required changes are found to be economic. The final section provides a description of the design improvements or features that are still under development and can be incorporated in the next advanced CANDU designs in the 3 to 5 year time frame. These advanced design improvement developments are guided by the product improvement targets set in the product development plan.

2.0 Current Design Improvements

2.1 Automation with Modern Information Technology

The use of modern advanced information technology in the nuclear control system design will improve operational simplicity. The CANDU 9 control centre design will provide accurate, timely information and advice to reactor operators, therefore enabling them to make correct decisions, which is crucial to the safe and reliable operation of a nuclear power plant. Automating the routine tasks and providing detailed plant status displays will help the operators in the handling of plant upsets. In addition, the control systems are designed with standardized equipment, reducing spare parts and enhancing interchangeability for ease of maintenance to reduce operations / maintenance costs, to reduce occupational exposure and to facilitate repair and replacement of equipment.

A powerful annunciation system is provided for filtering of unimportant alarms, and grouping and prioritizing in the display of key alarm messages. The system is capable of providing multiple prioritized alarms with multiple set points (e.g. first warning, significant deviation, reactor setback imminent etc.) with pre-set filtering features. The CANDU 9 annunciation system has been designed to provide a fast, user friendly procedural action follow-up aid which will enhance operating staff recognition of events, plant states and required corrective procedural actions.

The CANDU 9 computer systems for control, plant display, testing and communications are provided with a "system health" display for the operator so that the location of a faulted instrument or device can be readily indicated or confirmed. The transfer of the control or communication function to the standby processor on the operations display and the maintenance terminal would be detected by a background diagnostic routine and the faulted device address is scanned by the "system health" routinely to flag that particular device as unavailable. The provision of maintenance diagnostic features will facilitate the rapid recognition and correction of system faults improving system reliability.

2.2 Designs for Improved Operability

Improvement of the man-machine interface is a key factor in reducing the probability of operator or maintainer induced accidents. Human factors engineering meeting the required standards at all stages of the design, is being followed in the plant design process. The systematic analytical approach to system design consists of requirement definition, function allocation, function analysis and task analysis to define operator information and information presentation requirements. The resultant operator display is then verified against the requirements to provide confidence that adequate and correct information are presented to the operator for the required tasks.

The CANDU 9 control centre design, [2] incorporating the full set of advancements to the man-machine interface, is being proof-tested by the construction of a mock-up at AECL's Saskatoon Office. This mock-up allows detailed event simulation, and will allow operator response procedures to be developed and tested before the power plant is built.

Reliability of normal operation is enhanced by advances in control room design. The CANDU 9 control room is based on the well-established single unit panel layout for the CANDU 6. The use of computerized information displays and increased automation of routine tasks, such as safety system testing, maintains the operator in a supervisory role. The risk of errors in normal operating conditions has been reduced by system changes, suggested by feedback from CANDU operators. The operator's role in an abnormal incident is to manage the plant automatic response, rather than to make rapid decisions and control actions. In general, the grace period before the operator is required to respond to an accident is increased from 15 minutes to 8 hours.

In addition, the main control room and the secondary control area are designed meeting the safety grouping and separation philosophy, and the necessary structures and systems have been hardened to ensure sufficient monitoring and control capability following seismic and other external events, as well as accident conditions. As noted above, the operator role is one of managing the event rather than being forced to take rapid action. This also helps to make emergency management off-site into an easier process.

For CANDU 9, safety system computer functions are also enhanced. The CANDU 9 design uses and builds on the application of safety software verification established by the Wolsong 2,3,4 project. Improvements to the automated testing capability successfully introduced in the Darlington power plant will be incorporated, to reduce the operator's burden in the on-line testing of safety system availability.

2.3 Improvements for Operational Safety

The safety design for CANDU 9 systems continues the emphasis on defence in depth and evolutionary improvement to proven concepts that maximizes confidence in their safety to the public. The requirements deal with the complete range of operating conditions from normal operation to low probability events such as severe accidents.

In a CANDU reactor, the probability of severe core damage is very low, because of two special shutdown systems and because of the separation of high temperature - high pressure reactor coolant from low temperature - low pressure moderator. Should a severe accident progress to the point of failure of core components, the presence of this shield water is capable of removing the heat from the molten core and preventing contact of the failed core with the containment boundary. The design features such as moderator piping nozzle location change to improve moderator circulation and a calandria tube with optimized heat transfer characteristics, will reduce the probability of a large release to even lower levels. The design and manufacturing of the new calandria tube features high emissivity and low ridges for heat transfer enhancement. The ability to provide makeup to the moderator and the end shield will improve the reliability of this heat sink.

CANDU 9 heat removal capability for accidents is increased to add redundancy and to increase design margin. The pressurizer is increased in size, to cater to complete reactor coolant system depressurization and cooldown, from the full power operating condition to the zero power cold condition. In addition, the design of the emergency core cooling (ECC) system, which is based on the single-unit system design used on the Wolsong plants, includes enhancements designed to increase reliability, and to provide further performance margin via faster coolant injection. All process components, except for the pressurized gas tanks and recovery pumps, are located inside the reactor building, effectively increasing the injection pressure available during the intial core refill.

CANDU 9 incorporates a high degree of redundancy in heat sink capability, to ensure both a low Severe Core Damage Frequency (SCDF) and to ensure two redundant means of heat removal for any external event such as earthquake etc. In addition to the emergency core cooling and moderator heat removal capability for LOCA events, the design includes a fully-qualified high-pressure feedwater system, and in addition, the separate shutdown cooling system is capable of heat removal from the the hot standby condition immediately after shutdown. In addition to these heat sinks, a passive, gravity-fed emergency water supply to the steam generators is provided through the reserve water tank.

The CANDU 9 design includes a large, high-level reserve water tank inside the reactor building. This is a backup cooling water reserve, in addition to its role in supplying medium pressure emergency core cooling. The reserve water tank provides make-up water to the steam generators, and also to the reactor shield cooling tank, which gives a continuing core heat removal capability in the event of a loss of normal cooling.

Relatively easy access to the reactor building during plant operation has been a traditional CANDU advantage, and by careful attention to segregation of higher activity water vapour from lower activity areas, this can be retained, while achieving the target of less than one Sv/year occupational dose for CANDU 9. A higher performance drier with reliable performance is being qualified for use in a reactor building environment to reduce the spread and emission of heavy water vapour.

2.4 System Reliability And Service Life Improvements

For equipment that is designed to be replaced during the life of the plant, planned replacement considerations are explicitly factored into the CANDU 9 design and allowances made in the plant construction to accommodate these requirements.

The CANDU 9 fuel channels are fitted with improved spacer design that is a tight fit on the pressure tube so that it will not displaced. Spacers at early CANDU units moved during construction commissioning, which allowed some pressure tubes to sag into contact with the cooler calandria tubes surrounding them. To avoid such pressure tube and calandria tube contact which can result in brittle hydride blisters and potential failure, displaced spacers are being relocated in some operating reactors.

In addition, pressure tube specifications for CANDU 9 will specify significantly reduced hydrogen concentration and increased fracture toughness by careful control of the ingot melting practice. Intallation procedure has been modified to put pressure tube back end at the fuel channel inlet rather than the outlet, to significantly reduce the rate of diametral strain. This reduces the undesirable effect of flow bypass around the fuel bundle as the pressure tube creep increase towards the end of design life.

CANDU 9 also features a simplified emergency core cooling system with improved reliability, by the use of one way rupture discs and a floating ball inside the high pressure injection water tanks. This drastically reduces the number of injection valves that are required to operate during accident conditions.

2.5 Improved Engineering Tools

CANDU 9 design engineering utilizes advanced engineering tools such as 3-Dimensional Computer Aided Design (CADDS) tools for better design quality and lower engineering costs. The 3D CADDS model is used to decide the layout configuration, optimization of the fabrication sequence and construction, and the choice of composite steel or structural steel modules depending on the layout and complexities of systems.. The ability to thoroughly check the design before construction starts will avoid costly rework and delays during construciton

Design and other engineering document production is another area where the implementation of advanced information technology will result in a enhanced document production process. A program has been developed which allows designers to have access to a common data catalog containing all the key design data values used for the technical documents. This not only allows better control of design changes, it also assists in providing consistent control data through document text linking, avoiding transcription errors.

3.0 Further Enhancements for Next Generation CANDU

3.1 Flexibility To Meet Future Needs

The designs of advanced reactors must provide the flexibility of using other advanced fuel cycles that take advantage of the neutron economy of heavy water reactors without design concept configuration changes. This operational flexibility will provide utilities of different backgrounds the opportunity to optimize resources utilization, to reduce uranium requirements and to minimize refuelling costs.

With the natural uranium fuel cycle, the advanced CANDU systems will utilize improved fuel bundle design that provides reduced fuel element rating, enhanced thermal-hydraulic characteristics, and improved operating margin. The designs of advanced CANDU will provide the flexibility of using other fuel cycles such as thorium cycle to exploit the large indigenous thorium resources in some countries, or recovered uranium, which contains 0.9% U235, a by-product from Light Water Reactor(LWR) spent fuel re-processing.

Due to the synergism of CANDU and LWR, the LWR can be viewed as an efficient source of fissile material for future advanced CANDU via a number of potential once-through combined fuel cycles (by re-using unseparated Uranium and Plutonium), one of which, DUPIC (direct use of LWR spent fuel via dry processing) is currently being investigated in a joint program between Canada-Korea-U.S. Also the use of transuranic mix from spent fuel reprocessing can help to reduce long lived waste

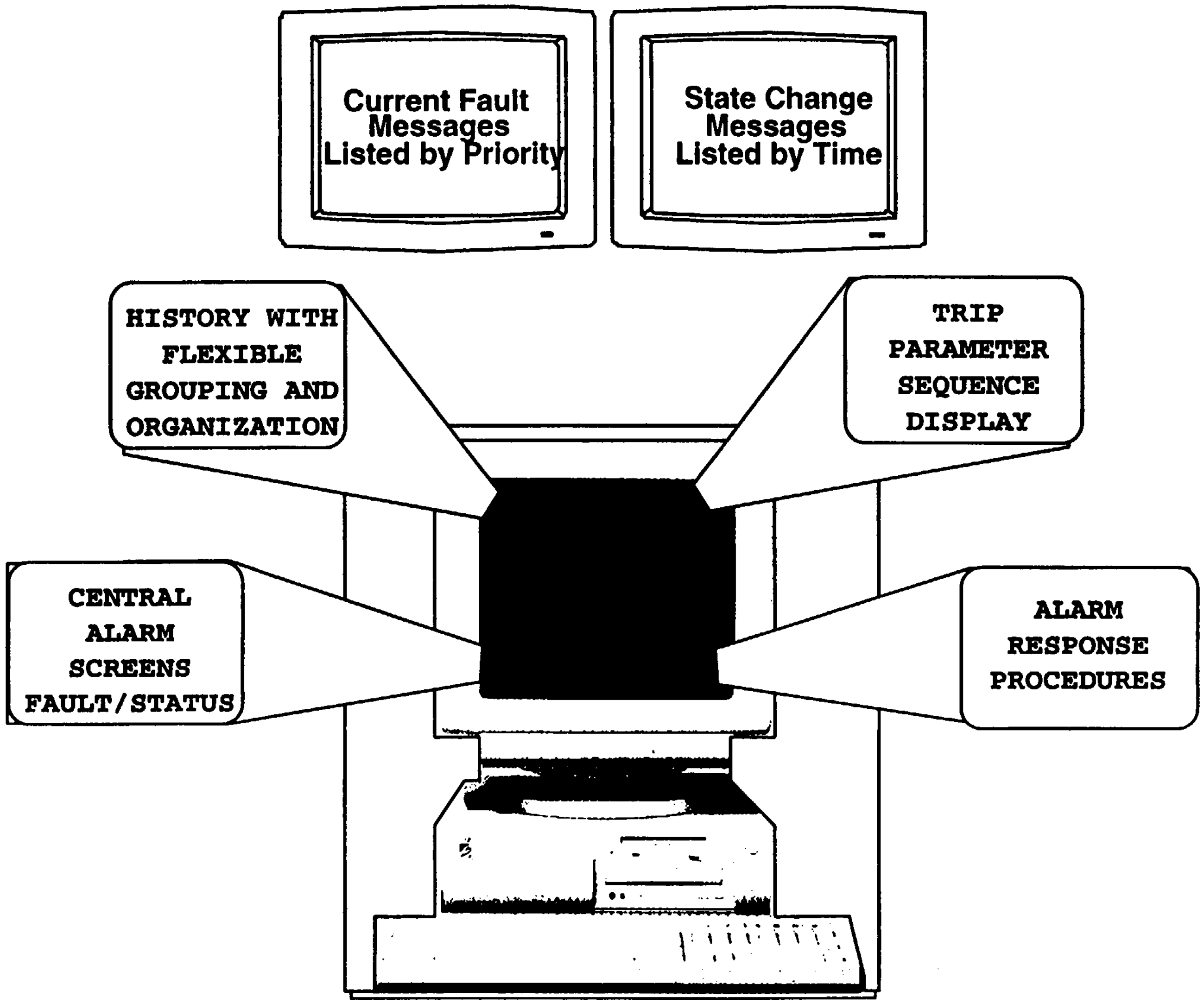
3.2 Improved Fuel Channel Assemblies

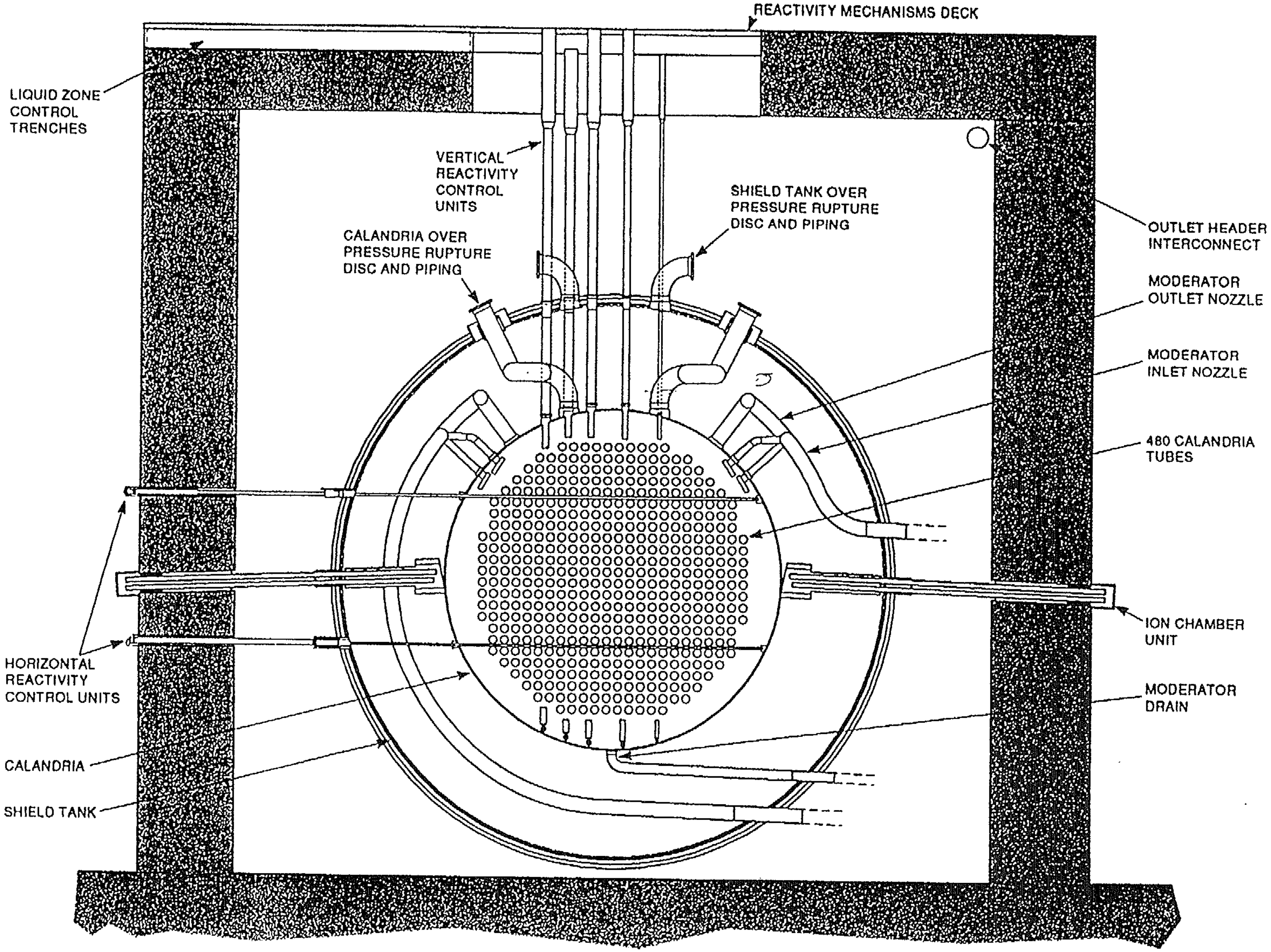
Design and development testing are proceeding on a calandria tube design that has a higher resistance to sag. One possible design is to use a calandria tube with thick ends in order to optimize sag resistance without severe fuel burn-up penalty. Other design improvements being considered are: redesign of the fuel channel components to allow optimum use of automated tube replacement tooling which will result in a cost and schedule reduction for a retubing operation.

3.3 Reduce Construction Costs And Schedule

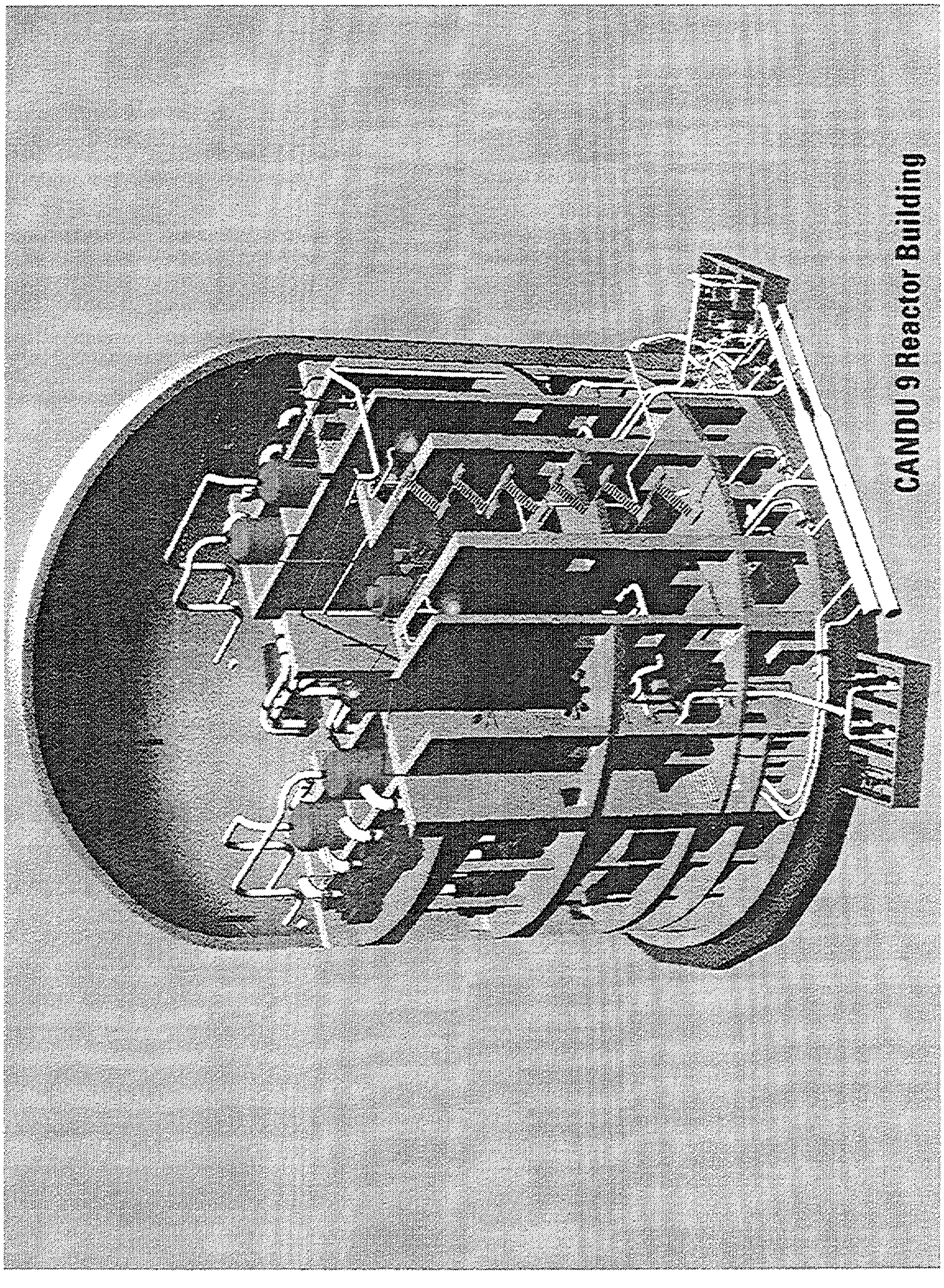
The use of advanced construction methods such as modularization will reduce the construction schedule. The interest during construction is a major component in the plant economics. R & D on advance structural material such as high performance concrete and composite structure construction methodology are required to obtain underlying structural design performance.

CAMLS - Interface





Reactor Structures Assembly – Front View



CANDU 9 Reactor Building

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SAFETY AND LICENSING ASPECTS

(Session 3)

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TRENDS IN CANDU LICENSING

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Abstract

To date the licensing experience with CANDU has been relatively good, with plants successfully licensed both domestically and in other countries. Nevertheless it is now clearly recognized that the utilities need more formal evidence that the licensing risk is low, if they are to proceed with new nuclear power projects.

In response AECL pioneered "up-front" licensing. The 450 MWe CANDU 3 product was executed as a Standard Product Design, with the intention of completing all the conceptual design and most of the detail design before project commitment. The Canadian nuclear regulator, the Atomic Energy Control Board (AECB), was invited to begin licensing review at the early stages of the CANDU 3 conceptual design. A series of licensing milestones was defined as follows:

1. Licensing Basis
2. Identify Potential Problems
3. No Major Problems Foreseen, and
4. Accept Standard Design

Currently Milestones 1 and 2 are almost complete. Milestone 3 requires resolution of all AECB-identified issues and actions, and when complete, provides the equivalent from a design point of view of a construction license for the standard design. When a site is selected, only site specific licensing will be required in order to obtain the actual construction license. From a design point of view, Milestone 4 is the equivalent of the operating license for the standard design.

The more recent CANDU 9 programme is centred on an evolutionary design of approximately 900 MWe. It is based on already-operating CANDUs, with improvements in the areas of constructability, economics, plant layout, operations and maintenance, and safety. Even so, both domestic and foreign potential users require evidence of licensability in Canada as of "today". An "up-front" licensing process, using a modified CANDU 3 model, has been established to give this assurance. For CANDU 9, this involves a two-year intensive review by the AECB, which will end in January 1997.

The documents submitted for review encompass all the safety design requirements, design requirements and design descriptions of the major systems, a preliminary Safety Analysis Report, a preliminary Probabilistic Safety Assessment, a Human Factors Programme Plan, the procedures for producing safety-critical software, and others. The review scope includes comparison of the design and safety assessment results with current licensing requirements. Each document, or group of documents, receives a formal written response from the AECB, so that both parties can measure the progress. An agreed dispute resolution mechanism involving

senior managers and Executives of both AECL and the AECB kicks in when there is disagreement on major items. The review will be complete once there is a finding that there are "no fundamental barriers" to licensability in Canada. This is sufficient to allow foreign regulators to proceed with detailed licensing with confidence, and therefore assures utilities in Canada or overseas that they can proceed with a project with minimal licensing risk.

1. *Introduction*

Modern utilities view nuclear power more and more as a commodity—it must compete "today" with current alternatives to attract their investment. With its long construction times and large capital investment, nuclear plants are vulnerable to delays once they have been committed. Delays can come from a number of sources - labour strikes, technical "surprises", poor project management, intervention by special-interest groups, and regulatory licensing. For nuclear power to succeed in this era of cheap fossil fuels, each of these sources of risk must be countered. Reduction of licensing risk for CANDU is the subject of this paper.

There are two related issues. Where the purchaser and the regulator are experienced in CANDU, the thrust is a very practical one: to identify and resolve major licensing risks at a very early stage in the project. Thus for a Canadian project, the designer (AECL) and the prospective purchaser would deal directly with the AECB. However CANDU has also been successfully licensed in other countries, including Korea, Romania, Argentina, India and Pakistan. Each of these countries has its own regulatory agency responsible for licensing the plant. In addition, however, the foreign customer and regulator may seek input from the AECB, up to and including a statement of licensability in Canada; this is not normally needed for a "repeat" plant and/or if the customer is experienced in CANDU, but can be requested if the plant configuration has been modified significantly from an already-operating CANDU. It is thus the responsibility of the designer to initiate early discussions with the AECB so the foreign CANDU meets the expectations of its customers.

2. *Canadian Licensing Process*

The licensing process followed for many years in Canada[1] has three formal steps, similar to those in other countries. First comes Site Acceptance, initiated by a formal Letter of Intent to the AECB. From the regulatory viewpoint, Site Acceptance confirms the suitability of the site for a nuclear power plant, whose conceptual design is presented at this point. More recently, an environmental assessment is required as part of Site Acceptance to ensure public knowledge of the proposal and to confirm that the environmental effects are acceptable. The next step is Construction Approval, supported by a preliminary design and safety analysis. Key documents reviewed at this stage include all the design requirements documents (for example, Safety Design Guides) and the Preliminary Safety Report. The third step is granting of an Operating Licence, supported by the detailed design documentation, a Final Safety Report, a commissioning programme, operating policies and principles, emergency plans, authorization of operators, etc.

The Canadian licensing philosophy places the fundamental responsibility for safety on the plant owner, with the AECB acting in an audit role. The AECB sets overall safety objectives which the design must satisfy, but to a large extent leaves the means to satisfy these objectives up to the designer. It does of course exercise independent and informed judgement as to whether the design will meet the overall safety objectives with reasonable confidence. This approach has two benefits: it encourages innovation, and avoids the potential conflict of the regulator sitting in judgement on its own design rules. On the other hand, there is a potential for licensing delays and/or added costs if there is a misunderstanding of what the requirements mean, or what constitutes an adequate method of meeting them. However past U.S. experience has shown that highly prescriptive regulatory design requirements do not obviate this problem, especially when the requirements are subject to rapid change and escalation.

To address this uncertainty, a better process was sought that would give more assurance to owners of a smoother licensing process, and to regulators that requirements would be well-defined up-front so that the potential for misunderstanding is reduced. These concepts, although initiated on a proposed second unit at the Lepreau site in New Brunswick, came to fruition with the CANDU 3 Standard Product Design programme, described in the next section.

3. CANDU 3 Experience

CANDU 3 is a small (450 MWe) evolutionary single-unit plant aimed at utilities with small electrical grids and "first-time" nuclear buyers. It is a Standard Product Design. The goal was to do all the conceptual and most of the detailed design, and obtain assurance of licensability, before a site-specific project was committed. This will give an assurance of low risk to potential customers.

Reduction of licensing risk was addressed from the beginning. AECL formally asked the AECB to engage in pre-licensing of the CANDU 3 Standard Product Design[2 3], and submitted documents describing the design to date and how we thought the licensing process could proceed.

Milestones

AECL and the AECB agreed that pre-licensing of the CANDU 3 design would use a process of four Milestones, listed in Table 1 and described as follows.

1. Licensing Basis

This first Milestone defined the basis for the licensing requirements, and required the submittal of fourteen documents:

- a Licensing Basis Document (LBD), the top level document which identifies, among other things, the codes, standards and regulatory documents which will govern;
- twelve Safety Design Guides (SDGs), covering a number of topics such as tornadoes, fire, earthquakes, etc. which designers use to ensure proper and systematic application of safety in design;
- a systematic review of the plant design to identify all possible initiating failures, as required by AECB document C-6[4] which defines how safety analysis shall be performed.

Achieving Milestone 1 requires approval of all documents by the AECB, and is conditional on a number of future activities:

- AECB acceptance of Safety Analysis Basis (SAB) documents, which describe the assumptions and methods to be used in safety analysis, and
- consideration or incorporation of any new and significant findings from safety-related Research and Development (R&D), safety analysis or experience from operating plants.

The SABs are formally required for Milestone 2, but the AECB has tied Milestones 1 and 2 together through acceptance of the SABs

2. Identify Potential Problems

This milestone is essentially achieved by AECB Staff review of the design, as described in a Technical Description. Most of the CANDU 3 design is a straight-forward evolution from CANDU 6s operating throughout the world. AECB review concentrated on new design features which represent simplification over operating CANDUs, in particular:

- process and safety systems grouping philosophy
- unidirectional core flow and single-ended refueling
- steel-lined containment without dousing
- distributed control system, and
- computerized safety systems.

When AECB Staff are satisfied they have identified all significant design and safety issues, Milestone 2 is achieved. The safety issues are reviewed through a compliance document, prepared by AECL, which describes the methods used to show compliance with three key safety regulatory documents R-7[5], R-8[6], R-9[7] and the SABs.

3. No Major Problems Foreseen

During the process of achieving Milestones 1 and 2, AECB staff opened actions against the CANDU 3 project to provide more design information or safety analysis. When these actions have been addressed to the satisfaction of the AECB, by more information, or if necessary, by design changes, Milestone 3 is achieved.

There is still discussion on the amount of safety analysis required. AECL expected that a "conceptual" report of analysis, covering the traditional cases which have affected CANDU design, will suffice. AECB staff have indicated that if the analysis is sufficient, achieving Milestone 3 will also represent the equivalence of a construction license for the design. When a site is selected for a CANDU 3, only site specific licensing would be required in order to obtain the actual construction license.

4. Accept Standard Design

AECB staff need to do a thorough review of the detailed design and safety analysis in order to achieve Milestone 4. Milestone 4 is the equivalent of the operating license for the standard design. All actual operating requirements, such as operator training, would have to be reviewed and approved by AECB staff for a real plant.

Progress to Date

Currently Milestones 1 and 2 are almost complete. All documents, except for a number of SABs, have been submitted and reviewed. Almost all of the documents for Milestone 1 have been approved.

The conceptual reports, for accident analysis and probabilistic analysis, have been submitted for Milestone 3. AECB staff review resulted in a number of actions against the CANDU 3 project. AECL has closed a few of these actions but most remain open currently.

For Milestone 4, AECL has produced much of the detailed design description necessary for AECB review, but has not submitted this material yet. AECL's priority was on the first three milestones.

Successes and Lessons Learned

The process worked reasonably well. It allowed AECL to maintain a current and detailed understanding of AECB licensing requirements as they evolved through the 1980s and 1990s. The experience AECL gained from this process was applied directly to the Wolsong 2,3,&4 plants currently being constructed in Korea. They are being licensed in Korea to Korean and Canadian requirements as of December 1989, and used CANDU 3 work on Licensing Milestones 1 and 2.

There were, of course, technical disputes, but the whole idea of the process is to resolve these before the plant is committed and/or being built. In many cases, design changes were implemented or committed to the design to accommodate AECB concerns. Of course, making these changes during design rather than during construction is a benefit to be realized when the first CANDU 3 is built.

In general, the way up-front licensing worked in detail could only be defined by trying to do it. Much was learned on both sides. In that respect the CANDU 3 experience points the way for future up-front licensing initiatives, such as CANDU 9.

Currently, AECL is devoting much more of the design effort to CANDU 9 (see below) and to repeat CANDU 6s. The regulatory review of CANDU 3 has likewise been re-focused to CANDU 9, and AECL's current effort on CANDU 3 is directed towards schedule definition and cost reduction. However the CANDU 9 licensability review was able to start rapidly by using the lessons learned in the CANDU 3 process and by building on the issues identified and resolved on CANDU 3; without the years of work on CANDU 3, this would not have been possible. Similarly, any agreements reached with the AECB on CANDU 9 will benefit the future CANDU 3 design.

TABLE 1. CANDU 3 UP-FRONT LICENSING IN CANADA

Milestones and Deliverables

Milestone 1	Milestone 2	Milestone 3	Milestone 4
Licensing Basis	***Identify Potential Problems***	***No Major Problems Foreseen***	***Accept Standard Design***
Licensing Basis Document	**Technical Description**	**Conceptual Safety Report**	**General Safety Report**
12 Safety Design Guides	**Compliance with R-7, R-8 and R-9**	**Conceptual PSA Report**	**Detailed design**
Systematic Plant Review	**Safety Analysis Bases**	**Resolve Major Problems**	

4. *CANDU 9—Licensability in Country of Origin*

The more recent CANDU 9 programme is centred on an evolutionary design of approximately 900 MWe. It is based on already-operating CANDUs (Bruce-B and Darlington), with improvements in the areas of constructability, economics, plant layout, operations and maintenance, and safety. Even so, both domestic and foreign potential users require evidence of licensability in Canada as of "today". An "up-front" licensing process, using an improved CANDU 3 model, has been established to give this assurance. For CANDU 9, this involves a two-year intensive review by the AECB, which will result in their first report in the summer of 1996 and the final report in January 1997. Specifically the AECB is asked for an informed opinion on the licensability of CANDU 9, as an input to the detailed licensing review by a foreign regulator.

CANDU 9 is not currently viewed as a Standard Product Design. Should a CANDU 9 be sold overseas, the foreign regulator would award a construction licence, not the AECB. Thus much of the detailed licensing would be done overseas after the commitment of the project. The CANDU 9 licensing objective is therefore to get assurance from the AECB, before a project is committed, that CANDU 9 would be licensable in Canada; and that this assurance is in sufficient detail, with sufficient work done to back it up, that a foreign regulator could proceed with confidence. The finding by the AECB should therefore be one of "no fundamental barriers" to licensability in Canada, equivalent to Milestone 3 on the CANDU 3 SPD programme.

The document submission schedule to the AECB ensures that *requirements* are submitted (and agreed) first, followed by the description of how these requirements are *implemented* in design and satisfied in safety analysis. This organization makes it possible to do the review in a two year period, since it ensures that we are not still debating requirements after the detailed design is well advanced.

A summary of the document submission schedule follows. All major documents will have been submitted by mid-1996, allowing time for AECB Staff review and their first report, and resolution of any outstanding issues before the final AECB report in January 1997.

TABLE 2 - HIGH LEVEL DOCUMENT SUBMISSION SUMMARY

Documents	Submission to AECB
FAMILIARIZATION	
Licensing Plan	**September 1994**
Technical Description Rev. 0	**February 1995**
REQUIREMENTS DOCUMENTS	
Licensing Basis	**January 1995**
QA Programme	**March 1995**
Safety Design Guides	**March - May 1995**

Documents	Submission to AECB
Systematic Plant Review Methodology	April 1995
Safety Critical Software Standards and Procedures	May 1995 - December 1995
Compliance with Regulatory Documents	May 1995 - August 1995
Human Factors Engineering Programme Plan	June 1995
Safety Analysis Initial Conditions and Standard Assumptions	August 1995
Probabilistic Safety Analysis Methodology	August 1995
Design Requirements for Safety Related Systems	September 1995
Distributed Control System Software Requirements	October 1995
Disposition of Generic Licensing Issues	December 1995
Severe Accident Programme	February 1996
DESIGN DOCUMENTS	
Flowsheets for Safety Related Systems	September 1995 & February 1996
Overpressure Protection Summary Reports for Safety Related Systems	September 1995 & February 1996
Technical Description Revision	January 1996
Safety Critical Software Sample Implementation	March 1996 - September 1996
Distributed Control System Sample Implementation	March 1996 - September 1996
SAFETY ANALYSIS DOCUMENTS	
Exclusion Area Boundary	September - December

Documents	Submission to AECB
	1995
Accident Analyses	October 1995 - July 1996
• large LOCA	
• small LOCA	
• LOCA with loss of Emergency Core Cooling	
• LOCA with impaired containment	
• small LOCA	
• loss of flow	
• loss of reactivity control	
• single channel events	
• steam & feedwater line breaks	
Probabilistic Safety Assessment (Level 1)	December 1995 - July 1996

Some particular requirements documents warrant attention:

- The **Licensing Plan** records the deliverables to be submitted to the AECB, and the submittal schedule. When agreed to by the AECB, along with a review schedule, it acts as the road-map through the licensing process.
- The **Licensing Basis** is a high-level listing of the major licensing requirements. It calls up the appropriate regulatory documents and codes and standards, and interprets, in case of ambiguity, how the licensing requirements will be applied. It includes **both** AECB requirements **and** the requirements of the foreign regulatory body.
- The **Systematic Plant Review** lists the design basis accidents. In Canada the onus is on the designer/owner to determine and justify a complete set of design basis accidents, and assign them to frequency classes using guidance provided by the Regulatory documents. Each frequency class has an associated public dose limit. Clearly this is an area open to much debate with the regulatory agency, and must therefore be established early on.
- **Safety Design Guides** are the interpretation in detail of the requirements of the safety and regulatory documents on the project. They are written by safety experts in AECL, and act as detailed instructions to designers in the execution of their design. They must be approved by the AECB.
- **Generic Licensing Issues** are those identified by the AECB which apply to more than one plant. New plants are expected to make significant progress in clearing these issues.

Currently AECL is on schedule in submission of documents. AECB reviews, while limited by the requirement to increase resources rapidly, have focused on both the plant requirements and on any features of CANDU 9 that are perceived to be different from as-built CANDUs, particularly on the conversion from the multi-unit Bruce-B/Darlington design to a single-unit.

Disputes naturally arise in the course of such a review. They need to be resolved rapidly. The process is as follows:

- AECL submits each formal deliverable
- AECB Staff formally reply with its review
- an informal meeting is held with AECB Staff to ensure that both sides understand the comments, and to indicate AECL's response
- AECL responds formally to the AECB Staff
- AECB Staff either close off the item, or, if there are issues still in dispute, they are sent up the management hierarchy of each organization, with attempted resolution at each management level, up to the Executive of AECL and the AECB.

At this stage in the review, the process can be said to be working satisfactorily.

5. Conclusions

The Canadian approach to minimizing licensing risk relies on those unique aspects of CANDU licensing philosophy—a non-prescriptive, judgmental process which places the primary responsibility for safety on the designer/owner; a well-focused, one-product industry, where all players know the product well; and successful export of this philosophy to CANDU customers and their regulators overseas. The success to date of up-front licensing on CANDU 3 is now being used on CANDU 9 in a shorter time-scale. The result will assure both domestic and foreign customers that they can embark on a project construction without delays due to licensing disputes.

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SAFETY AND LICENSING ISSUES FOR INDIAN PHWRs

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Abstract

India has achieved competency in design, construction, commissioning and operation of Pressurized Heavy Water Reactor based Nuclear Power Plants and has completed more than 120 reactor operating years with an extremely satisfactory safety record. In this paper, the safety management in NPCIL and operational safety aspects are discussed, licensing & regulatory approach is described and some of the main safety issues for Indian PHWRs are brought out.

1.0 INTRODUCTION

The Indian Pressurized Heavy Water Reactor (PHWR) programme is the first stage of the three phase programme (envisaged almost four decades ago) ultimately aimed at exploiting India's vast Thorium resources to meet long term energy needs. The design, construction, commissioning and operation of Nuclear Power Plants is the responsibility of Nuclear Power Corporation (NPC), a Government of India enterprise under the Department of Atomic Energy (DAE). The Bhabha Atomic Research Centre (BARC) is responsible for research and development of thermal reactors. All phases of nuclear activity are regulated by an independent body, the Atomic Energy Regulatory Board (AERB).

With concerted and dedicated efforts in the past four & half decades, India has been successful in establishing a self reliant nuclear power industry, the related nuclear fuel cycle facilities and has also mastered the technology of nuclear waste handling.

With eight PHWRs and two Boiling Water Reactors (BWRs) now in operation, India has completed more than 120 reactor operating years with an extremely satisfactory safety record and plants have operated with a capacity factor of about 65%. Four PHWRs are presently under construction and 12 more are planned. NPC has evolved a standardized design of 220 MWe PHWRs and the design for larger size units of 500 MWe capacity is at an advanced stage. The basic design of 220/500 MWe units is similar, however, a number of significant state-of-the-art design changes have been made progressively from RAPS-1&2 to MAPS-1&2 to NAPS-1&2 to Kaiga-1&2 and finally to the 500 MWe Units. These changes have been made from the consideration of currently prevailing safety criteria, seismicity, improved availability, requirement of in-service inspection, ease of maintenance, etc. as appropriate to Indian conditions.

The radioactive effluent releases through air and water routes from all Nuclear Power Stations are fractions of the respective permissible technical specification limits. The calculated and measured radiation doses in the public domain are fractions of permissible exposure as per AERB and ICRP limits. India is one among the first few countries who have adopted the new ICRP 60 individual dose limits. NPC is pro-active on safety management and has at its own initiative taken up review and implementation of various corrective actions in consultation with AERB. A Business Plan is under implementation for strengthening **Safety Culture** in an attempt to move towards excellence in the area of safety.

In this paper, safety management in NPC and operational safety aspects are discussed, licensing & regulatory approach is described and some of the main safety issues for Indian PHWRs are brought out.

2.0 SAFETY MANAGEMENT IN NPC

Safety is given utmost importance in the Nuclear Power Corporation. It occupies a predominant position in the mission statement and safety policy document. In order to achieve overall safety (be it nuclear, conventional or environmental), comprehensive efforts are required at all stages of a Nuclear Power Plant (NPP) starting from site selection, design, manufacture, construction, commissioning, operation and decommissioning.

The safety management of NPC strives to fulfill the overall safety objectives (general nuclear safety, radiation protection and technical safety) at each stage of a NPP. The safety policy of NPC has been defined and being implemented. This shall be periodically reviewed to make it responsive to changes in statutory regulations, technological advances and the reaction of the public to safety and environmental issues.

Safety organization structure at headquarters and stations clearly identifying the corporate and individual responsibilities has been established. Also, the accountability for safety at various levels in the organization been defined.

Corporate Business Plan for strengthening Safety Culture and Safety Culture Indicators has been issued. This shall be further developed and propagated at all the units of NPC.

2.1 Internal Safety Review/Verification

Within NPC, the Directorate of Health, Safety, Environment & Quality Assurance (HSE&QA) is responsible for the independent verification of design safety as well as independent evaluation of operational safety. In order to focus the safety significance of the issues involved and to ensure quality and relevance of submissions, a system of formal, independent inter-disciplinary safety review has been established for review of station proposals and documents prior to their submission to AERB. To facilitate the detailed review at design stage as also for final vetting of applications from operation/commissioning units to Regulatory body, exhaustive checklists have been prepared. Headquarters Instructions (HQIs) have been issued to all projects and stations specifying the guidelines for submission of reports/documents to AERB (or any of its constituent committees) as also for review and further processing of reports/documents received from AERB (or its committees).

A system has been instituted for ensuring operational safety in a pro-active manner (through systematic peer reviews, Internal Safety Review of Operating Station 'ISROS', Assessment of Safety Significan Event Team 'ASSET' reviews, Assessment of Safety Culture in Operating Team 'ASCOT', etc.), far exceeding the expectations of the regulatory body. The aspects of a) Human factor engineering; b) Environmental qualifications; c) ALARA; d) Ageing, back-fitting and life extension; e) Assessment & trending of system performance and events; etc. are being given due importance.

2.2 **Safety Assessment Criteria**

The proof of compliance with the safety objective requires in-depth safety assessment as described above. These assessments check with the compliance with a set of criteria including the technical/safety codes, guides, standards and several other documents as indicated below:

a) Design Manuals, Design Basis Reports, Safety Reports (Vol-1: Technical Description, Vol-2: Accident Analysis)

b) Operating documents like Operating Manuals, Flow Sheets, Station Instructions

c) Technical Specifications

d) Operating Policies

e) SARCOP/unit SRC Recommendations

f) AERB Manuals, Codes and Guides for Safety of NPPs for -

- Safety Manual Governing the Authorization Procedure for NPPs
- Siting (SC/S, SG/S-1 to 12)
- Quality Assurance (SC/QA, SG/QA-1 to 5)
- Design (SC/D, SG/D-1 to 25)
- Civil Engineering Design (SC/CSE, SG/CSE-1 to 8)
- Operation (SC/O, SG/O-1 to 11)
- Radiation Related Aspects (SG/HS-1, SG/Decom., RPM, RPR, M/NISD-1 to 2)
- Governmental Organisation (SG/G-1 to 8)

(While some of the above have already been published, others are under preparation/publication)

g) IAEA Safety Series -

- INSAG Reports	75
- Code of Practice	50-C
- Safety Guides	50-SG

h) ISI (Indian Standards Institution) Standards

i) IEEE standards, etc.

Safety assessments also include the analysis of accident scenarios representative of the whole range of events and processes which have potential radiological consequences.

2.3 Application of New Safety Significant Information

The importance of keeping abreast with the latest trends in safety is given due emphasis in India. Several experts are members of INSAG, OSART & IAEA safety series working groups. We also get benefit from NRC, WANO, COG and IRS reports. A system has been established for assimilation of state-of-the-art developments and experiences elsewhere and to make swift corrections, as relevant, to enhance safety & performance of our plants. Subsequent to the accidents at TMI-2 and Chernobyl, exhaustive reviews of Indian PHWRs were taken up and several upgradations (i.e., 39 after TMI and 47 after Chernobyl), were carried out.

2.4 Probabilistic Safety/Risk Assessment (PSA/PRA)

This is one area where only a beginning has been made. System reliability analysis has been performed for all operating PHWRs (RAPS, MAPS, NAPS and KAPS), with the studies getting more elaborate and comprehensive for successive plants. At present, attention is focused on performing level 1 PSA (for internal events) for all our plants, since the major benefits and insights are expected from level 1 PSA. Level 1 PSA has been completed for NAPS. Plant specific failure rate data from operating plants is being collected. Treatment of external events will be taken up next. PSA level 2 and 3 work will be taken up subsequently. In general, strengthening would be required in almost all areas i.e., fast running computer modeling, uncertainty analysis, efficient sampling technique, accident analysis techniques, accident frequency, progression and source term analysis, risk integration and finally consequence analysis.

2.5 Emergency Preparedness

More than 100 emergency conditions have been postulated and procedures called "OPEC" (i.e., Operating Procedures Under Emergency Conditions) have been prepared for the same. These are to be used by plant operators for mitigating the effects in the unlikely chance of any of those abnormal incidents. Efforts are being made to evolve symptom based Emergency Operating Procedures (EOPs) to facilitate ready identification of the evolving scenario and appropriate action(s) based on safety function maintenance approach. Introduction of intelligent computer based systems for assisting the operator in decision making is also being envisaged. The OPEC procedure for Station Black Out (SBO) was satisfactorily followed during the NAPS fire incident.

The emergency preparedness procedures are extremely satisfactory and it is a practice to conduct quarterly station emergency drills and yearly off-site full scale exercises. Feed back sessions are held after each drill/exercise with all including responsible Government authorities and deficiencies are corrected.

2.6 Quality Assurance

Quality Assurance in all phases, is an essential ingredient for overall safety management. The safety requirements & design commitments made in system Design Basis Reports and stipulations by safety bodies i.e., Project Design Safety Committee (PDSC), Civil Engineering Safety Committee (CESC) and Advisory Committee for Project Safety Review (ACPSR), etc. shall be systematically incorporated into subsequent documents (specifications, procedures, Design Manuals and Operating Manuals) for ensuring comprehensive compliance/ implementation and traceability. The verification of the same is an important component of QA in design. QA practices during construction and operation involve a detailed analysis of tasks to be performed such as, the identification of desired skills, the selection and training of personnel, the use of appropriate equipment and procedures, document control and record systems, clarification and recognition of individual responsibilities, etc. A topical QA Document has been issued for implementation. HSE&QA Directorate of NPC is entrusted to act as nodal agency to ensure the above.

3.0 OPERATIONAL SAFETY OF PLANTs

Operational safety and reliability are given the top most consideration. Continuous efforts are being put to reduce the Safety Related Unusual Occurrences (SRUORs) and Technical Specification Violations (TSVs). Due to these efforts, the numbers per unit, of both SRUORs & TSVs have come down (refer Figs. 1 and 2).

It is the combination of sound design and operating practices that go to make a safe reactor. Recognizing this, the operation of nuclear power stations in India is characterized by:

a) Strict adherence to technical specifications for station operation, which set safety limit on various parameters, and lay down requirements regarding operability, surveillance and testing of key equipment.

b) very careful selection, training, qualification and re-qualification of operations personnel. (This has paid rich dividends in the form of prompt operator response, acclaimed internationally, during NAPS fire incident and KAPS flood incident.)

c) a well planned in-service inspection programme.

d) regular preventive maintenance.

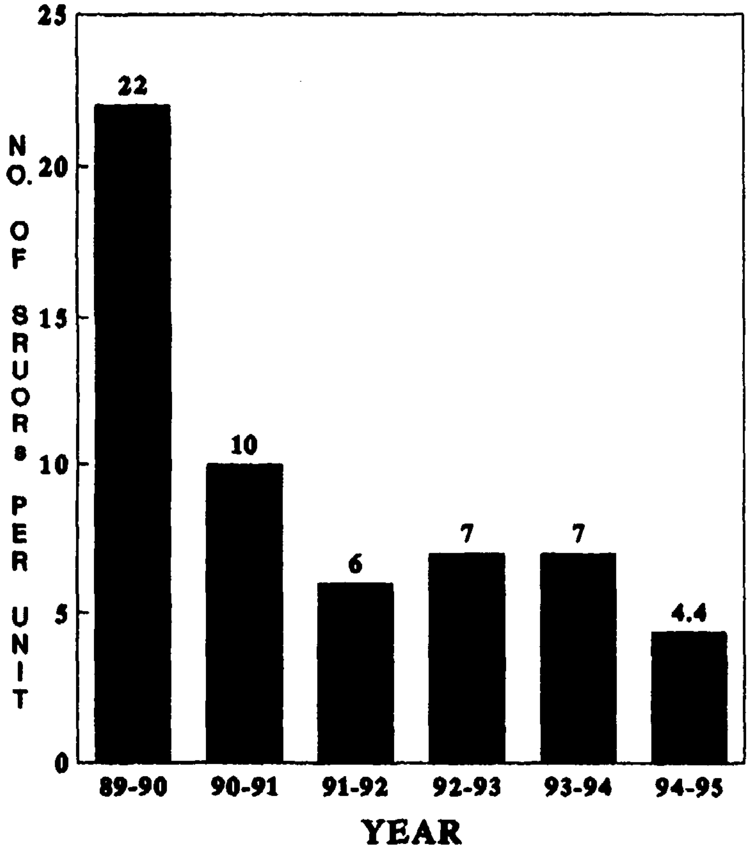

Fig.1: Safety Related Unusual Occurrence Reports (SRUORs) in PHWRs

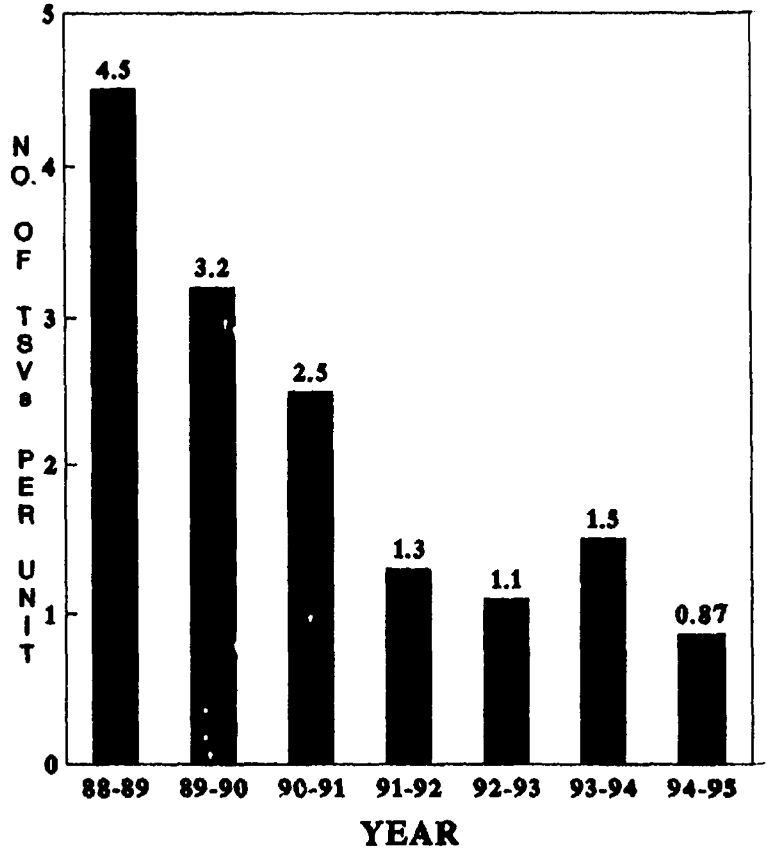

Fig.2: Technical Specification Violations (TSVs) in PHWRs

e) rigorous and multi-tiered review of operations as part of station procedures, as well as part of the regulatory process during operations.

f) Continuous review of all SRUORs, near miss events (for ASSET type review) and corrective actions at all stations. Continuously reducing trend is the objective.

g) Similar approach (as above) for technical and operational policy violations, on-power entries, over exposures, use of jumpers, non-conformance to stipulations, HQIs, etc., is followed.

h) Reduction of the station man-rem, effluent releases, active waste disposal, etc., to ALARA.

i) Standardization of all mandatory procedures (through HQIs) like containment tests, emergency exercises & drills, fire drills, testing of ESFs (diesels, emergency transfer batteries, etc.).

j) Mid-term technical reviews of various systems (such as D2O to H2O leak detection system, chromotograph, programmable digital comparator system, control room computer system, regulating system, etc.) in their present status to meet the original design & safety intent. Such reviews may be part of root cause analysis/ASSET reviews.

h) Development of standard procedures on the basis of operational experience for use in all stations.

Periodic testing, maintenance and observed deficiencies provide a large amount of information about the effective behaviour and reliability of components, system and staff. Structured collection of this information allows comparison with design assumptions, correction of weaknesses and possible improvements. Performance indicators are being developed and implemented to be able to monitor, in a more systematic way, the quality of operation and maintenance, and the overall safety performance of both technical system, staff and management. With this in view, continuous review of performance of Engineered-Safety-Features (ESFs) is being done. Table-1 lists the documents and reporting system which constitute a built-in mechanism for safety audit.

3.1 Operational Safety Review

The operational regime has a some what overlapping nine-tier safety control organisation to continuously monitor, review and ensure maintenance/improvement in the safety performance of various operating units. These include design built (engineered) safety features, operating organization's internal safety practices, internal safety audits, adherence to technical specifications and operating policies, SORC, unit SRC, SARCOP and AERB. Elaborate reporting and administrative system ensure timely and exhaustive reviews by experts at appropriate stage. Personnel involved in the reviews have sufficient independence from cost, schedule and other production considerations.

TABLE-1

DOCUMENTS AUTOMATICALLY RESULT IN SAFETY AUDIT

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1. Safety System Test Forms
2. SRUORs
   - 24 Hour Report
   - 20 Days' Report
3. a) Technical Specification Violation
   - Report
   - Condonation

   b) Operational policy Violation
   - Report
   - Condonation
4. Routine Safety System Setting Checks
5. Applications to Safety Organisation
6. Technical Audit Reports
7. AERB/SARCOP (Safety Review Committee for Operating Plants) Audit Proforma
8. Jumper Forms
9. Industrial Safety Forms
10. Technical Specifications Checks Compliance Forms
11. WANO Performance Indicators
12. SARCOP/SRC (unit Safety Review Committee)/SORC (Station Operation & Review Committee) recommendation status
13. Quarterly Reports of SARCOP
14. Monthly Reports Giving Effluent Discharge, Manrem, Active Waste Disposal Details, etc.
15. Safety Assessment Report for Renewal of Authorisation (SARRA)

Continuous emphasis is being placed for strengthening the overall safety culture. Considerable efforts have already been put to improve safety by devising a system/organisation which allows in-depth safety assessment/verification and internal audits to achieve the necessary safety standards. However, in order to get the desired results, the system of safety verification must cover the whole range of activities and procedures such as independent assessment of safety in design, quality assurance at all stages including site selection, construction, commissioning, etc. These are to be followed during the operational phase by in-depth periodic safety reviews of operating units throughout the operating life until safely decommissioned. In addition to paying attention to specific safety issues at plant level, the operating organisation carries out periodic reviews to confirm the continued validity of the design intent and safety features of the installation. Such reviews include consideration of the cumulative effects of modifications and changes of procedures, of the ageing of those components that have not been recently replaced or revalidated, of operating experiences and of technical developments. In addition to using established methods of safety analysis, the quality of safety management, operation and maintenance are subjected to special peer reviews. The AERB and Directorate of Health & Safety of NPC ensure that such systematic programmes exist, provide adequate coverage of the issues and assess the quality of the results.

Apart form one time reviews as above, a system of routine Internal Safety Review of Operating Station (ISROS) has been set up in 1991 to conduct yearly in-depth review of each operating station. These are generally based on OSART (Operational Safety Review Team) & SALP (Systematic Assessment of Licensee Performance of NRC) reviews with relevant additions necessary in the Indian context. The combination of routine and annual surveillance and periodic safety re-assessments together with peer reviews ensure that the operation of the installation always remains safe.

4.0 LICENSING AND REGULATORY PROCESS

AERB, which functions independently, is the apex body with regard to safety of NPPs. It maintains control over nuclear safety in siting, design, construction, operation and decommissioning of NPPs through regulatory actions that authorize activities, establishes requirements and conditions governing the performance of these activities and, where appropriate, places time limits on the validity of the authorizations. One of the primary means by which the regulatory body achieves this objective is the review and assessment of the applications for authorizations including the information submitted by the plants in their support. The authorization procedure adopted by AERB for PHWRs is laid out in AERB's Safety Manual Governing the Authorization Procedure for NPPs

4.1 Regulatory Safety Review

The safety reviews are carried out at various levels by committees composed of experienced engineers/scientists and experts (from both inside and outside DAE) who as a body, comprise expertise in nearly all aspects of NPP safety. In general, for each of the major authorization stages, a three-tier review process is followed by AERB before any activity is authorized in a NPP. The first level of regulatory review is by the Site Evaluation Committee (SEC), Project Design Safety Committee (PDSC), Civil Engineering Safety Committee (CESC) or unit level Safety Review Committee (SRC), as appropriate. These committees review the submissions, starting from Site Evaluation report (SER) upto the final test and commissioning reports. They also visit the project sites to review quality surveillance results and other site test results. The next level of review is conducted through Advisory Committee for Project Safety Review (ACPSR) or Safety review Committee for Operating Plants (SARCOP), which discuss the recommendations of SEC/PDSC/CESC/SRC on the related authorizations. Finally, the board of AERB is the statutory authorizing agency. It considers the recommendations of ACPSR/SARCOP and decides on the authorization under consideration. Table - 2 summarizes the levels of regulatory review at various stages.

TABLE-2

LEVELS OF REGULATORY REVIEW AT VARIOUS STAGES

Siting	Design/Commissioning	Operation
SEC	PDSC/CESC	Unit Safety Review Committee
ACPSR	ACPSR	SARCOP
AERB	AERB	AERB

4.2 Site Acceptance

The basic objectives at the site acceptance stage are to establish the conceptual design of the facility and to determine whether it is feasible to design, construct and operate the facility on the proposed site while meeting the safety objectives and requirements established by the AERB through various codes and guides for siting. The primary documentation required is the SER providing a summary description of the proposed station and information on land use, present and predicted population, principle sources and movement of water, water usage, meteorological conditions, seismology and local geology. The AERB, itself, is primarily concerned with the inter-relationship of the site and plant and evaluation of environmental impact is entrusted to associated environmental agencies.

4.3 Construction Authorization

Prior to granting the second formal authorization for site construction, the AERB must be assured that the design is such that the AERB safety principles and requirements will be met and that the plant will be built to appropriate quality standards. In order to do this, it in necessary that the design be sufficiently advanced to enable the safety analyses to be performed and their results assessed. The primary documentation required includes a Preliminary Safety Analysis Report (PSAR) (which is essentially a commitment document that combines the essential information of the SER, a description of the reference design, and the preliminary safety analysis), an overall Quality Assurance program for the project together with a specific program for construction quality assurance and preliminary plans for operation.

4.4 Commissioning Authorization

The third major authorization stage is prior to commissioning. For Indian NPPs, the authorization for commissioning is given in several interim stages. For PHWRs, these interim stages, in general, are:

a) hot conditioning of the primary system.

b) fuel loading of the reactor core and heavy water addition to storage and cooling systems (except core).

c) initial approach to criticality.

d) low power physics calibration experiments (Phase B).

e) low power engineering tests (Phase C) and tests at 25%, 50% and 75% of rated power.

f) tests at rated power.

AERB may alter, if necessary, the order of above-mentioned interim stages, to add new stages or to combine some stages, in specific cases, if safety considerations so require. The documentary submissions for each of the commissioning stages are therefore, related to the safety systems and safety features which are relevant to the commissioning stage concerned.

4.5 Operating License

Before issuing an operating license, the AERB must be assured, primarily, that the plant, "as-built", conforms to the design submitted and approved, and that the plans for operation are satisfactory. The requirements include submission of a Final Safety Analysis Report (FSAR) (documenting the "as-built" of the station and up-to-date analysis of possible accidents and the capability of safety systems to cope with them and limit their consequences), completion of previously approved commissioning program, examination and authorization of senior personnel, approval of operating policies and principles, preparation of plans and procedures for dealing with emergencies and a specific program for quality assurance during operation. Preparation of emergency preparedness manual (which is a document approved by respective State Government) and conducting a successful full-scale emergency exercise have now been made mandatory requirements by AERB before giving operating license.

Provided all has proceeded to the satisfaction of AERB, a full operating license is issued for a term not exceeding five years. Among the terms of an operating license is the requirement that the station inform the AERB promptly of any occurrence or situation which could affect the safety of the plant. AERB continuously updates and monitors safety requirements of NPPs and retains the right to impose additional conditions at any time. Stations seeking renewal of the operating license from AERB have to submit Safety Assessment Report for Renewal of Authorization (SARRA) once in five years.

Although the primary responsibility for the safe operation of the plant remains with NPC, there is continued surveillance by the AERB committees, annual reviews of operation and major reviews at times of renewal of operating license. Formal approval of the board will be required for decommissioning, although the situation has not yet arisen.

4.6 Authorization of Operators

The practice to date has been that those members of the operational staff who serve as Shift Charge Engineers, Assistant Shift Charge Engineers and Control Engineers must be specifically authorized by the AERB. Subsequently, they are required to be re-qualified once in three years. In the operating plants, these positions bear the primary responsibility for the day-to-day operation. AERB also monitors various health physics activities at plants.

5.0 SAFETY ISSUES

Some of the main safety issues concerning Indian PHWRs are discussed in the subsequent sections.

5.1 Radiation Embrittlement of End Shields of RAPS-1&2 & MAPS-1

During the time when RAPS-1&2 and MAPS-1 were built, not much information was available regarding the irradiation behaviour of the material used for end-shields. In these reactors, the Calandria side tube sheets, lattice tubes and main shell of end shields are made of 3.5% nickel steel material. Due to irradiation embrittlement, the Nil-Ductility-Transition-Temperature (NDTT) of end-shields' material has crossed the operating temperature within a short period of operation. The end-shields of these units are under operation with embrittled material. Only in case of RAPS-1 the cracking of the tube sheet ligament at lattice-tube to tube-sheet weld joints was observed. This has been attributed to the local stress relief treatment done during initial manufacturing stages before installation. No such stress relief treatment was done in case of RAPS-2 and MAPS-1 end-shields. Under the normal operating conditions, being a low stress component, the catastrophic failure is not envisaged based on the analysis carried out at BARC.

After detailed review of the subject by the experts, adequate precaution been incorporated in the operating instructions of RAPS/MAPS end-shields to avoid thermal shocks. Condition monitoring using acoustic emission technique has been planned during en-masse coolant channel replacement activities and fracture analysis work is being done.

5.2 Pressure Tube to Calandria Tube (PT/CT) Contact

The potential for PT/CT contact exists in RAPS-1&2, MAPS-1&2, NAPS-1&2 and KAPS-1 where zircaloy-2 has been used as the pressure tube material, due to creep characteristics of the material combined with possibility of shifting of garter springs. Though PT/CT contact in itself does not constitute any failure, it is known to encourage hydrogen pick-up and blister formation over a period of time and consequent weakening of pressure tubes. KAPS-2 onwards this problem is not there as zirc-niobium (Zr-2.5%Nb) pressure tubes with tight fitted garter springs have been used. Zr-2.5%Nb has much better creep characteristics. This problem is known to NPC after the incident in 1984, at Pickering-2 station in Canada. Some movement of garter springs has been observed in RAPS-2 and MAPS-2. There has been no movement observed in Narora from the results based on in-service-inspection. It is felt that the feature of moderator dumping may also be contributing to shifting of the loose garter springs, apart from the construction activities and high coolant flows during hot-conditioning.

At present, RAPS-1&2 are already under capital maintenance outage for en-masse coolant channel replacement and other retrofitting. Efforts are being made to assess the residual life of the contacting/non-contacting pressure tubes for other units and at appropriate stage, en-masse coolant channel replacement of these units will be taken up.

5.3 Tritium Removal From Heavy Water

There is a need to provide detritiation units at our PHWRs, particularly at RAPS and MAPS where system tritium levels are quite high. This, if feasible, will reduce the tritium exposures at out plants. R&D efforts towards the above are progressing satisfactorily at BARC.

5.4 Protection Against Fire & Flood

NAPS fire incident, which resulted in total station blackout for about 17 hours, did highlight certain strengths of design and operation as well as focussed on several weaknesses. The incident was reviewed in detail by two independent expert committees, common cause failure study was carried out and necessary modifications were incorporated. AERB has issued a fire protection code which embodies broadly the provisions of review practices followed internationally. Fire hazard analysis task forces have been constituted at all stations to carry out fire hazard analysis and to ensure compliance with the AERB fire protection code.

The design basis flood level of an NPP is always chosen to have a very low probability of exceedance per annum based on several site related factors. Based on the experience of flooding incident of KAPS (June 16, 1992), which was analysed using ASSET methodology, various protective measures have been implemented at all stations. The combination of parameters which lead to a flooding situation are dynamic in nature due to natural and man influenced activities and hence, protective measures need review on a continuing basis.

5.5 Station Blackout (SBO)

The present design of Indian PHWRs is capable to cater to a SBO of upto the duration of 8 hours (eventhough NAPS withstood a SBO of 17 hours satisfactorily). Detailed review of the subject was carried out by an expert committee and its recommendations will be implemented at all stations after getting necessary clearance from AERB.

5.6 Severe Accident Management

Since safety is never absolute in any venture, continuous efforts are required to improve upon the safety (by minimizing the risks) and performance of existing designs. In line with the international trends, consideration is being given to beyond design basis events also. As part of the above, preliminary identification and analysis has also been done for beyond-design-basis-accidents and severe accidents management guidelines for the plant operators are also being prepared.

6.0 CONCLUSION

In conclusion, it is summarized that safety of Indian Nuclear Power Plants is being given overriding importance right from the stage of site selection, followed by design, construction, commissioning, operation and decommissioning stages. Well structured quality assurance are being followed. A nine tier system for control of safety through internal and external organisation, development of safety culture, etc. are ensuring strict adherence to safety standards that can be compared to any international levels. The above have resulted in extremely safe operation of more than 120 reactor operating years without any major safety significant event. Experience with the operating stations in the country has demonstrated that these reactors are capable of operation with high reliability while ensuring safety of plant personnel and the surrounding population, and with minimal impact on the environment.

ADVANCES IN SYSTEM DESIGN

(Session 4, Part 1)

Chairman

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XA9846600

PASSIVE HEAT REMOVAL IN CANDU

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Abstract

CANDU has a tradition of incorporating passive systems and passive components whenever they are shown to offer performance that is equal to or better than that of active systems, and to be economic. Examples include the two independent shutdown systems that employ gravity and stored energy respectively, the dousing subsystem of the CANDU 6 containment system, and the ability of the moderator to cool the fuel in the event that all coolant is lost from the fuel channels.

CANDU 9 continues this tradition, incorporating a reserve water system (RWS) that increases the inventory of water in the reactor building and provides a passive source of makeup water and/or heat sinks to various key process systems.

The key component of the CANDU 9 reserve water system is a large (2500 cubic metres) water tank located at a high elevation in the reactor building. The reserve water system, while incorporating the recovery system functions, and the non-dousing functions of the dousing tank in CANDU 6, embraces other key systems to significantly extend the passive makeup/heat sink capability.

The capabilities of the reserve water system include makeup to the steam generators secondary side if all other sources of water are lost; makeup to the heat transport system in the event of a leak in excess of the D_2O makeup system capability; makeup to the moderator in the event of a moderator leak when the moderator heat sink is required; makeup to the emergency core cooling (ECC) system to assure NPSH to the ECC pumps during a loss of coolant accident (LOCA), and provision of a passive heat sink for the shield cooling system.

Other passive designs are now being developed by AECL. These will be incorporated in future CANDU plants when their performance has been fully proven.

This paper reviews the passive heat removal systems and features of current CANDU plants and the CANDU 9, and briefly reviews some of the passive heat removal concepts now being developed.

1. BACKGROUND

AECL has traditionally incorporated passive features and systems into CANDU plant designs whenever such systems were shown to economically and reliably meet all of the system requirements. An example, a passive feature relied upon during normal plant operation, is the ability of the reactor coolant to cool the fuel via natural circulation on the loss of power to the heat transport system (reactor coolant system) pumps and the inclusion of sufficient rotational inertia in the heat transport system pump motor assemblies to assure fuel cooling during reactor power and pump rundown. The heat transport system arrangement also caters to other design basis events, for example the 'figure-of-eight' HTS arrangement, with reactor coolant pumps in series, caters to postulated pump seizure events.

The use of passive features and systems and concepts extends to the special safety systems in CANDU plants. For example, two passive, fully capable shutdown systems (one driven by gravity and the other by stored energy) are incorporated. Other examples include the gravity dousing system for post-LOCA reactor building pressure suppression in CANDU 6, and the vacuum building containment system employed in multi-unit Ontario Hydro stations.

The ability to assure fuel cooling under a range of design basis and beyond design basis events has received particular attention in CANDU plants. These systems which remove decay power from the fuel by passive mechanisms, or which provide makeup to cooling systems by passive mechanisms, are the focus of this paper. The provisions incorporated in operating CANDU plants are reviewed, followed by a brief review of the enhancements incorporated in CANDU 9, and a discussion of new features that are under development, and which may be incorporated in future plants.

2. THE MODERATOR HEAT SINK

2.1 Reactor and Heat Transport System Configuration

CANDU is a horizontal pressure-tube reactor, with the fuel bundles located inside several hundred 10.5-cm diameter, 0.48 cm thick pressure tubes (Figure 1). Twelve 0.5 m-long fuel bundles reside within each pressure tube. The 37-element fuel bundle is in close proximity to the pressure tube, separated from it by means of 1.1-mm high bearing pads on the outer fuel elements. The heavy water coolant flows over and through the fuel bundles and is contained by the pressure tubes within the core.

The pressure tube operates at approximately the coolant temperature (300°C), and is thermally insulated, during normal operation, from the heavy water moderator (70°C) by the carbon dioxide filled annulus formed between the concentric pressure tubes and calandria tubes. The calandria tube forms the outer boundary between the gas and the moderator (Figure 2). The assembly of fuel, pressure tube, gas annulus and calandria tube is collectively called the fuel channel. The total radial distance between the fuel and the moderator is 1.5 cm.

The moderator is contained within a low pressure tank, called the calandria. During normal operation, about 4.4% of the thermal output of the core is deposited in the moderator, a small amount by conduction from the channels, but mostly by direct deposition of fission gamma rays. This heat is removed via dedicated external circuit that includes heat exchangers and pumps; the pumps circulate the moderator D_2O through the heat exchangers and provides momentum to assist the mixing of the moderator within the calandria (see Figure 1). They are powered by normal Class IV electrical power, backed up by Class III emergency diesel power when required.

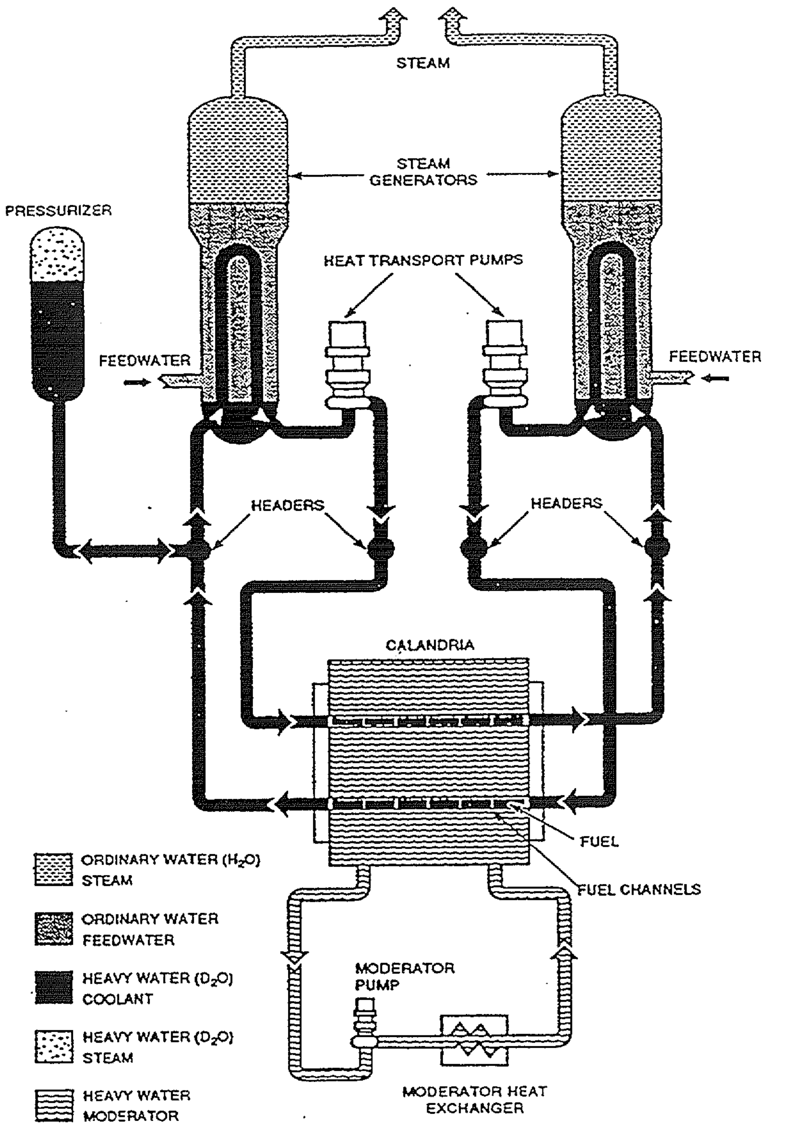

FIGURE 1: The CANDU Nuclear Steam Supply System

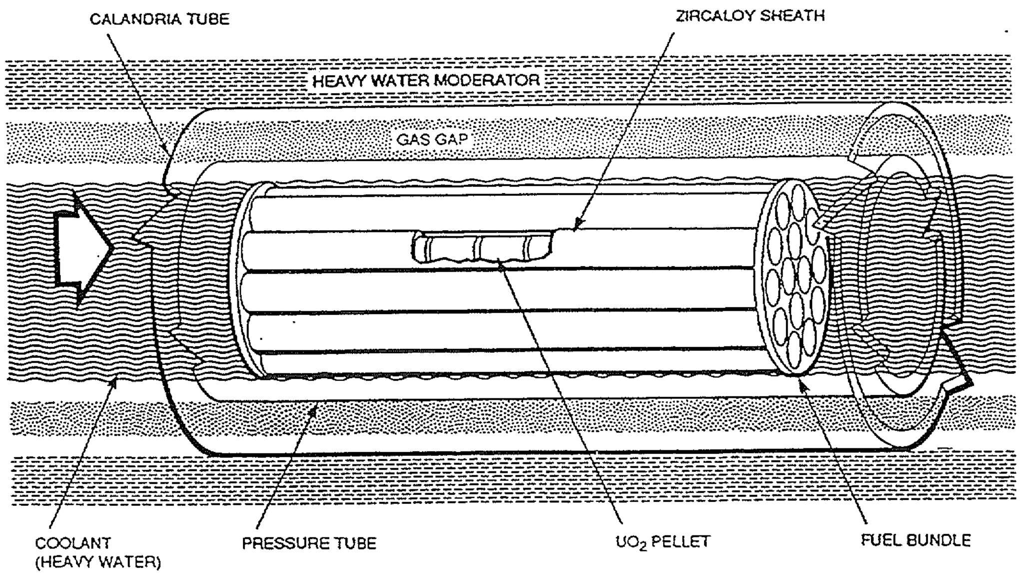

FIGURE 2: Separation of Coolant and Moderator

All large pipes in the CANDU Reactor Coolant System (RCS) are above the core. They consist of headers, or collectors, to which each channel is connected via a 6-cm to 8-cm diameter inlet and outlet feeder pipe; plus pump suction and discharge piping and steam generator inlet and outlet piping. A large break in one of these pipes would cause rapid voiding of the pressure tubes. As with other water-reactor designs, the emergency core cooling system (ECC) provides high-pressure injection of water to refill the core. In CANDU ECC water is supplied to all the reactor headers (see Figure 1).

2.2 The Moderator as an Emergency Heat Sink

The moderator system heat removal capability is enough to continuously remove all fuel decay heat 15 seconds after reactor shutdown. The moderator specific volume is typically 8 litres/kW(th) at 1% decay power, or enough to absorb (through heat-up and boil-off) over 5 hours of decay heat from the fuel, assuming no heat removal from the moderator D_2O.

A post-LOCA failure of ECC in light-water reactors, will, if uncorrected, lead to a meltdown of the core. In CANDU, a loss of coolant with a failure of ECC will be arrested by the transfer of heat from the fuel to the moderator short of UO_2 melting. The mechanism is as follows:

The fuel will heat up due to decay power, since no heat is being removed by the RCS. Since the pressure-tube is close by, it will also heat up, by conduction and radiation from the fuel, and convection by the steam remaining in the channel. At about 800^{0}C, the pressure tube will start to plastically deform under the loads from the weight of the fuel and any residual coolant pressure. If the coolant pressure is high (for example, for medium-sized breaks with failure of ECC), typically above 1 MPa, the pressure tube will strain radially outward until it contacts the cool calandria tube (Figure 3). If the pressure is below 1 MPa, the pressure tube will preferentially sag, until again it contacts the cool calandria tube. As long as the calandria tube remains cool, it is strong enough to arrest the deformation of the pressure tube. Heat can then be removed from the fuel, by conduction and radiation to the pressure tube and calandria tube, and then by convection to the bulk moderator. From there it is removed by the moderator cooling system. The pressure-tube thus acts as a passive fuse, deforming only when it overheats in an accident, and so creating a low-resistance heat transfer path to the moderator. This path can remove decay heat from the fuel without the UO_2 melting even with no coolant in the pressure-tube. This is due to the short physical distance from the fuel to the pressure-tube, the relatively thin walls of the pressure-tube and calandria tube, and the enhanced heat transfer through the two tubes when they touch.

The calandria tube can be kept cold by preventing dryout on the outside surface at the time of pressure-tube contact. The surface heat flux at contact is determined by the pressure-tube temperature, the interface heat transfer coefficient and the moderator subcooling. The former cannot practically be controlled, but the latter two can. For existing CANDU reactors, a moderator temperature of about 70°C is sufficient to prevent calandria tube dryout.

The pressure-tube "fuse" is sensitive to moderator temperature, but NOT to active moderator heat removal - it is truly passive. However the moderator pumps and heat exchangers are used to bring the severe accident to a controlled steady-state.

Measures are taken to assure that the pressure tube does not fail before it reaches the calandria tube. Although such failure would not prevent the moderator from performing its emergency role, the sequence is less complex if the pressure tube remains intact. Pressure tube integrity depends on the pressure at which the pressure-tube strains - the higher the pressure, the more sensitive is the strain to non-uniformities in pressure tube temperature, and the higher the chance of failure before contact with the calandria tube. The pressure parameter varies slightly with the design of the RCS (HTS).

Another severe accident results from assuming *all* heat sinks for the RCS are lost. This is an unlikely sequence because the following systems are each capable of removing decay heat from an intact RCS:

- the main feedwater system
- the auxiliary feedwater system
- the shutdown cooling system (which can be brought in at full RCS temperature and pressure)
- the Group II emergency feedwater system (this is a separate means of adding water to the steam generators, taking its supplies from a separate seismically qualified source and using independent seismically-qualified power)
- a gravity supply of water to the steam generator from the high level dousing or reserve water tanks

If however they are all lost, the RCS will pressurize and the fluid will gradually be lost through the relief valves, and the fuel will overheat. Since this sequence occurs at or above operating pressure, typically 10 MPa, the overheated pressure tubes will start to

- Close isolation valves on the suction line from the dousing tank.
- Open cooling water valves to the ECC heat Exchangers.

2. 3 Performance

The CANDU 6 ECC system, through the provision of appropriate redundancy, meets all the requirements set for the system. The system, however, incurs relatively high capital and maintenance costs. Since the system does not operate during normal reactor operation, regular on-power testing must be performed to meet the unavailability target of 10^{-3}. A complete system operation cannot be tested on-power, therefore, a series of overlapping tests, generally on a monthly basis, are conducted to check sub-system operation, including valve operation. This increases operating cost. The incentive to reduce the above costs prompted the studies into ECC simplification.

3. THE CANDU 9 EMERGENCY CORE COOLING SYSTEM

The CANDU 9 ECC system utilizes the same high pressure, gas-driven water tank concept employed on the CANDU 6, Bruce and CANDU 3 power plants. However, the design is modified to significantly reduce the number of valves and achieve other system simplifications. A simplified flow diagram of the CANDU 9 ECC system is shown in Figure 2.

The principal improvements and simplifications made in the CANDU 9 ECC system include the following:

- Replacement of D_2O isolation valves with one-way rupture discs.
- Elimination of high pressure injection and test valves and incorporation of a floating ball seal in the water tanks.
- Location of additional ECC components inside the reactor building.
- Elimination of the ECC system medium pressure stage.

Each of these modifications is described in more detail below.

3.1 Replacement of D_2O Isolation Valves with One-Way Rupture Discs

In the current CANDU 6 reactors, normally closed D_2O isolation valves are provided at the interface of the ECC system and the heat transport system. Low pressure rupture discs upstream of the D_2O isolation valves provide separation of light water and D_2O. The D_2O isolation valves isolate the high pressure heat fail before they contact their respective calandria tubes. The higher powered channels will fail first, and the pressure tubes will relieve the rest of the RCS fluid. This will reduce the RCS pressure and allow the moderator to act as an emergency heat sink as described above.

The pressure-tube/calandria-tube heat transfer has been fine-tuned. First, to reduce the sensitivity to initial moderator subcooling: When the two tubes contact in an accident, the stored heat in the pressure tube is transferred in a "pulse" through the calandria tube. To reduce the magnitude of this pulse, which sets the margin to CHF of the calandria tube, the inner surface of the latter has been roughened slightly (50 micron ridges). This "smears out" the heat transfer over a longer period of time and reduces the peak inter-tube heat flux. Second, to enhance the heat transfer after sag contact, and so to reduce the quasi-steady-state fuel temperatures, the inside of the calandria tube has been blackened.

3.2 SHIELD TANK AS HEAT SINK FOR THE CALANDRIA

The AECB requires that all combinations of a reactor system failure *and* the unavailability of a safety system be evaluated as design bases events - for example, the previous example of a large LOCA and failure of ECC injection. **Severe accidents** within this set, i.e., those for which the fuel heat is not removed by the RCS, result in damaged fuel, but do not lead to loss of pressure-tube geometry. Accidents which

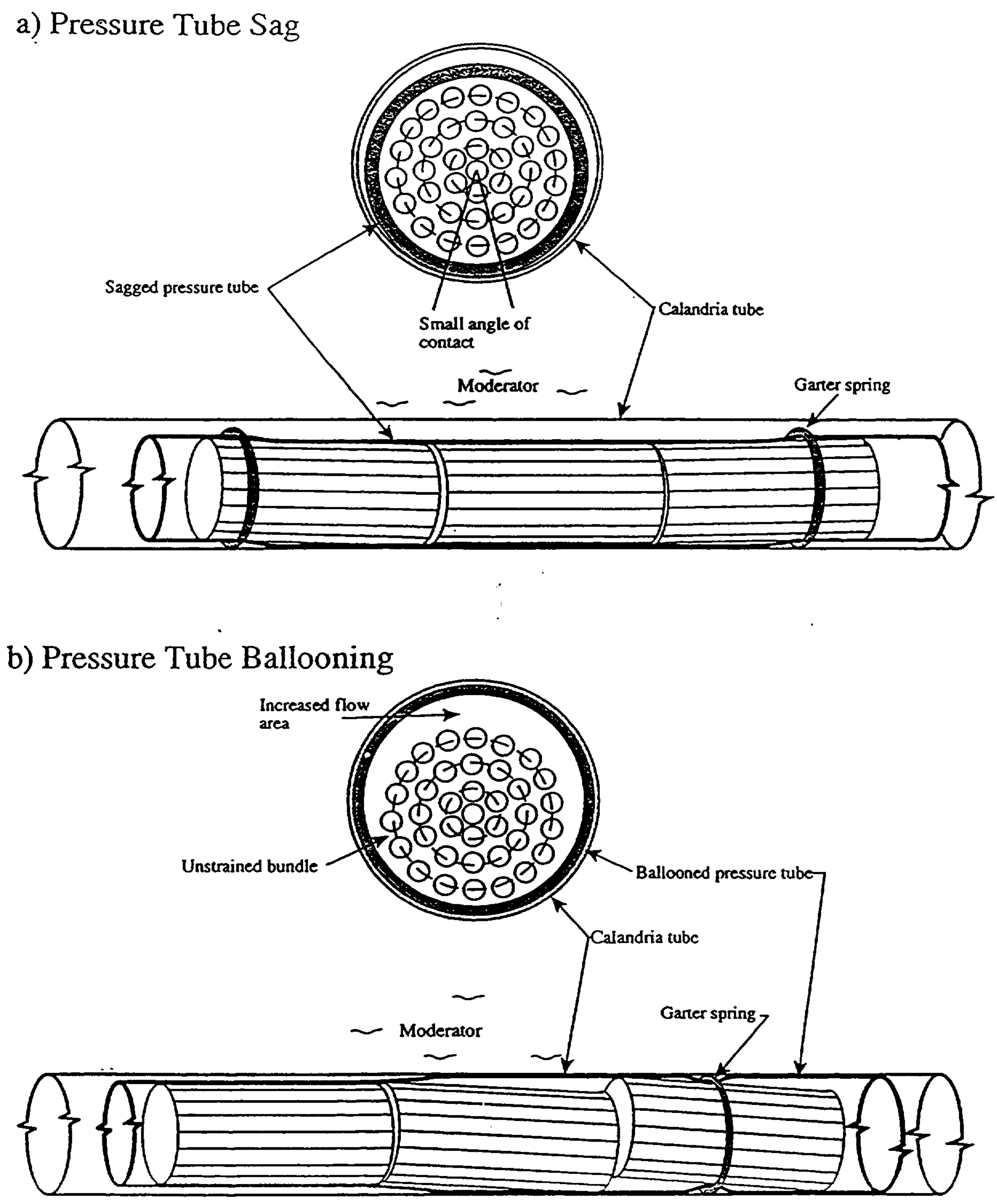

FIGURE 3: Pressure Tube Deformation Modes During a Postulated Loss-of-coolant Accident: a) Sagging, b) Ballooning

combine yet further failures are generally outside the design basis. They may result in loss of core geometry, in which case they are called **severe core damage** accidents. The two types of accidents are usually synonymous in other reactor types, but because the moderator can arrest severe accidents before the core geometry is lost, in CANDU they are distinct.

Severe core damage accidents in CANDU include sequences such as:

- loss of all feedwater and loss of cooling to all alternate heat sinks including the moderator

- loss of coolant, loss of ECC injection, and loss of moderator cooling.

The frequencies of such combinations are of the order of 10^{-7}/year, and are thus not within the scope of licensing analysis. They are, however, examined in the context of Probabilistic Risk Assessment. Because of the low frequency, the emphasis has been on scoping calculations rather than extensive experimental verification of detailed codes.

The calandria is contained within a shield tank, which provides biological shielding during normal operation and maintenance (Figure 4). It is a large steel or concrete tank filled with ordinary water. During normal operation, about 0.4% of the thermal output of the core is deposited in the shield tank and end shields, through conduction from the calandria structure and fission heating. This heat is removed via the end shield cooling system, consisting of pumps and heat exchangers.

The shield tank's role as an emergency heat sink for the fuel in a severe core damage accident is discussed below. In this role, its active heat removal capability is enough to continuously remove all fuel decay heat a few days after reactor shutdown. The shield tank specific volume is typically 16 litres/kW(th) at 1% decay power, or enough to absorb (through heat-up and boil-off) more then ten hours of decay heat from the fuel, assuming no heat removal from the shield tank water.

For such sequences, the moderator water will heat up and boil off. This will take some hours, during which time the pressure tubes will start to fail and the debris will collect in the bottom of the calandria. As long as there is water in the shield tank, the calandria shell will remain intact; the heat generated by the debris is less than the critical heat flux on the outer surface of the calandria. However, the shield tank heat removal rate is insufficient to keep up with the decay power until a few days have passed, so the shield tank water will boil off and the calandria shell will be penetrated. Nonetheless, the heat-up and boil-off of the moderator and shield tank buys valuable time, about 24 hours, so that accident management can be put into effect before the debris even reaches the concrete floor of the containment.

To ensure that steam is relieved from the shield tank without overpressurizing the vessel, engineered relief paths have been provided on the newer designs, sized to take the steam flow generated by decay heat removal.

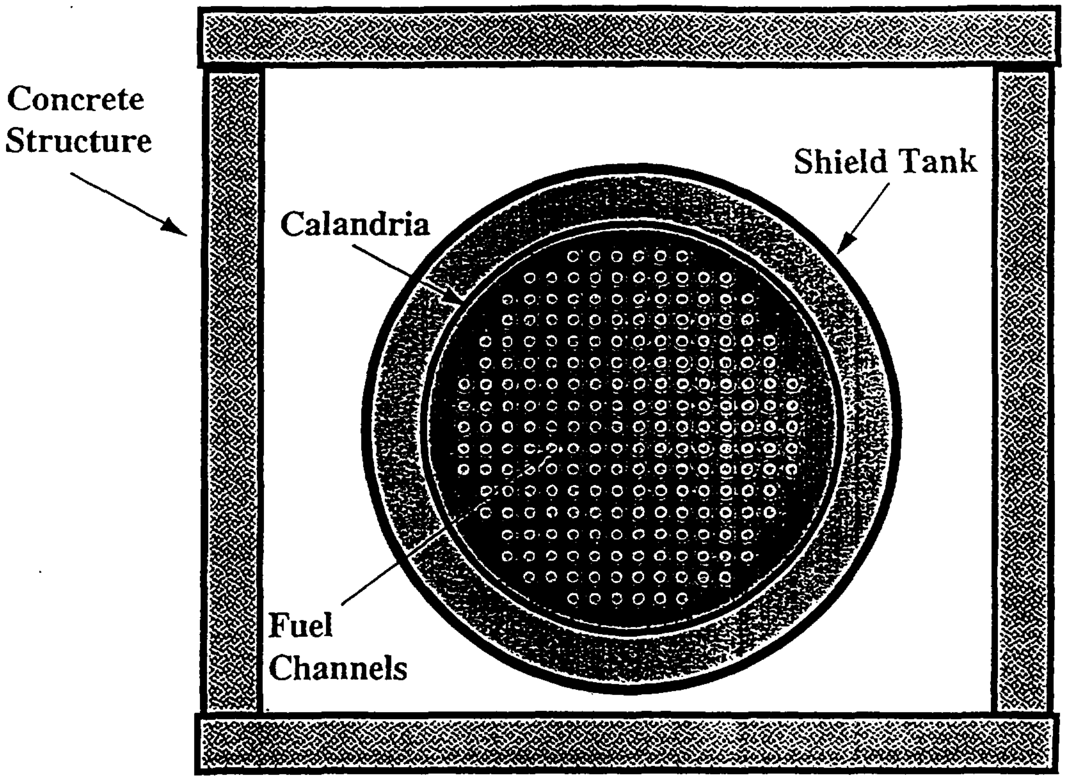

FIGURE 4: The Calandria - Shield Tank Arrangement

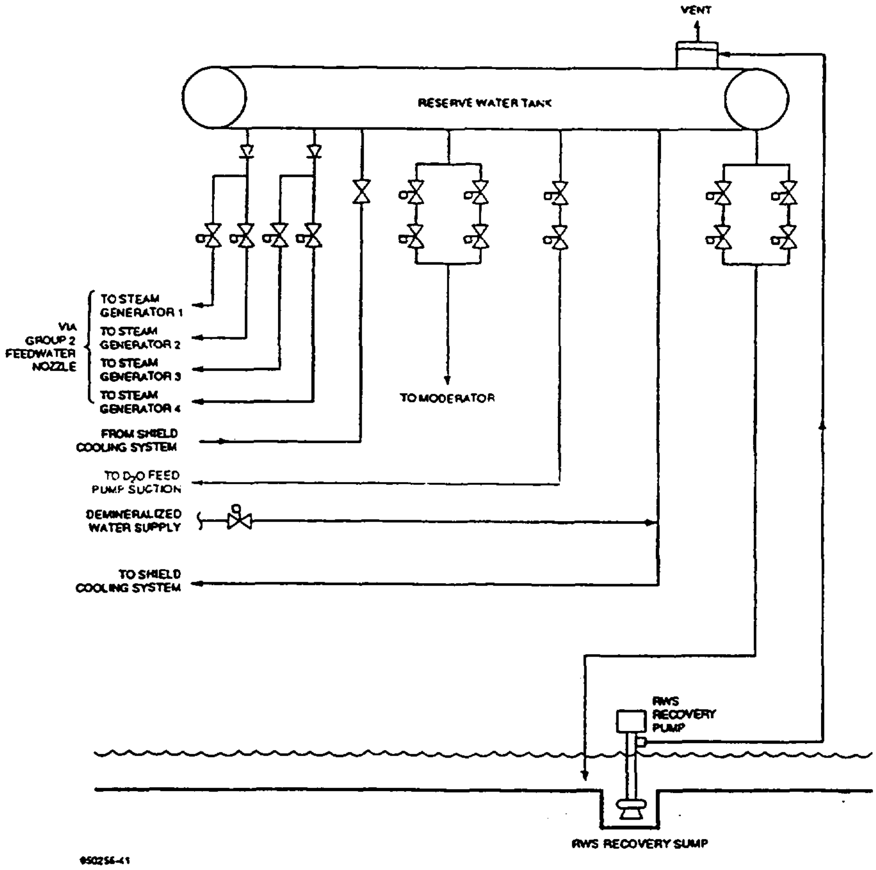

FIGURE 5: The CANDU 9 Reserve Water System

4. CANDU 9 ENHANCEMENTS TO IMPROVE PASSIVE HEAT SINKS

4.1 Moderator and Shield Tank Cooling Enhancements

Based on the previous descriptions, it is obvious how to extend the passive heat sinks provided by the moderator and the shield tank - simply add water. The advanced evolutionary CANDU 9 has done just that. An elevated reserve (Figure 5) water storage tank in containment provides emergency makeup water to the moderator and permits passive heat removal by thermosyphoning from the shield tank (Figure 6). The amount of water is sufficient for more than 40 hours of decay heat removal. During or after that time, a recovery pump collects water from the reactor building sumps and returns it to the reserve water tank. The heat is removed from containment through a combination of passive conduction through the reactor building walls and actively by containment air coolers

4.2 Provisions for HTS, Steam Generator and ECC System Makeup

The CANDU 9 also utilizes the reserve water tank to provide a gravity supply of makeup water to the steam generators, the ECC system, and to the heat transport system (Figure 5). The first two functions are conceptually similar to functions provided by the dousing tank of CANDU 6. Water can be fed from the reserve water tank to the secondary side of the steam generators to maintain the steam generator heat sink in the event that both Group 1 and Group 2 feedwater is lost.

Water is also fed from the reserve water tank to flood the reactor building floor on a loss of coolant accident (LOCA) signal to temper the D_2O from the break and provide net positive suction head to the ECC system pumps. In addition, CANDU 9 provides a connection from the reserve water tank to the D_2O feed pumps, which can be utilized to feed water to the heat transport system in the event of a very small LOCA.

The piping connection to the serve water tank assures that inventory is maintained in the reserve water tank to fulfill the various functions noted.

5. THE NEXT STEPS

A number of systems with passive features, generally extending the use of the CANDU 9 reserve water tank, are under development and evaluation. These are discussed in the following sub-sections.

5.1 Passive Moderator Cooling

A conceptual arrangement for passive moderator cooling upon the loss of the normal heat exchanger and/or pumps is shown in Figure 7. The moderator system circulates heavy water through the calandria to remove the nuclear heat generated in the moderator, and the heat transferred to the moderator from the fuel channels.

During normal operation the moderator pumps draw D_2O from the top of the calandria via the moderator heat exchanger. In the event that the moderator system pumps and/or heat exchangers are lost, the reactor is shut down, and flashing in the riser (see Figure 7) initiates natural convection circulation of the moderator. In this

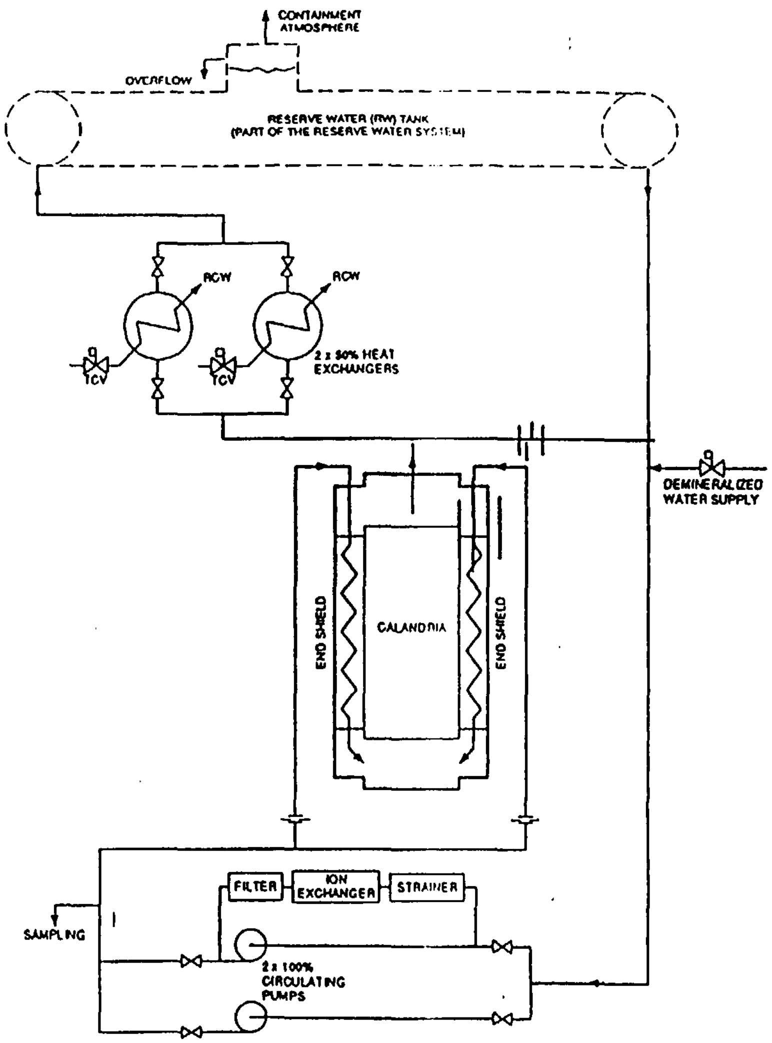

FIGURE 6: Passive Shield Cooling System

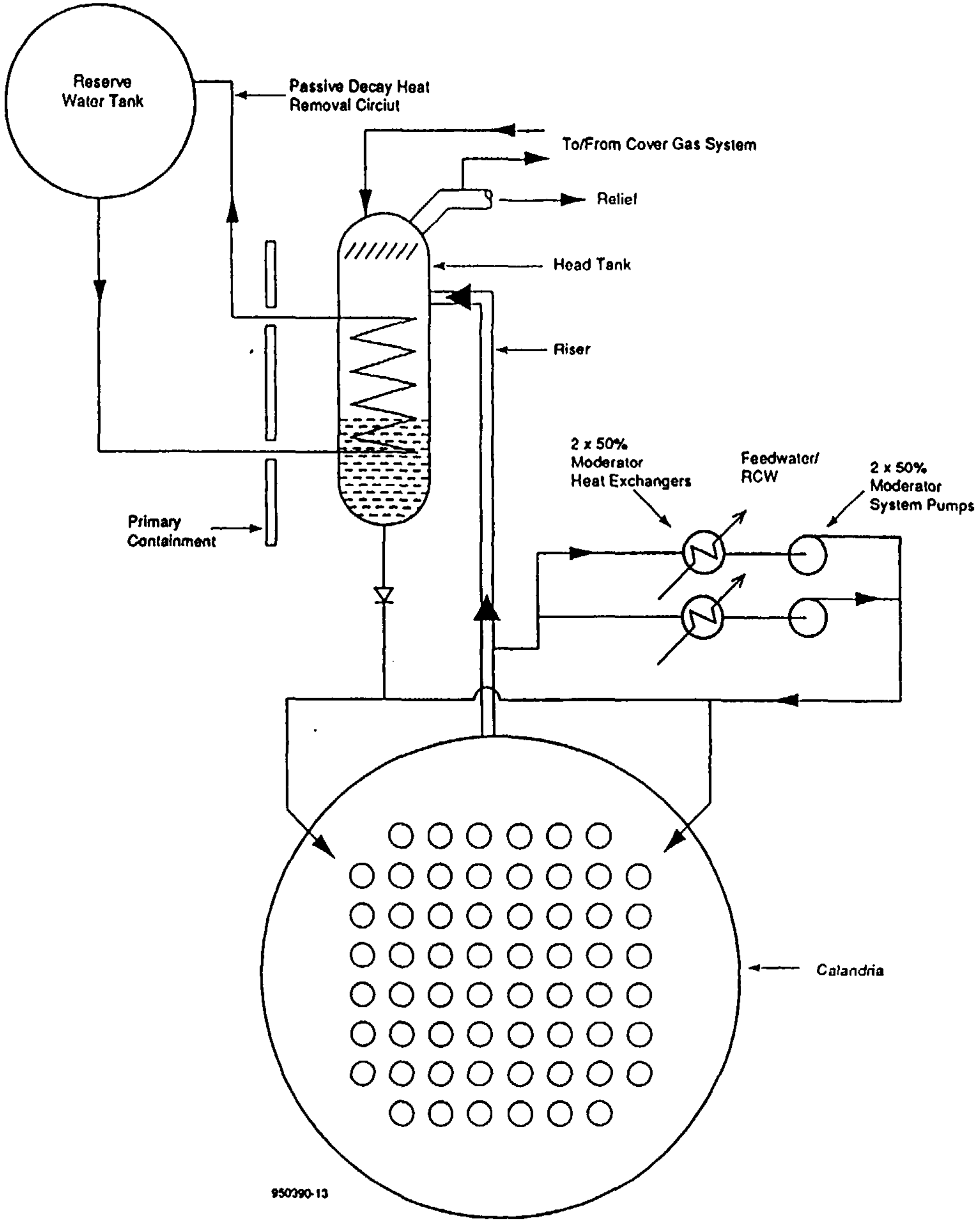

FIGURE 7: Passive Moderator Cooling System

mode, heat is rejected to the water in the reserve water tank via natural convection. The capacity of this mode of moderator cooling is sufficient to remove decay power under conditions of coincident loss of coolant and loss of emergency core cooling, without makeup or cooling being provided to the reserve tank for a period of 72 hours.

Testing is complete at AECL's Chalk River Laboratories on a full height / 1/60 diameter scale rig to evaluate the natural circulation mechanism. Work is now underway to assess the effect of D_2 and O_2 produced by radiolysis of the moderator on natural circulation during normal full power operation. The hope is to utilize natural circulation during both normal and upset conditions, thereby further reducing system cost.

Initiation of moderator cooling via the head tank cooling coil/reserve water tank is totally passive. No operator or control action is required.

5.2 Passive Steam Generator Heat Removal

The steam reject system being evaluated, shown in Figure 8 includes a reject condenser, connected to each steam generator.

When water level in the steam generator is within the specified operating range, the condensing coil in the reject condenser is filled with water, and circulation is prevented by the vapor (steam) lock. In the event that feedwater flow is lost, and the water level in the steam generator drops significantly below the top reject condenser coils, steam is condensed in the reject condenser coils, and the condensate is returned to the steam generator down
comer.

The secondary side of the reject condenser is cooled by natural convection, via flow from and to the reserve water tank. The water available to the reject condensers from the reserve water tank (water in the reject condenser compartment plus the water in the common portion of the reserve water tank) is sufficient to remove decay heat, via evaporation, for a period of 72 hours without makeup or cooling to the reserve water tank.

The operation of the reject condenser system is fully passive; no valve operation or operator action is needed to initiate operation.

5.3 Passive Containment Cooling

In the system under consideration, the containment is cooled via a water circuit, with coils located within a cooling duct inside the containment, and heat rejection to the reserve water tank (see Figure 9). During normal operation, circulating fans located in the cooling duct and pumps in the water circuit assure that temperatures in the containment do not exceed design values.

Following a postulated accident (loss of coolant accident or a steam line failure within the containment), all circulation fans and pumps may be lost. Under these conditions, natural convection in the water circuit and the heat rejection duct maintain the primary containment temperature below 125°C (except for an initial transient period).

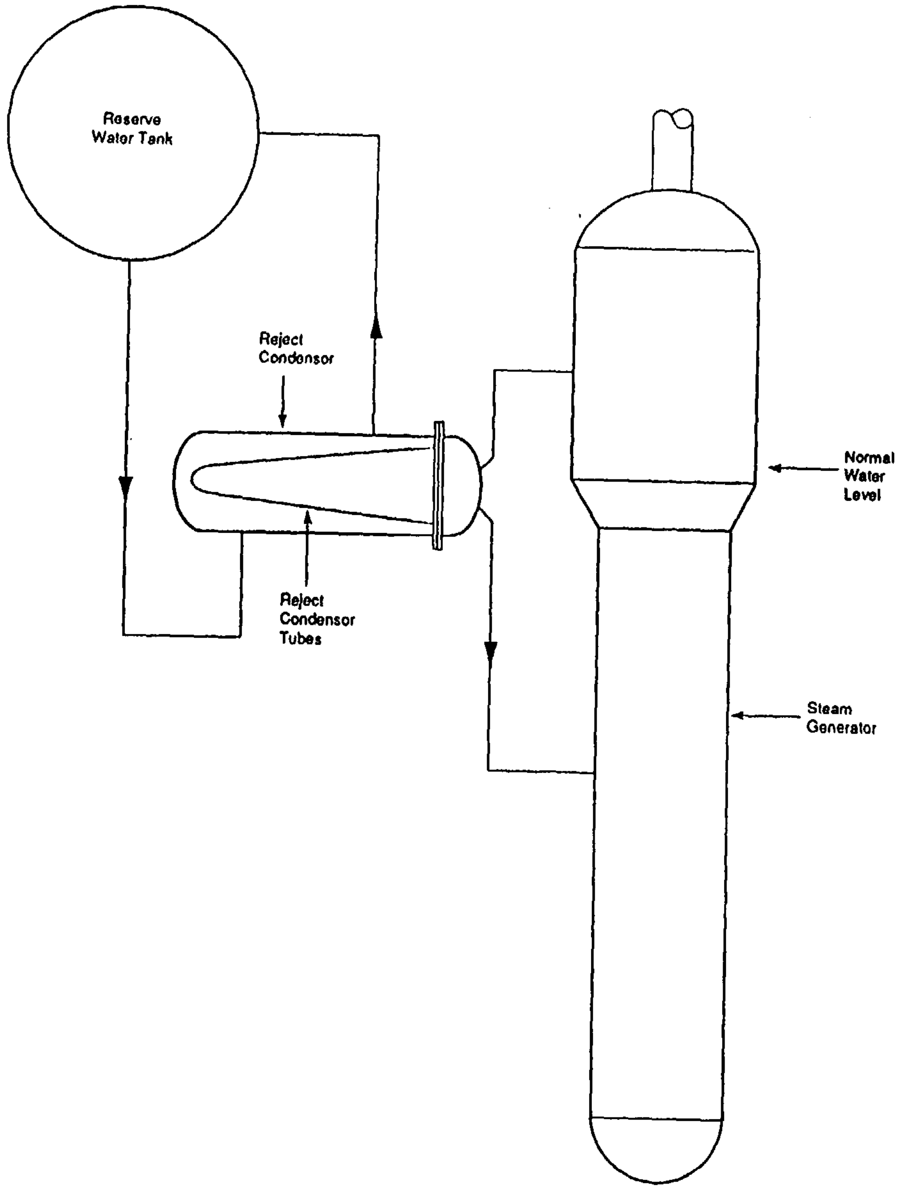

FIGURE 8: Reject Condenser Arrangement

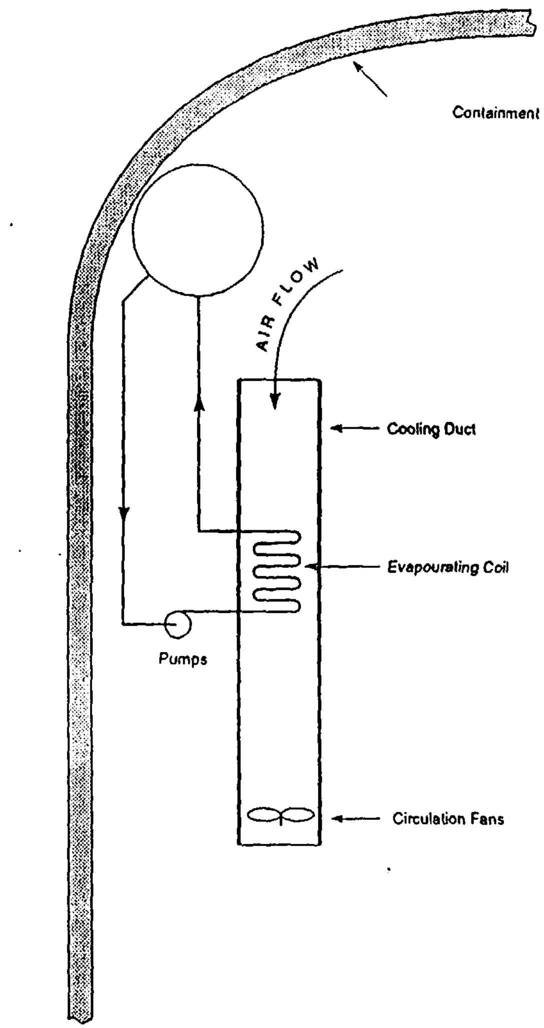

FIGURE 9: Passive Containment Cooling System

SUMMARY

Operating CANDU plants make extensive use of passive systems and features when their use has been shown to be economic and capable of meeting all requirements. The CANDU 9 increases the use of passive features, particularly in the area of decay heat removal capability under design basis and beyond design basis events. Additional passive features are under evaluation and development at AECL, and will be incorporated in future CANDU plants when their capability has been fully established.

EMERGENCY CORE COOLING SYSTEM SIMPLIFICATION

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Abstract

The emergency core cooling system (ECC) is one of four special safety systems employed in the defence in-depth approach to CANDU safety. Each of the special safety systems is routinely tested on-power to statistically demonstrate a failure rate of less than one in one thousand demands.

Studies and development programs at AECL over the last several years have been directed at simplification of the ECC system, with the objective of increasing reliability, reducing cost, and reducing maintenance and testing costs. This work has resulted in a substantial simplification of the ECC system for CANDU 9, including a reduction in the number of valves of over 50% relative to previous plants.

The "one-way" rupture disc and the floating ball seal developed and exhaustively tested by AECLs Chalk River Laboratories, are the foundation for the simplification achieved in the CANDU 9 ECC system. While building on the high pressure gas driven accumulator designs used at CANDU 6 plants and Bruce, introduces a "one-way" rupture disc to separate the ECC system from the heat transport system during normal reactor operation.

Conceptually, the "one-way" rupture disc assembly consists of a conventional rupture disc, supported from one side by a robust perforated plate. Pressure applied from the support plate side acts on the entire rupture disc area, causing it to rupture at low pressure (about 0.5 MPa); pressure applied to the rupture disc from the other side presses the disc against the perforated support plate, resulting in rupture (of small areas covering the perforations in the support plate) at a high pressure (above 20 MPa). Hence, the "one-way" rupture disc withstands full heat transport system pressure during reactor operation, but opens to allow emergency core coolant to flow into the heat transport system at a low differential pressure.

The floating ball seal blocks the discharge from the ECC system water tanks at the end of the injection stage, thereby preventing gas flow to the heat transport system. The passive floating ball seal replaces the valves previously used to prevent gas discharge.

This paper reviews the CANDU 9 ECC system design, and reviews the "one-way" rupture disc and floating ball seal development programs.

1. INTRODUCTION

CANDU nuclear power plants incorporate four special safety systems, i.e. shutdown system 1, shutdown system 2, the containment system and the emergency core cooling (ECC) system, as part of their defence in-depth approach to assuring safety. These special safety systems are diverse and functionally independent from each other and from the systems that provide reactor control and fuel cooling during normal reactor operation.

The ECC system assures fuel cooling following a loss of coolant accident by refilling the core and recirculating coolant through the core for long-term heat removal from the fuel. The ECC system is required to meet many demanding performance and reliability requirements. These include:

- the capability of maintaining or re-establishing sufficient cooling of the fuel and fuel channels so as to limit the release of fission products from the fuel in the reactor and maintain fuel channel integrity;
- a design such that the unavailability of the system can be demonstrated to be less than 10^{-3};
- a redundancy such that no failure of a single active component of the ECC system can result in the impairment of the system;
- the assurance that all maintenance and availability testing can be carried out without a reduction in the effectiveness of the system.

Historically, the ECC systems have been complex in terms of the number of valves and their operation, in order to meet the above and other requirements. This is evident in the description of the CANDU 6 ECC system presented in Section 2.

Over several years AECL has undertaken studies directed at the simplification of the ECC system. These studies have resulted in the simplified ECC system design adopted by the CANDU 9, which utilizes substantially fewer valves, simplifies operation and hence, improves reliability. This simplification also results in reduced operating costs due to reduced testing requirements and reduced maintenance costs.

2. THE CANDU 6 EMERGENCY CORE COOLING SYSTEM

The CANDU 6 ECC system supplies emergency coolant to the reactor core in three successive stages in the event of a loss of coolant accident: high, medium and low pressure. The high pressure stage uses gas pressure to inject light water from water tanks into the reactor core. The medium pressure stage supplies water from the dousing tank to the reactor core using one of the ECC pumps. When the ECC portion of the water in the dousing tank is depleted, the low pressure stage recovers the D_2O and H_2O mixture that has collected on the reactor building floor, and pumps is back into the reactor core via the ECC heat exchanger where heat is rejected to cooling water. A simplified flow diagram of the CANDU 6 ECC system is shown in Figure 1.

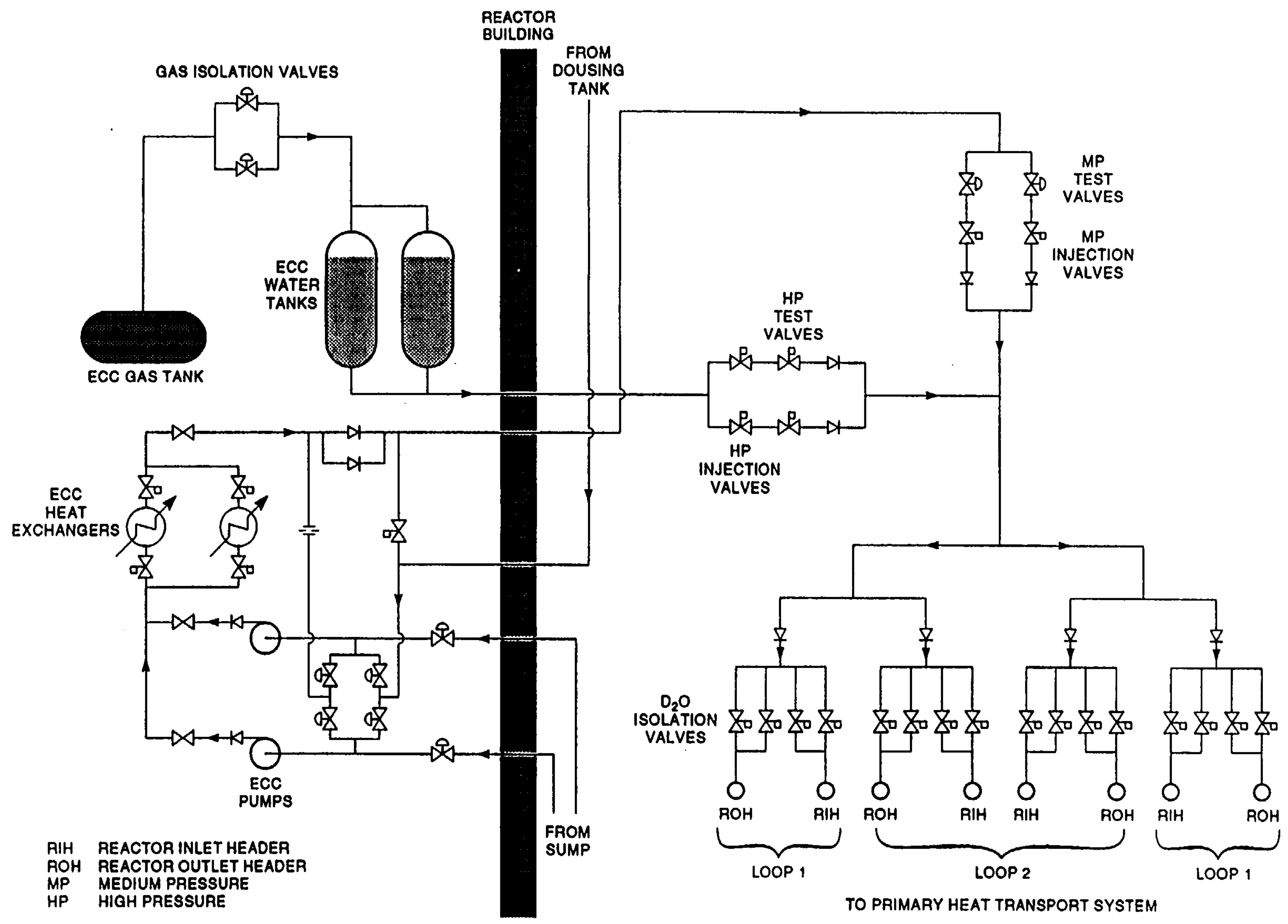


FIGURE 1 CANDU 6 EMERGENCY CORE COOLING SYSTEM

2.1 High Pressure Emergency Core Cooling Stage

One gas tank and two water tanks, located outside the reactor building, supply high pressure emergency cooling water to the reactor. The water tanks are normally isolated from the gas tank by two pneumatically-operated valves and maintained at low pressure in order to minimize gas dissolution, which could be detrimental to fuel cooling. The gas isolation valves are in parallel to assure system reliability.

Two normally closed isolating valves, known as the high pressure injection valves, are provided in parallel to isolate the high pressure system from the injection piping downstream. Two normally open valves, in series with and downstream of the high pressure injection valves, are provided to facilitate testing of the high pressure injection valves. Before testing any high pressure injection valve, the test valve in series with it is first closed to protect the downstream rupture discs. When the testing is complete, the high pressure injection valve is returned to the closed position and the corresponding test valve to the open position. By having a test valve in series with each of the high pressure injection valves, testing can take place without blocking out the high pressure injection flow path to the reactor core. All four valves, i.e. the two high pressure injection valves and the two test valves, are closed at the end of the high pressure injection stage in order to prevent gas injection.

Two rupture discs, one in each of the common injection lines, are used to provide a physical separation between the light water in the ECC system and the heavy water in the heat transport system. Each of the injection lines to the reactor headers contain two motorised valves in parallel, known as the D_2O isolation valves. They are normally closed to isolate the heat transport system from the ECC system. Two parallel valves are provided to ensure reliability.

2.2 Medium Pressure Emergency Core Cooling Stage

Two 100% ECC pumps, located outside the reactor building, supply water from the dousing tank to the reactor core in the medium pressure stage. A normally closed valve is provided on each pump suction line from the dousing tank to prevent the rapid loss of water from the dousing tank via an inadvertently open valve on the pump suction line from the reactor building basement. The pump discharge is directed to the heat transport system via the ECC heat exchangers and the medium pressure injection valve station. Two parallel, normally closed valves, called the medium pressure injection valves, provide the isolation of the medium pressure injection system from the downstream piping during normal reactor operation. Two normally open test valves, in series with and upstream of the medium pressure injection valve, are used during the testing of the medium pressure injection valves in a manner similar to that described earlier for the high pressure injection valves. With the provision of the test valves, testing of the medium pressure injection valves can take place without blocking the flow path to the reactor core.

2.3 Low Pressure Emergency Core Cooling Stage

The long-term low pressure emergency core cooling stage utilizes the same ECC pumps as the medium pressure stage, but recovers the D_2O and H_2O mixture collected in the basement of the reactor building and pumps it back to the heat transport system via the ECC heat exchangers.

Each ECC pump takes suction from the reactor building basement via a separate suction line containing a normally closed pneumatic valve. This valve acts as a containment isolation valve and prevents the water in the ECC system from draining into the reactor building.

2.4 Valve Operation

A number of valve actions are required for the successful operation of the CANDU 6 ECC system, as indicated below:

- High Pressure ECC
  - Open one gas isolation valve.
  - Open one high pressure injection valve.
  - Open one D_2O isolation valve to each of the reactor headers.
- Medium Pressure ECC
  - Open isolating valves on the suction lines from the dousing tank.
  - Open one medium pressure injection valve.
  - Close one valve on each of the high pressure injection paths in the high pressure injection valve station.
- Low Pressure ECC
  - Open isolation valves on pump suction lines from the reactor building basement.
  - Close isolation valves on the suction line from the dousing tank.
  - Open cooling water valves to the ECC heat Exchangers.

2.5 Performance

The CANDU 6 ECC system, through the provision of appropriate redundancy, meets all the requirements set for the system. The system, however, incurs relatively high capital and maintenance costs. Since the system does not operate during normal reactor operation, regular on-power testing must be performed to meet the unavailability target of 10^{-3}. A complete system operation cannot be tested on-power, therefore, a series of overlapping tests, generally on a monthly basis, are conducted to check sub-system operation, including valve operation. This increases operating cost. The incentive to reduce the above costs prompted the studies into ECC simplification.

3. THE CANDU 9 EMERGENCY CORE COOLING SYSTEM

The CANDU 9 ECC system utilizes the same high pressure, gas-driven water tank concept employed on the CANDU 6, Bruce and CANDU 3 power plants. However, the design is modified to significantly reduce the number of valves and achieve other system simplifications. A simplified flow diagram of the CANDU 9 ECC system is shown in Figure 2.

The principal improvements and simplifications made in the CANDU 9 ECC system include the following:

- Replacement of D_2O isolation valves with one-way rupture discs.
- Elimination of high pressure injection and test valves and incorporation of a floating ball seal in the water tanks.
- Location of additional ECC components inside the reactor building.
- Elimination of the ECC system medium pressure stage.

Each of these modifications is described in more detail below.

3.1 Replacement of D_2O Isolation Valves with One-Way Rupture Discs

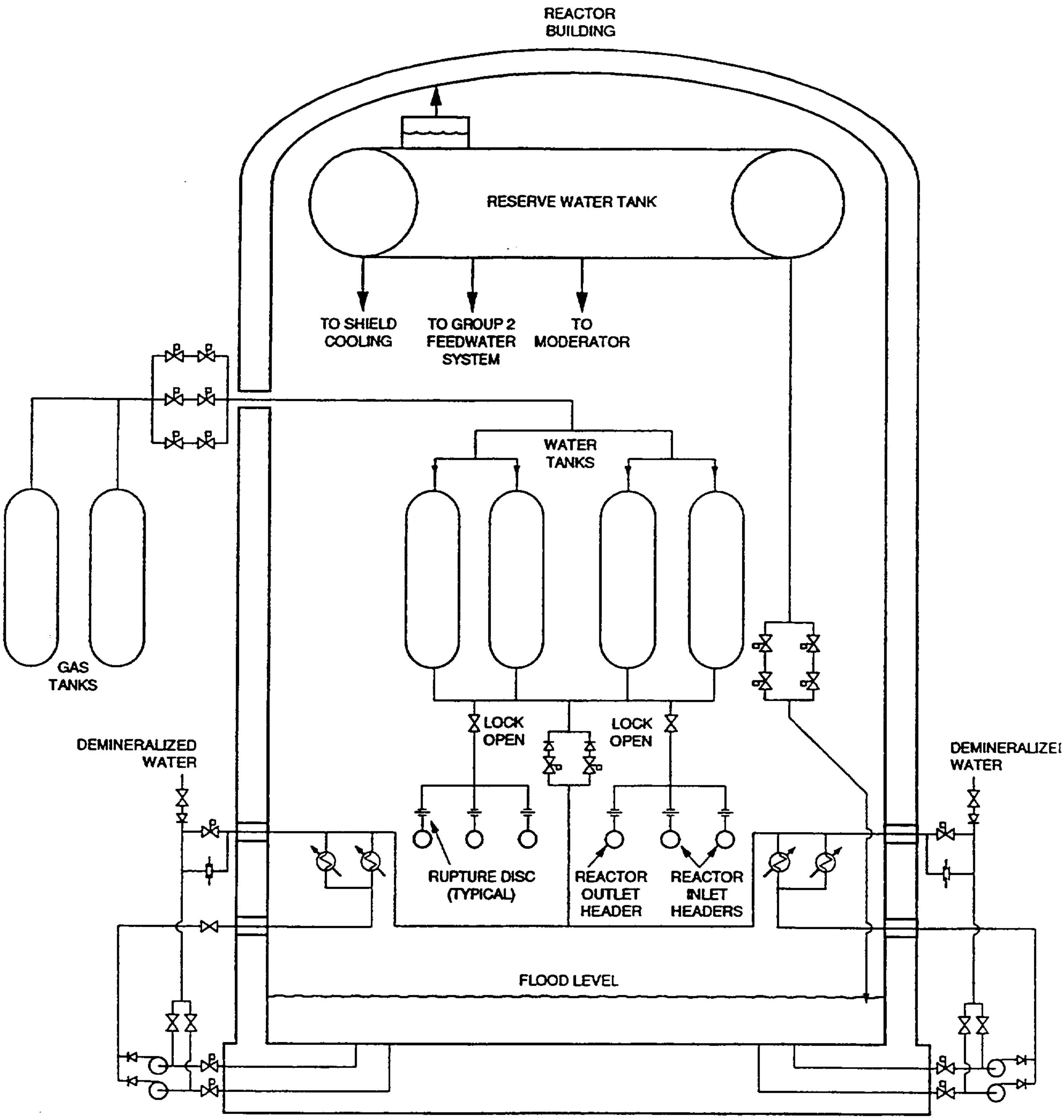
In the current CANDU 6 reactors, normally closed D_2O isolation valves are provided at the interface of the ECC system and the heat transport system. Low pressure rupture discs upstream of the D_2O isolation valves provide separation of light water and D_2O. The D_2O isolation valves isolate the high pressure heat transport system from the ECC system, which is normally at low pressure. The D_2O isolation valves open on a LOCA signal to permit injection of water from the water tanks.

CANDU 9 ECC system uses a specially designed, one-way rupture disc in lieu of the D_2O isolation valves to isolate the heat transport system from the ECC system (Figure 3). These rupture discs fail at low differential pressure in the forward direction (ECC system to heat transport system), but withstand the high differential pressure that is normally present in the reverse direction (heat transport system to ECC system).

The use of one-way rupture discs instead of the D_2O isolation valves provides significant benefits, including system simplification, higher system reliability, less testing due to fewer active components, reduction of D_2O hold-up and associated costs, and lower construction, capital and maintenance costs.

3.2 Elimination of High pressure Injection Valves and Incorporation of Floating Ball Seals in Water Tanks

The high pressure injection valves on the CANDU 6 ECC system are closed during normal reactor operation to protect the downstream rupture discs from rupture during system testing. These valves are required to open on a loss of


REACTOR
BUILDING
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TO GROUP 2
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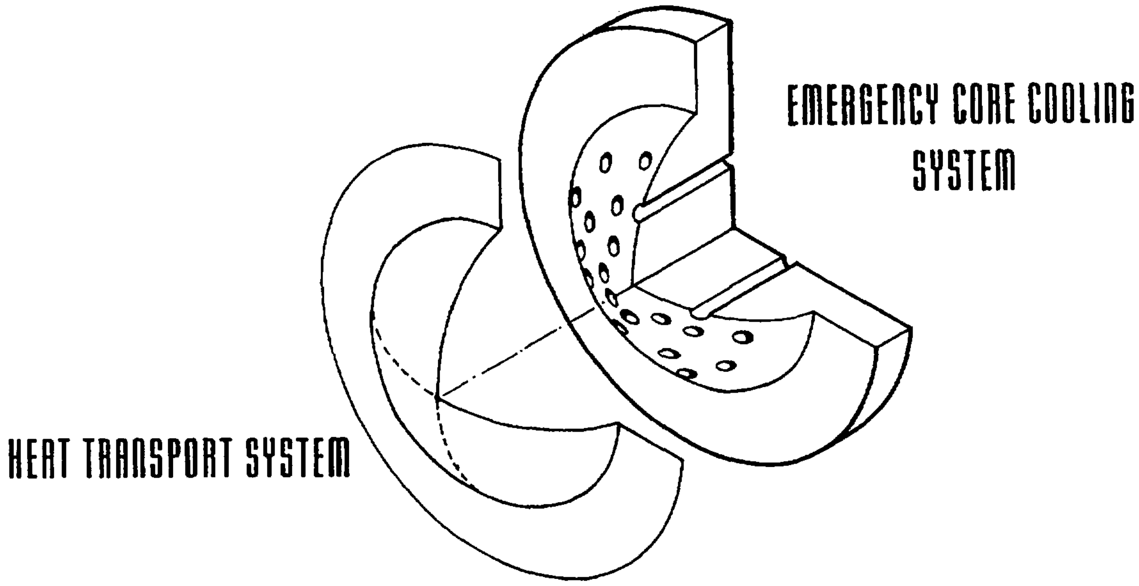

FIGURE 3: ONE-WAY RUPTURE DISC

coolant accident signal to admit emergency coolant into the reactor core from the water tanks and close on low water tank level signal at the end of the high pressure injection to prevent gas injection. The specially designed rupture discs for the CANDU 9 ECC system, described in Section 3.1, do not require protection to prevent rupture during normal plant operation. This allows the elimination of the high pressure injection valves and their associated test valves on the CANDU 9 ECC system.

Gas injection at the end of the high pressure injection on CANDU 9 is prevented by the action of the floating ball seal in each water tank (Figure 4). These floating ball seals, which normally float at the top of the water tanks, seal against the outlet of the tanks at the end of the high pressure injection.

Elimination of the high pressure injection valves and their associated test valves reduces the operating costs due to maintenance and testing, in addition to a reduction in the capital cost.

3.3 Location of Additional Emergency Core Cooling System Components Inside the Reactor Building

The high pressure equipment consists of two high pressure gas tanks located outside the reactor building and connected via a valve station to the top of four high pressure water tanks inside the reactor building. The water tank outlets join a distribution header from which two separate lines symmetrically feed the reactor headers (two inlet headers and one outlet header at each end of the reactor). This modification in the design reduces the piping length between the water tanks and the reactor headers and hence, the flow resistance during the injection stage.

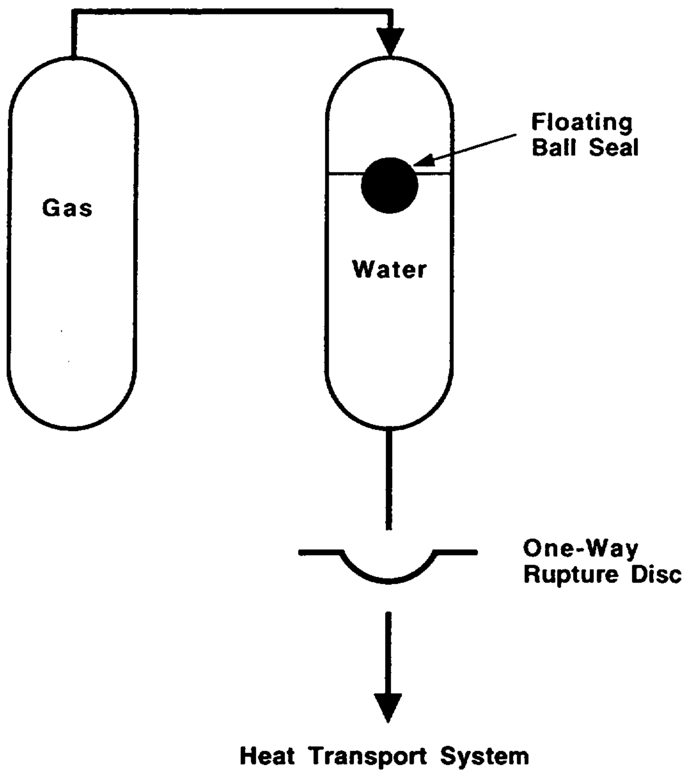

FIGURE 4: FLOATING BALL SEAL ARRANGEMENT

In the CANDU 9 design, the ECC system heat exchangers are located inside the reactor building. This increases the overall system reliability since a potential leak from the ECC system heat exchanger to the environment does not require the isolation of the heat exchanger, thereby allowing it to remain in service. The location inside the reactor building also reduces potential demands on the collection system serving the ECC system equipment located outside of the reactor building.

3.4 Elimination of the ECC System Medium Pressure Stage

In CANDU 9, downcomers from the reserve water tank feed the ECC system water from the reserve water tank to the ECC system sumps. The valves in the downcomers open on a LOCA signal. The ECC system water from the reserve water tank tempers the water released by the heat transport system break (LOCA) and floods the reactor building floor to a depth of about 1.5 metres, thereby assuring net positive suction head to the ECC system recovery pumps. This arrangement significantly reduces the number of valve actions required to provide post-injection stage fuel cooling, thereby increasing system reliability and reducing operation and maintenance costs.

3.5 Flow Control Component Operation

The number of ECC system flow control components (valves, check valves and rupture discs) in the CANDU 9 design has been reduced to less than half of the number in CANDU 6 plants. Consequently, the valve actions required to operate the CANDU 9 emergency core cooling system are also greatly reduced, and the reliability is increased.

- Injection Stage ECC
  - Open gas isolation valves
    - Rupture discs automatically burst when the heat transport system depressurizes to about 0.5 Mpa below the ECC system pressure.
    - Floating ball seals automatically prevent gas from being injected into the heat transport system once the water tanks are almost empty.
  - Open valves in downcomer from the reserve water tank.
- Recovery Stage ECC
  - start ECC system recovery pumps.
  - Open recovery stage ECC valves.

4. AECL DEVELOPMENT PROGRAMS AIMED AT SIMPLIFYING ECC SYSTEMS

A two year development program is near completion at the AECL Chalk River Laboratory for the one-way rupture discs and floating ball seal. These development programs involve conceptual assessment, bench scale tests, full size prototype tests and full size simulated service tests.

These programs are described below.

4.1 One-Way Rupture Disc

The CANDU 9 ECC rupture disc application requires a ratio of about 50 to 1 between the pressure differential required to fail the disc in the reverse direction (HTS to ECC) and the pressure differential to fail it in the forward direction (ECC to HTS). Also, the disc must be robust and avoid fragmentation under high flow conditions (~1000 kg/s); loose pieces could cause problems in the rest of the reactor.

The approach chosen is to support the disc with a robust perforated plate on the ECC side (see Figure 3). The support prevents the disc from rupturing in the reverse direction during normal operation due to heat transport system pressure pushing the disc towards the unpressurized ECC system. Due to the concern about spurious rupture of the disc, it has been designed to support reverse pressures at least double the normal heat transport system pressure.

Another requirement is that the disc be capable of supporting 0.2 MPa forward pressure differential. This situation could occur when the heat transport system is partly drained for maintenance. Under these conditions, the pressure on the ECC side of the disc is 0.2 MPa due to the head of water in the tanks, and the heat transport side of the disc is exposed to the atmosphere.

When the ECC system is actuated during a LOCA, the disc will burst in the forwards direction (ECC into heat transport system) once the heat transport system depressurizes to about 0.5 MPa below the ECC system pressure. To permit the very high flow rates required during the high pressure injection phase, the disc support contains a large number of holes, with the total area of the holes approximately equal to the flow area of the pipe.

The development program consisted of conceptual design and feasibility testing, special effects tests and analysis, small-scale tests, and full-scale tests.

Special effects tests and analysis were done to investigate the behaviour of the disc near one of the holes in the support, as a function of disc thickness and hole size.

Small-scale testing had the following purposes: to optimize the design, to verify the forward and reverse characteristics of the disc, to verify that there would be negligible fatigue damage due to pressure cycling, and to measure the pressure drop characteristics of the disc as a function of flow rate. These tests were done with 0.1 m diameter discs. A brief summary of results follows:

- Forward bust pressure: 0.6 MPa
- Reverse burst pressure: 48 MPa
- Fatigue testing (alternating forward and reverse pressure differential):

  0.0 to -11.7 MPa, 1000 times
  +0.2 to -11.7 MPa, 500 times

- Flow testing:
  - no fragmentation at 70 kg/s (same flow velocity as ECC maximum)
  - pressure differential after rupture is 0 to 0.3 MPa depending on flow history, as shown in Figure 5.

Full scale testing on 0.5 m diameter discs is underway. A prototype has been fabricated and preliminary burst tests are complete, and full-scale flow tests on a disc support at a flow rate of 1000 kg/s have been done; these tests use the ECC system from AECL's NRX reactor in Chalk River, which is no longer in service. Future testing of production discs will verify the opening characteristics of the discs, and demonstrate that they will not fragment at these high flows.

4.2 Floating Ball Seal

The challenge of designing a floating ball capable of resisting large external pressures is similar to the problem of designing deep-sea submersible vehicles;

however, the ECC system application has the advantage of not requiring cargo space inside the floating ball.

Hence, the CANDU 9 floating ball seal is made of modern plastics. It has a lightweight, but stiff, inner foam core with a thick outer skin that is extremely resistant to mechanical damage and which conforms well to the seat at the bottom of the tank, when the tank is almost empty.

The development program consisted of three phases: exploratory finite element stress analysis of many different concepts, small-scale testing, and full-scale testing.

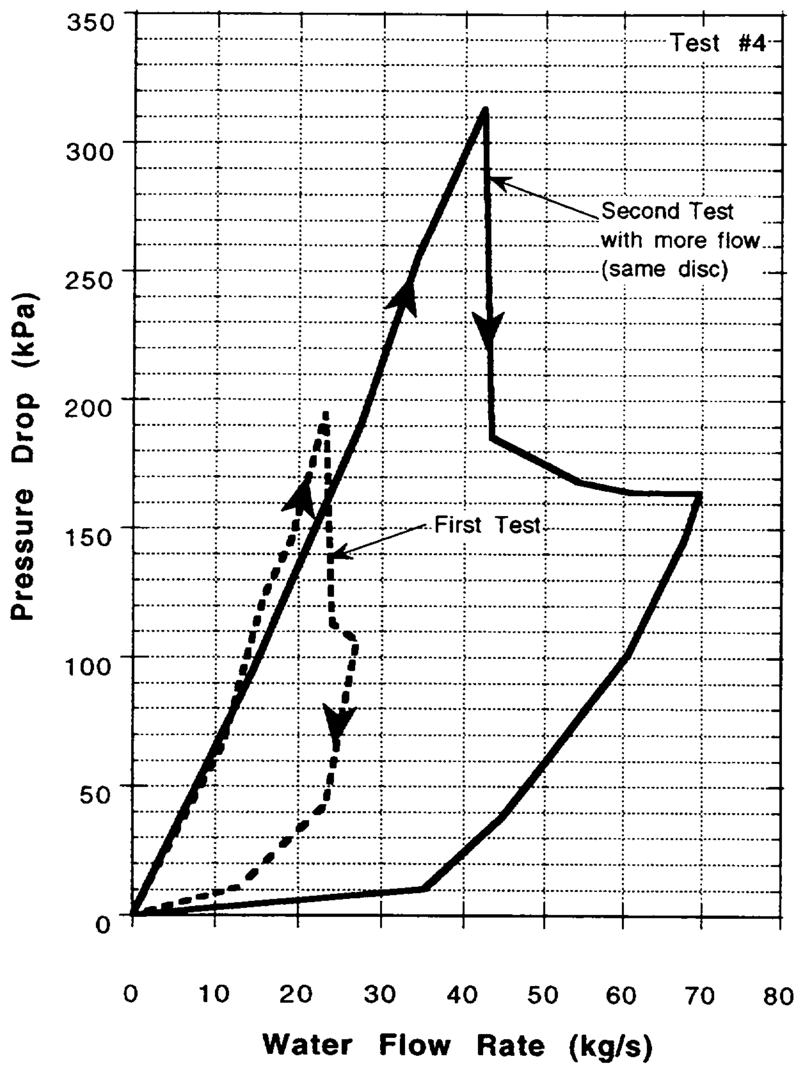

FIGURE 5: RUPTURE DISC PRESSURE DROP

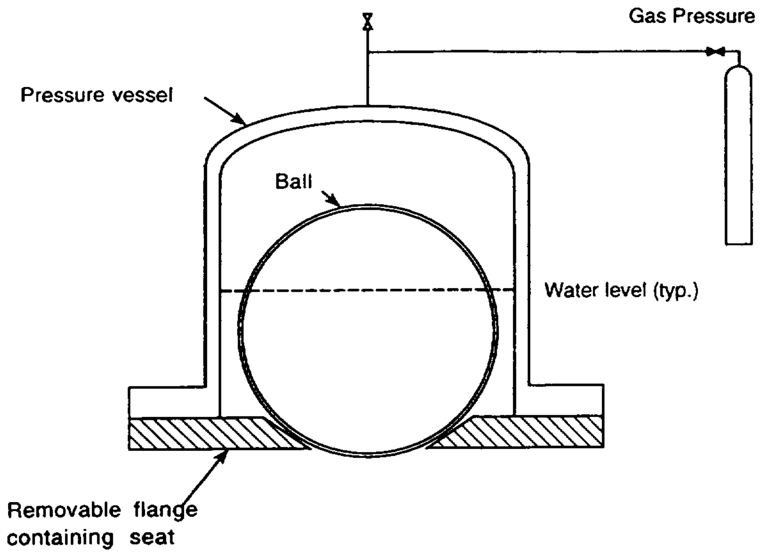

FIGURE 6: FACILITY FOR FULL-SIZE BALL SEAL TESTS

The stress analysis showed that the highest loads were in the contact area, and the ball needed to be about twice the diameter of the hole that it was to seal. The shape of the seat was optimized to reduce stress in the ball while still obtaining high sealing forces.

Small-scale static sealing tests were done on a 0.2 m diameter ball in water. These tests verified the analytical stress analysis and demonstrated that the concept would work. Tests were done over a wide range of pressures to determine the strength, deflection, creep and re-usability of the ball. Leakage past the floating ball seal was negligible.

Similar full-scale static sealing tests were done on a 0.6 m diameter ball in water, as shown in Figure 6. Again, the analytical predictions were verified. The ball is capable of sealing the bottom of the ECC water tanks once the injection stage is complete. It is also capable of sealing full heat transport system pressure against the port in the top of the tank, to prevent the heat transport system coolant from flowing backwards through the ECC system in the event of multiple failures of other components. Since the ball is much more flexible than most engineering components, the deflection can be significant, up to 5 cm under some conditions, excellent sealing is assured.

5. CONCLUSION

The CANDU 9 emergency core cooling system offers a significant improvement in the overall reliability while reducing capital, operations and maintenance costs. Options for further emergency core cooling system enhancement, including the use of steam instead of pressurized gas, continue to be studied and will be adopted when proven.

DEVELOPMENT OF A SECONDARY SHUTDOWN SYSTEM FOR THE 500 MW(e) PHWR

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Abstract

Pressurised Heavy Water Reactors (PHWRs) of advanced design are supposed to have two independent fast acting shut down mechanisms in order to cater to the present day safety requirement of defense in depth. The 500 MW(e) Indian PHWR is one such reactor which is being designed to have two independent shut down systems functioning on diverse principles. Of these, the second system referred to as the secondary shut down system introduces poison into to the moderator in the form of jets. This system is under development in Reactor Engineering Division (R.E.D.).

The system conceived consists of a vessal containing pressurised helium connected to poison tanks through quick acting solenoid valves. the tanks are connected to horizontal injection tubes in the calandria. On actuation, gadolinium nitrate ($GdNO_3$) solution from the tanks passes to the injection tubes which have a number of holes through which poison enters the moderator.

An experimental facility was set up to measure the poison jet growth rate and the jet spread. A formalism to estimate the reactivity worth introduced by these jets has also been developed by R.E.D. The description of the system designed, the experimental studies and the calculation based on these experiments is presented in this paper.

1. INTRODUCTION

The 500 MW(e) Indian PHWR which is in an advanced stage of design has two independent shut down systems operating on diverse principles. The first consists of mechanical shut off rods called SDS-1 which is a group of 28 vertical shut off rods which provide a shutdown worth of approx. 55 mk when two of the maximum worth rods of the set fail to actuate. The second, called SDS-2, is a liquid poison injection system which consists of six horizontal poison injection tubes with holes in them. When the poison solution at high pressure is pumped through these tubes, the poison enters the bulk moderator in the form of high speed liquid jets [1].

The engineering development of the system was mainly done in two stages. In the first stage, experiments were done to study the jet propagation with two different hole geometries namely, holes and slits. This stage also involved optimisation and prediction of system parameters. The second stage consisted of experiments on full scale mock-up where, the actual conditions seen in the reactor was simulated [2].

The physics development involved the calculation of the reactivity worth of these jets and a detailed safety analysis. A formalism has been developed for estimating the reactivity worth of the system. This method is different from what currently is in practice where the jets are cylindricalised with increasing radius as it penetrates into the moderator [1]. The geometry of the jet and its characteristics have been taken more accurately and the system parameters have been determined.

2. DESCRIPTION OF THE SYSTEM

The SDS-2 has been visualised as consisting of six horizontal poison injection tubes with holes drilled into them, so that when the system is triggered, it injects a concentrated solution of Gadolinium Nitrate directly into the moderator. These tubes are in a direction perpendicular to the pressure tubes.

The experimental set-up consists of a high pressure helium gas vessel connect to six liquid poison tanks through a bank of six quick opening solenoid valves (Fig. 1). Each poison tank is connected to a horizontal injection tube located in the calandria. The six tubes are along the intersection of two horizontal planes and three vertical planes (Fig 4a). Each injection tube has several holes (336 in all) distributed throughout the tube. The holes for injection are placed in clusters of 16 between two calandria tubes i.e, a lattice pitch. The cluster of 16 holes is further divided into 4 sets of 4 holes, each set being 3 cm apart. The holes are at an angular separation of 90 . Alternate set of holes are turned by 45 in order to have a more uniform distribution of the poison within the calandria (Fig. 4b).

The actuation of two out of three trip channels will open the solenoid valves. Pressurised helium rushes into the poison tanks and injects the liquid poison in the form of high speed submerged jets into the moderator. Each poison tank is provided with a solid soft floating ball. When the injection is completed, the ball will reach to the tank bottom and block off the outlet of the poison tank.

There is a liquid to liquid interface between the poison and the moderator. Over a period of time the interface moves by diffusion from high density poison solution towards the calandria. The movement of the interface is monitored by conductivity probes.

The jet growth will depend on the size of the hole and the pressure with which the liquid is injected into the moderator. The negative reactivity will depend on the volume of the poison and the poison distribution. the poison concentration in the jet will be a continuously varying function. A series of experiments were done to study the jet progression. Parameters like the shape of the jet,the rate of jet growth as a function of time, the discharge rate of the poison etc. were measured. the results were compiled as jet progression curves for different injection pressures and nozzle dimensions.

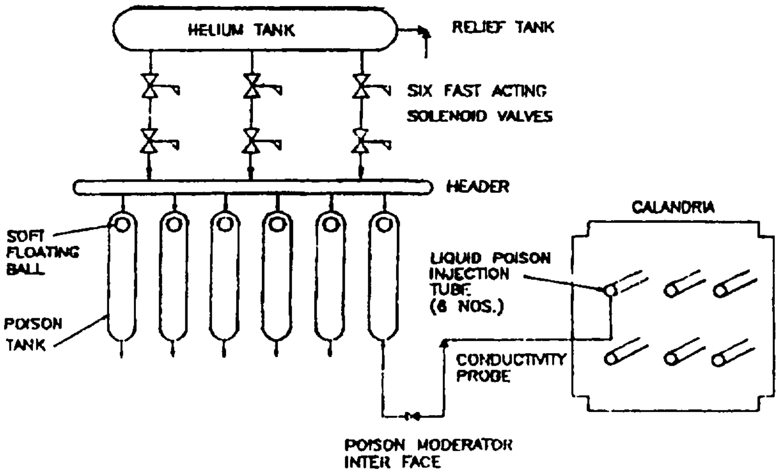


Fig. 1 Flow diagram for liquid poison injection system

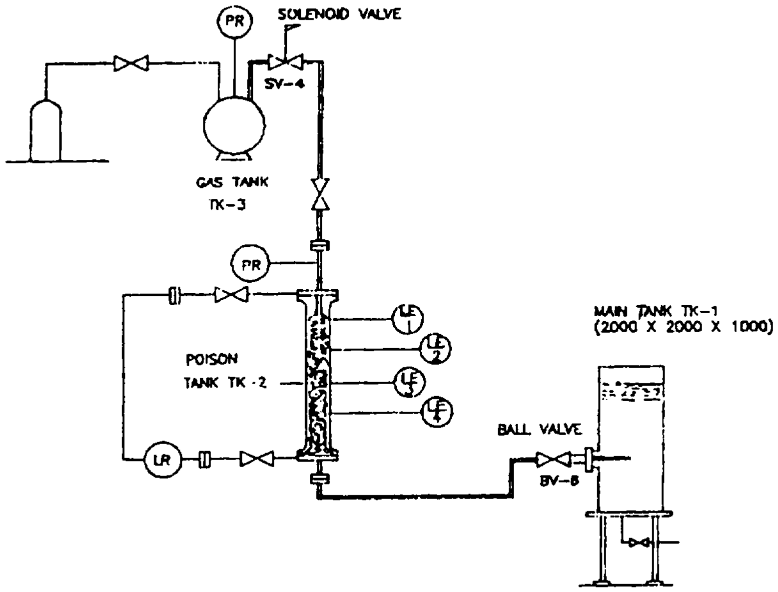


Fig. 2 Phase 1 set-up

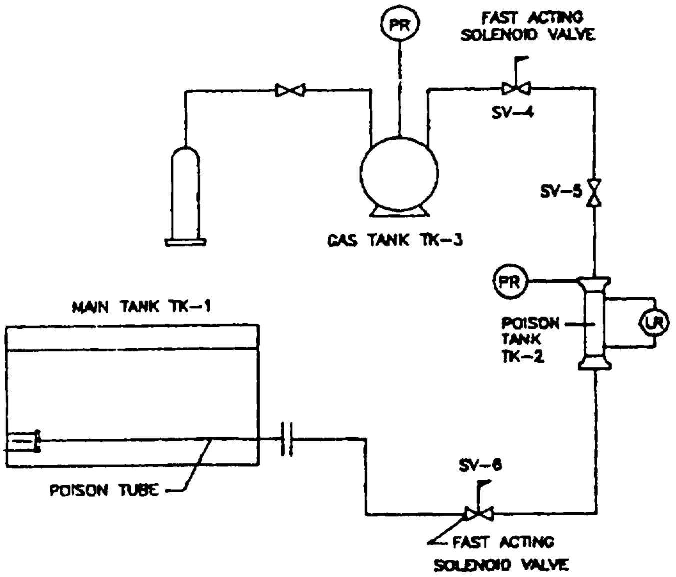


Fig.3 Phase 2 test set-up

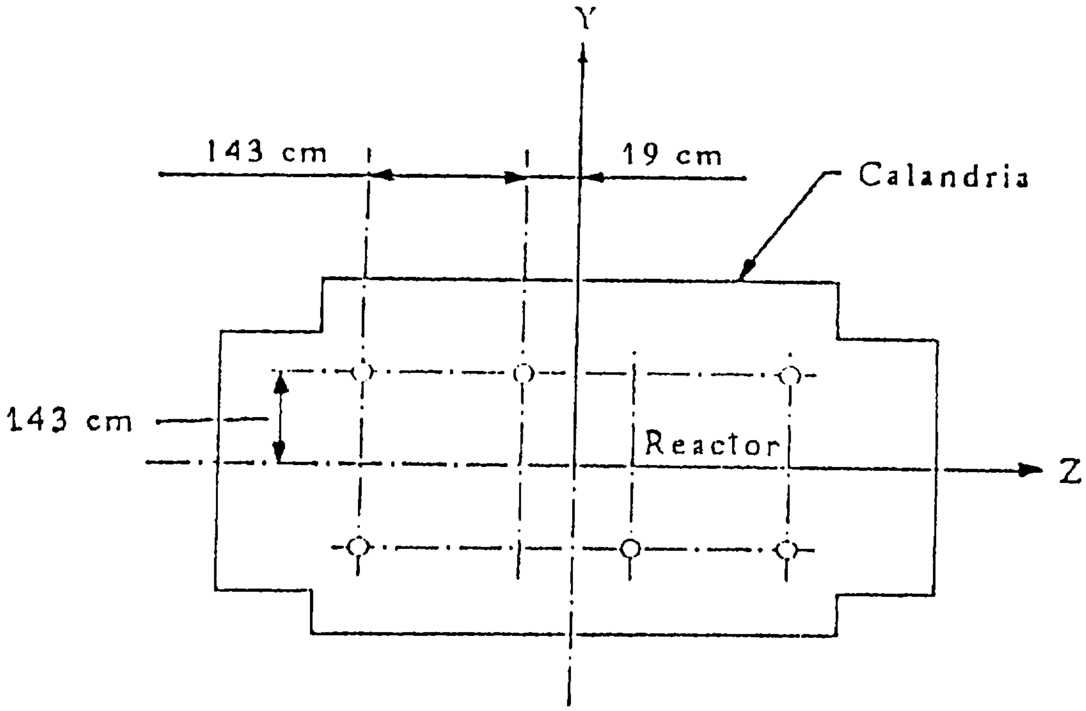


Fig. 4a Location of poison tubes in the calandria

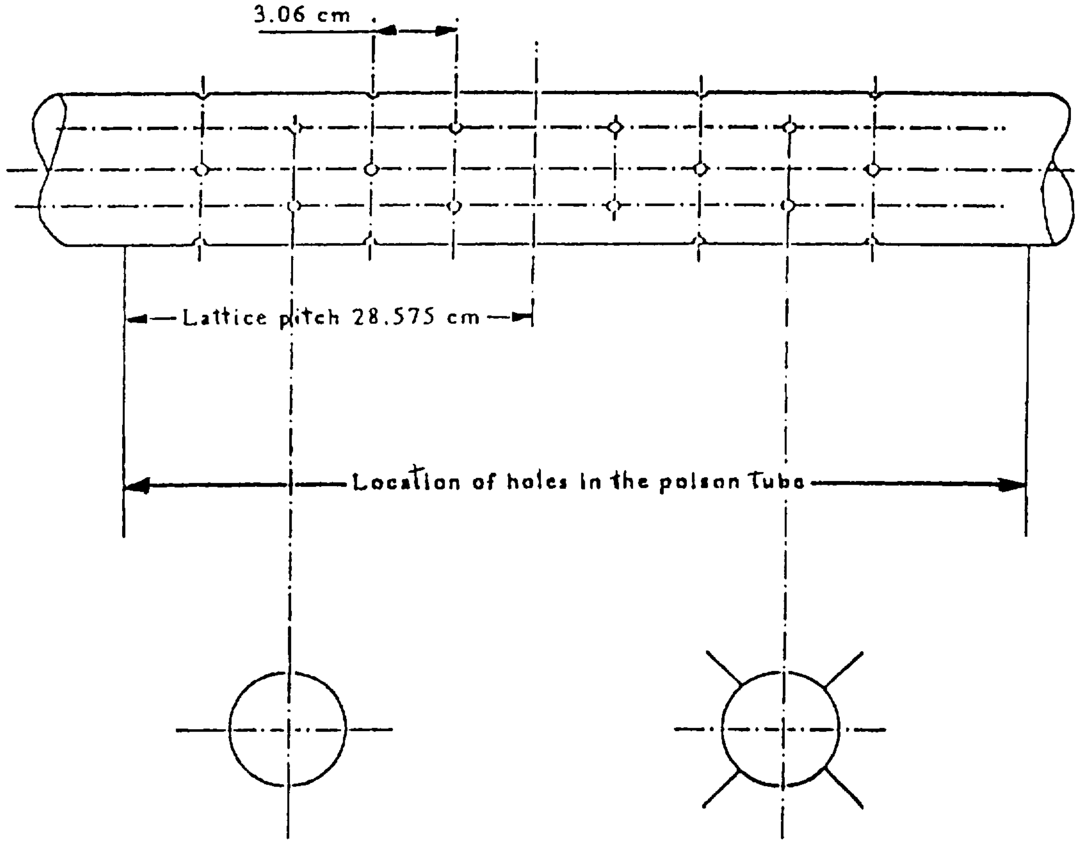


Fig. 4b Jets from alternate sets of holes

3. DEVELOPMENT STUDIES

For the development studies, various experiments were planned into two phases.

(a) Phase 1 : The main aims of the experiments were to study the spread, penetration and growth rate of the single jets for the calculation of reactivity insertion rate, optimising the hole size and predicting the performance of the full integrated system. The experiments under phase-1 have been completed.

Based on the results of phase-1 experiments, the performance of full integrated system in terms of process and nuclear parameters has been predicted.

(b) Phase 2 : The main aims of the experiments are to verify the predicted performance on a full scale mock up. In addition separate studies on solubilities, precipitation and diffusion are also being done. Various experiments under phase-2 have been conducted and still some are being carried out. The data on poison front movement and jet growth have been generated and being converted into equivalent reactivity worth values.

3.1. Description of the Phase 1 test set up

The experimental set up for the phase 1 studies shown in Fig. 2 consists of a rectangular main tank TK-1 with two sides made of transparent acrylic sheets for the observation of the jet(s). Facility for the interchangability of various injection nozzles has been provided. This tank is filled with ordinary water after installing a suitable nozzle with hole(s)/slit(s) in it. The poison is simulated with colour water (blue ink solution) for easy visibility. The data on various parameters such as jet length v/s time, jet growth rate and discharge rate was generated for about 1 second.

3.1.1. Phase 1 Experiments

Following experiments have been carried out.

(a) Single hole experiments for hole sizes from 2 to 8 mm diameter at various pressures have been performed to study the spread angle, jet growth rate and discharge rate.

(b) Single slit experiments for slits with length from 7.5 to 15 mm and width from 0.5 to 1 mm at various pressures have been performed to study the spread angle, jet growth rate and discharge rate.

(c) The multiple holes/slits experiments for 16 and 18 number of holes/slits have been performed at different pressures to study the poison front movement, growth rate and discharge rate.

3.1.2. Data Collection

The data on the discharge rate, spread angle and jet growth rate have been collected.

(a) Discharge Rate :

With the help of a fast recorder the fall in the poison tank level was recorded and from this discharge rate as well as initial jet velocity was calculated.

(b) Spread Angle Measurement :

The nature of the jet for the round hole is conical and the jet at any time is a frustrum of a cone. In the slit jet, the jet at any time is a frustrum of the pyramid. The spread angles have been measured from still photographs taken. Table 1 shows the measured values of spread angles for holes and slits.

(c) Jet Growth Equation :

The jet growth data has been generated from the video photographs. The recording speed of the video camera is 25 frames/sec, which gives 0.04 sec time for each photograph (frame). Thus jet growth study is made by observing the video photographs at the interval of or the multiples of 0.04 sec. Based on the results the jet length at time t appears to be :

$$t = \frac{Y^2}{C \sqrt{Ao}\, Vo} \qquad \text{for } Y < 140 \text{ cm}$$

where Y = The length of the jet in cm.
C = Constant and is a function of pressure.
Ao = Cross sectional area of the hole/slit in sqcm.
Vo = Initial velocity of the jet in cm/sec.
t = Time in sec.

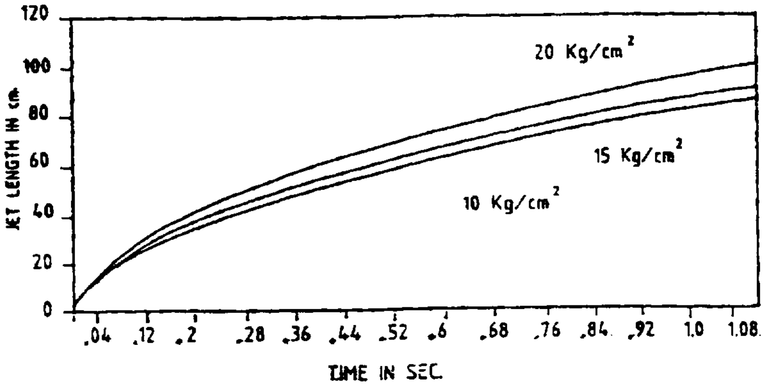
The jet development is shown in Graphs 1 and 2.

3.1.3. Discussion of Phase 1 Experiments

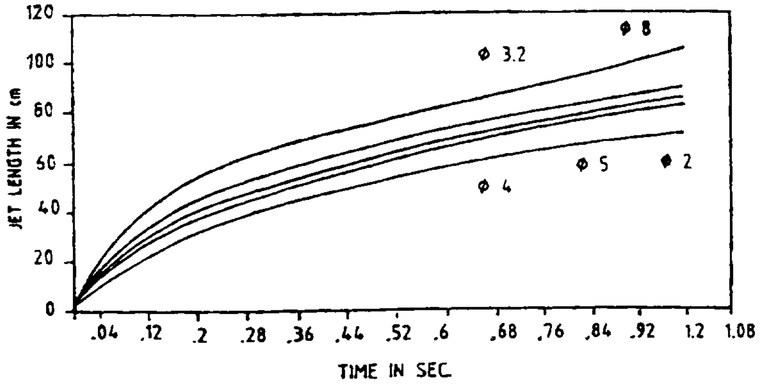
It is observed that for the same area of the hole or slit, (for the same jet length) the volume of the slit jet is more than the circular jet. But the jet growth rate for the slit is slower than the circular hole. The machining of the slits is a more involved procedure. Therefore circular holes are preferred over slits. If the holes of the diameter smaller than 3.2mm are used then the time for discharging the full quantity of the poison increases too much and poison front advances comparatively slower. If the diameter of the holes is greater than 4mm then the possibility is that the poison may not be reaching to the last pitch within a second because of high discharge rate. Thus we can say the optimised diameter for the holes is 3.2 to 4 mm. Based on the poison front equation and jet growth experiment, the choice is either 3.2mm or 4mm dia holes (336 nos.) uniformly distributed in 21 pitches for the reactor system.

3.2. Predicted system performance

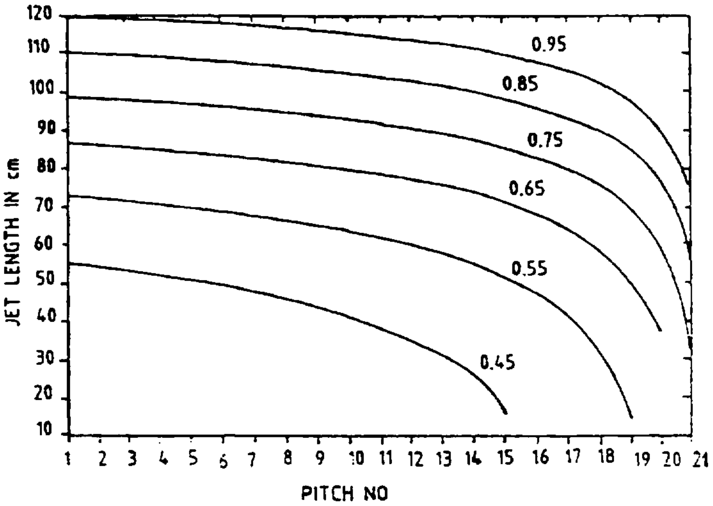
A second order differential equation has been formulated for generating the data on poison front movement [3]. Based on the results of phase 1 experiments and the poison front movement equation, the performance of the system has been predicted with the help of following process parameters.



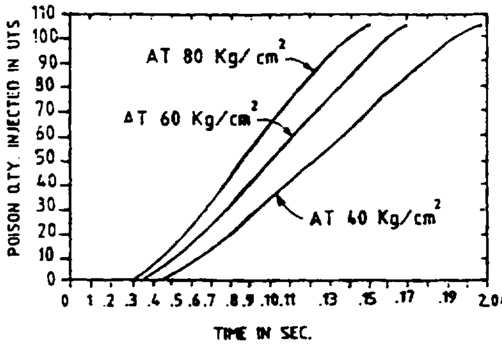
Graph 1 Jet development of ϕ 3.2 hole at various pressure



Graph 2 Jet development for single holes at 10 kg/cm^2



Graph 3 Jet of 3.2 mm diameter holes (336 NOs) at 80 kg/cm^2 (predicted results)



Graph 4 Poison injection vs. time (predicted results)

i) Jet length v/s time (Graph 3)
ii) Quantity of poison injected v/s time (Graph 4)

The predicted results are being verified experimentally in a full scale test set up (phase 2) for one injection unit.

3.3. Description of the Phase 2 test set up

The predicted results are being verified in a full scale set up for one injection unit. The set up consists of the main tank TK-1 and the corresponding circuit for poison injection is shown in Fig. 3 with one side transparent for the observation of the jets. The experiment differs from the actual reactor system is that all the four holes in one cross section lie in a quadrant with an angular se[paration of 30 deg. The poison is simulated with colour water for easy visibility. The poison tank is suitably instrumented with level probes and timer units to note the fall of poison level. The continuous change in the poison level is also recorded by a fast recorder. The jets are photographed by the high speed camera/video camera for the analysis purpose. Experiments will also be conducted to check the flow induced vibrations of the calandria tubes and the poison tubes due to entrainment of the surrounding fluid.

3.3.1. Phase 2 Experiments

The following experiments have been carried out.

a) 3.2mm dia., 336 No. of holes at various gas tank pressures from 7.0 to 100.0 Kg/Sqcm.
b) Leak tightness of solid floating ball and lower ball seat at various gas tank pressures (7 to 80 Kg/Sqcm).

The specially fabricated solid floating ball made of polypropelene was tested at various pressures from 7 to 80 Kg/Sqcm with conical and spherical design of ball seat. The testing done so far indicates the need for further improvements in the seat design to obtain perfect leak tightness of the ball against the ball seat.

The data on poison front movement, jet development and discharge rate have been collected. Videophotography was resorted to for data generation on poison front movement and jet development.

3.3.2. Discussion of Phase 2 Experiments

Due to constraints of the experiment for e.g. opening and closing of valve SV-6, in the experiment the time taken for the jet development is more than the predicted value. Experiments are in progress to establish this lag. This practical data of poison jet growth is being converted into equivalent reactivity worth values. From the present experiments it appears that to get the negative reactivity of 80 mk in the first second, certain design improvements may have to be made. Further experimental work is in progress.

4. CALCULATION OF REACTIVITY WORTH

The jets issuing from the poison tubes are all functionally similar. By the complexity of the design of the system, each jet is at a different orientation and position. Since the basis of reactor physics calculations is the reactor lattice cell, one has to calculate how these jets perturb the lattice cell. In order to calculate the effect of these jets in any reactor lattice cell, one has to know the orientation of the jet with respect to the cell. Since the volume of poison in a cell and its distribution are both functions of distance of the cell from the poison tube, we have assumed that the contribution to the extra absorption ($\delta\Sigma a$) from any jet is a function of distance of the cell from the origin of the jet.

4.1. Treatment of the jets

The formalism we have developed involves mainly three stages namely ;

(i) Treatment of one single jet in the moderator and calculation of the flux distribution over the jet and surrounding moderator.

(ii) Calculation of the contribution to the absorption of any reactor lattice cell using this flux distribution and the jet distribution in the reactor.

(iii) Calculation of the net absorption by adding the contribution duel to all such poisoned cells and hence estimation of the reactivity worth.

In order to determine the flux distribution over a moderator-poison domain, the jet has been approximated to two region problem defined by poison and its surrounding moderator. The jet development studies were done for two geometries i.e. slits and holes. The jet issuing out of slits would be trapezoidal or wedge shaped in nature and that through holes would be conical. The cross sectional area in both cases would be conical ignoring the width of the slits. So two independent formalisms were developed to treat the jets issuing out these two geometries. The jet spread was calculated from experiments.

4.1.1. Formalism developed for the slits and holes

In the case of slits, the flux distribution over this two-region domain is calculated by simulating the jet as shown in Fig.5a and b. The poison is confined to a sector made by half angle of the cone and the flux distribution is calculated in a plane perpendicular to the axis of the poison tube. The concentration variation has been considered explicitly.

In the case of holes, the poison has been confined to a central region of uniform poison concentration surrounded by an annular moderator region. The extent of the poison here would be the cross sectional area of the jet (Fig. 6a-d). The flux distribution is calculated in a plane perpendicular to the axis of the jet.

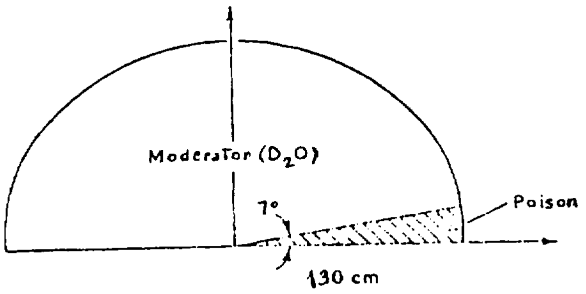


Fig. 5a

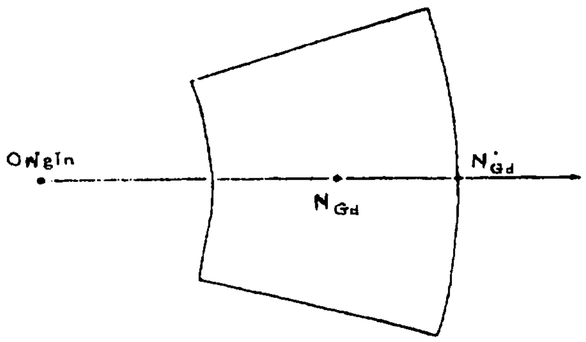
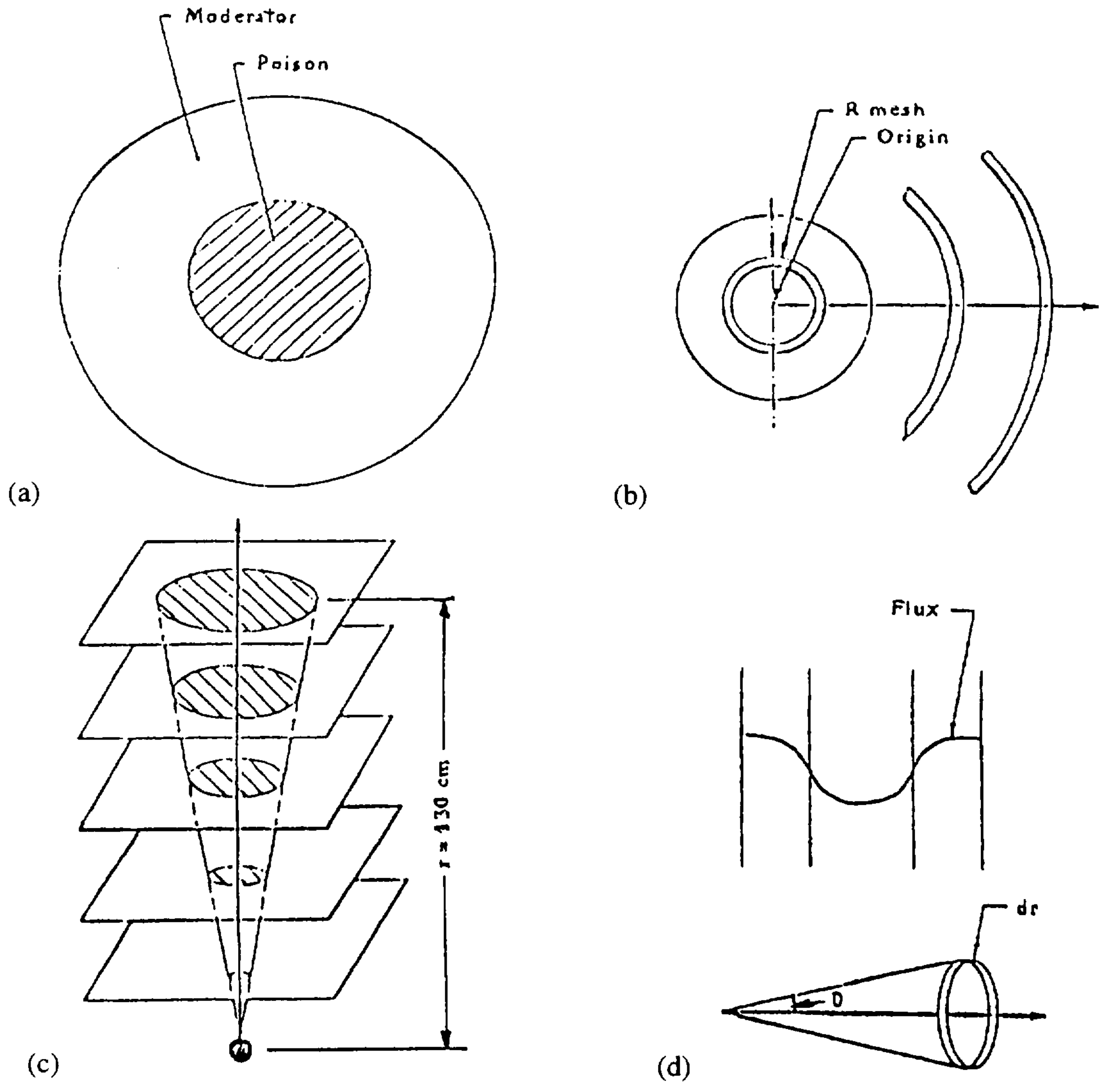


Fig. 5b R-θ mesh



(a) (b) (c) (d)

Fig. 6 (a)-(d)

4.1.2. Calculation of extra absorption in the cell

The contribution to the absorption at any distance from the poison tube is calculated by integrating the flux over the reactor lattice cell. The homogenised reactor lattice cell has been divided into thin strips, the extra absorption in each strip is calculated by integrating over space using the flux distribution obtained earlier. This is done throughout the length of the jet and the contribution to the absorption ($\delta\Sigma a$) is calculated as function of distance form the origin of the jet. We now have a parametric curve for the extra absorption over a cell due to the presence of poison as a function of distance from the poison tube (Graph 5).

4.2. Global Reactivity Calculations

The individual cells perturbed by the poison are identified using the jet distribution and the jet propagation profile obtained from experiments. The course of poison jets was then mapped over the reactor cell grid and the jet distribution within each cell was found. A cell could be perturbed by any number of jets and the length of the jet could be different depending upon its orientation. The total extra absorption of any cell is taken as the sum of the contributions due to all these jets as if they were present individually.

The global reactivity calculation were done using two group diffusion theory which involves the three dimensional core simulation of the 500 MW(e) PHWR. The neutron balance equations are solved over the finite reactor and the flux distribution and k-eff was calculated using finite difference methods. A code system has been developed for calculating the reactivity worth of the SDS-2.

Also some specific problems of interacting jets and terminated or hampered jets have been studied and corresponding correction factors have been included in the reactivity worth estimations.

5. ANALYSIS OF EXPERIMENTAL RESULTS

The reactivity worths have been determined as a function of time for different injection pressures and nozzle dimensions. However, it is the negative reactivity insertion during the first second that is of greater significance in safety evaluations. The worths have also been evaluated for different reactor loadings and in the presence of other reactivity devices such as adjusters and zonal control units.

5.1. Analysis of the Phase 1 Experimental results

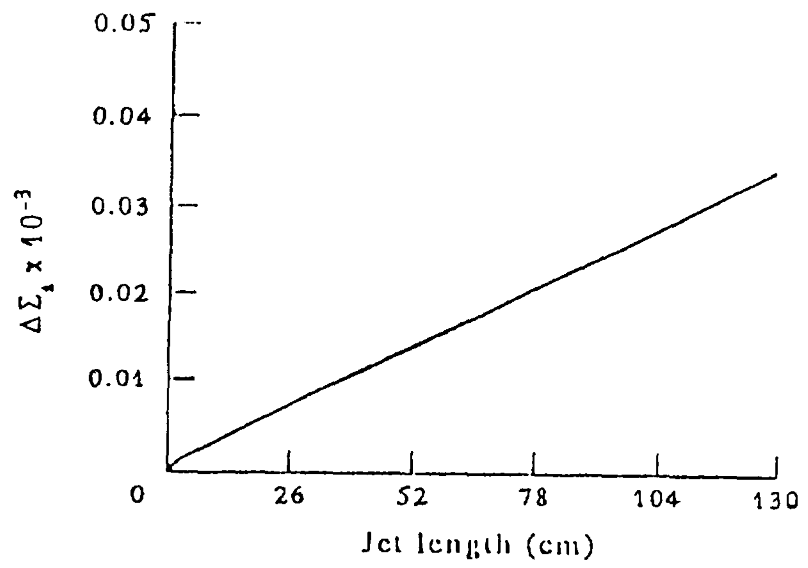
The following calculation have been done :

I. Reactivity worth as a function of time for a fresh core with all natural Uranium loading. No Xenon and no reactivity devices.

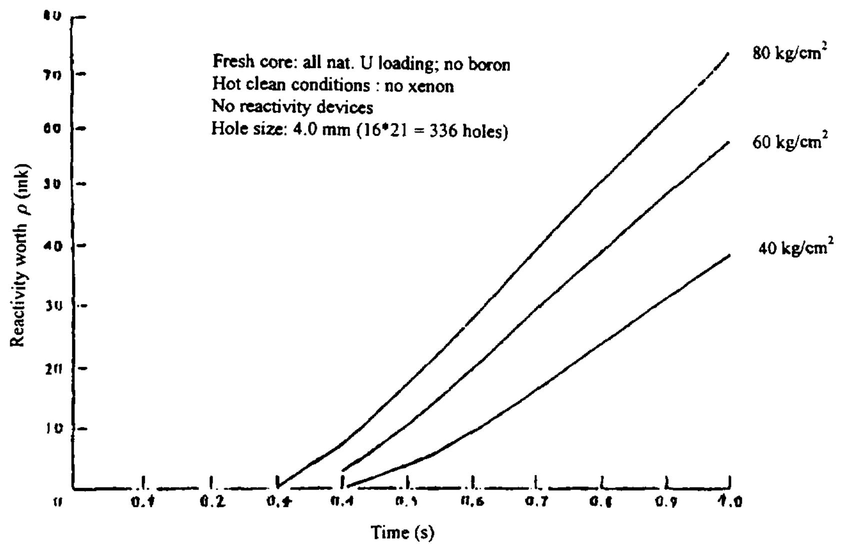
(i) Injection pressure = 40 Kg/Sq.cm
Hole sizes ; 3.2mm and 4.0 mm

(ii) Injection pressure = 60 Kg/Sq.cm
Hole sizes ;3.2 mm and 4.0 mm

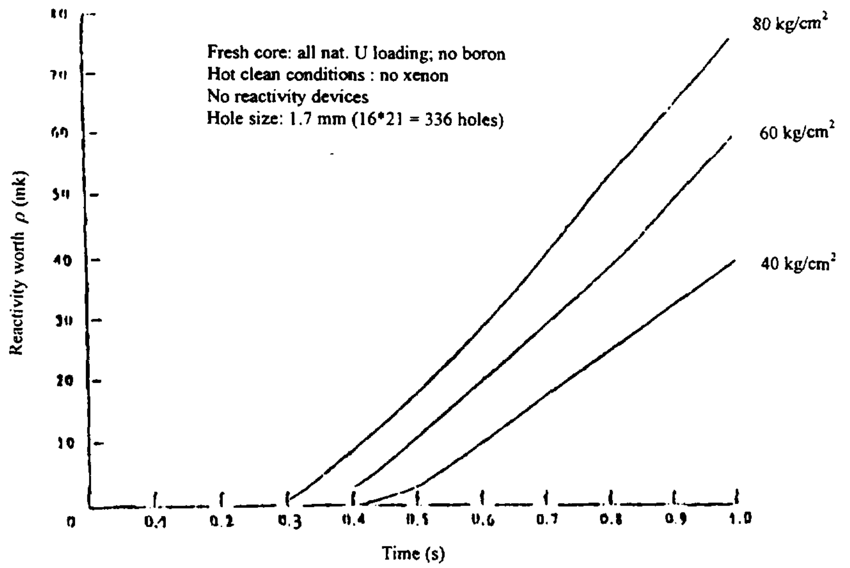
(iii) Injection pressure = 80 Kg/Sq.cm
Hole sizes ; 3.2 mm and 4.0 mm



Graph 5 Contribution to Σ_a (thermal) of the cell



Graph 6 Reactivity worth vs. time for 4.0 mm holes for SDS-2 of a 500 MWe reactor



Graph 7 Reactivity worth vs. time for 3.2 mm holes for SDS-2 of a 500 MWe reactor

II. Reactivity worth of SDS-2 in the presence of other reactivity devices and various core loadings. The jet growth considered was that of fully formed jets (i.e.at 1s) for an injection pressure of 80 Kg/Sq.cm.

(a) Fresh core ; all natural uranium loading ; No xenon.
  (i) No reactivity devices.
  (ii) Adjusters present.
  (iii) ZCU present.
  (iv) Both adjusters and ZCU present.

(b) Equilibrium core ; 8350 MWd/Te / 6450 MWd/Te.
  (i) No reactivity devices.
  (ii) Adjusters present.
  (iii) ZCU present.
  (iv) Both adjusters and ZCU present.

6. RESULTS

The experimental results of the Phase 1 experiments have been analysed. the reactivity worths evaluated is shown in graphs 6 and 7. The time t=0 corresponds to the moment of valve opening. The poison released from the tank would take some finite time to reach the first set of holes. Other delay times like sensing if SDS-1 has not operated and electrical circuit delays have not been accounted for.

The reactivity worth calculated for 4.0 mm hole size and 80 Kg/Sq.cm is 73.42 mk and for 3.2 mm hole is 75.42 mk. the worths in the presence of reactivity devices in both fresh and equilibrium fuel loading is tabulated in table 2.

Table 1 Spread angle for holes/slits

For single holes

S. No.	Hole diameter (mm)	Spread angle (degree)
1	2	15.8
2	3.2	16
3	4	17.9
4	5	16.1
5	8	16.43

For single slits

S. No.	Slit size (mm)		Spread angle (degree)	
	Length	Width	From length	From width
1	7.5	0.5	16.01	16.15
2	9	0.6	16.8	16.3
3	12	0.75	15.85	16
4	15	1	16.4	17.5

Table 2 Reactivity worth of SDS-2 for various loading

Injection pressure: 80 kg/cm^2
Time after the moment of valve opening: 1.0 sec

Reactivity device	Reactivity worth (mK)			
	Fresh core		Equilibrium core	
	3.2 mm	4.0 mm	3.2 mm	4.0 mm
No device	70.73	69.121	69.95	68.805
ARs IN	64.768	63.917	60.788	60.17
ZCU present	70.416	68.915	69.423	68.338
ARs + ZCU	61.45	60.852	57.165	56.77

7. CONCLUSIONS

It is fairly clear from the experiments that the secondary shut down system designed will be found reasonably adequate in providing fast shut down capability for the 500 MW(e) PHWR. The worths seem to depend on the injection pressure though not strongly on the hole size. Smaller holes give larger worths due to the fact that the poison travels for longer distances.

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ADVANCES IN SYSTEM DESIGN

(Session 4, Part 2)

Chairman

E. PRICE
Canada

COMPUTER BASED C&I SYSTEMS IN INDIAN PHWRs

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Abstract

Benefits of programmable digital technology have been well recognized and employment of computer based systems in Indian PHWRs has evolved in a phased manner, keeping in view the regulatory requirements for their use. In the initial phase some operator information functions and control of on-power fuel handling system were implemented and then some systems performing control and safety functions have been employed.

The availability of powerful microcomputer hardware at reasonable cost and indigenous capability in design and execution has encouraged wider use of digital technology in the nuclear power programme. To achieve the desired level of quality and reliability, the hardware modules for the implementation of these systems in the plants under construction, have been standardized and methodology for software verification and validation has been evolved. A large number of C&I functions including those for equipment diagnostics are being implemented. The paper describes the various applications of computers in Indian NPPs and their current status of implementation.

1. Introduction

The control and instrumentation in the earlier generation of PHWRs, up to MAPS, in India have been implemented using general purpose, modular single loop controllers, hardwired relay logics and conventional meters and recorders. The advantages of using computer based systems in control and operator information system applications is well recognized. Nuclear power plant control rooms are characterized by high levels of information content. Advanced operator information systems support operator cognitive functions in this environment by suitably optimizing the amount of information presented to the operator. Such systems can also include operator support functions which aid the operator in analyzing plant conditions in normal and abnormal situations.

To exploit the advantages of computer based systems, a phased approach has been pursued and it was decided to develop fairly complex but non-critical applications in the initial phase. Subsequently, in the recently commissioned power plants several control systems and operator information systems have been computerized. Microprocessor based system has also been incorporated in a safety system to provide advantages of self monitoring and resultant reduction in repair time, thus improving the availability.

In these systems, the emphasis was mainly on providing enhanced flexibility, improved man-machine interface and capability to incorporate sophisticated control algorithms. It is now being realized that microprocessor based systems can also offer possibilities of acquiring useful data for monitoring the condition of various plant subsystems to enable more effective maintenance. However, special efforts are needed to ensure that the software based systems are reliable and that they do not reduce the plant safety.

A brief description of various systems implemented in the existing PHWRs as well as those being developed for the future power plants is presented in the paper.

2. System Design Considerations

While implementing computer based systems for such applications, one of the main considerations is whether to adopt commercially available systems or to develop tailor designed systems. The main factors to be considered while evaluating commercially available systems are, availability of the state-of-the-art systems in India, support available for long term maintenance in NPPs, possibility of obtaining sufficient design details from the vendor to carry out failure mode analysis and the reviews by the licensing agencies and the overall system cost. Based on these considerations, tailor designed systems approach was found to be the better option for Indian NPPs.

As experience of usage of the tailor designed system in other similar applications may not be available, integration and testing of a prototype system using hardware modules from the same family as used for the final systems becomes essential. For ensuring high reliability of these systems in various possible operative conditions, full scale dynamic testing of prototype system using a real time simulator is considered desirable.

In a tailor designed system, the control system hardware has to be implemented using a family of microcomputer boards (modules). The selection of a suitable hardware module family has to be carried out such that high system reliability as well as long term availability of spares is ensured. The selection of a suitable microcomputer board family was carried out based on several considerations. These modules should be able to cater to the requirements of several safety related systems in the plant, should have on-line fault diagnostic features in every module and should use components which conform to MIL standards and are widely available. Derating policy for components should be adhered to. The agency producing these boards should ensure long term availability of spares.

To meet these requirements, a family of microcomputer modules was designed in-house and the designs were transferred to the Electronics Corporation of India (ECIL) for production. ECIL manufactures, supplies and installs C&I systems in Indian NPPs and provides maintenance services.

3. System Classification and Quality Assurance

3.1 System Classification

The quality and reliability considerations for computerized C&I systems are decided based on their safety functions. For the future NPPs, these systems are being classified as per IEC-1226 [1] into Safety systems (class-A), Safety Related Systems (class-B) and Information Systems (class-C). For the hardware and overall system aspects of these systems, well established practices are followed conforming to IEC-987 [2].

3.2 Software Quality Assurance

For development and review of software a Software Quality Assurance Plan, commensurate to the system's safety classification, is worked out based on the guidelines provided by international organizations [3]. Extreme care is taken during the software development cycle to minimize the chances of design and coding errors. Necessary documentation is generated during each phase of software development cycle to aid review by independent specialists.

The software is subjected to multistage review during design phase, static analysis of code, module level testing and integrated testing with the participation of software engineers, not involved in the original design. This enables corrective actions early in the development cycle and provides sufficient confidence during the phase of integrated system testing. For the safety and safety related systems, dynamic testing of the integrated system for various possible plant operating conditions is made possible through the plant simulation facility.

3.3 Software Analysis Tools

To implement effective V&V, supported by tools wherever possible, an on-going tools development program exists to develop automated tools for verification & validation of software. In the first phase of this program code analysis tools known as Static Analyzers have been developed and used for verifying code developed for safety and safety related systems. These tools verify the control flow, check compliance to structured programming rules, compute many software metrics and also impart visibility to the code through their graphical user interfaces and help in planning of unit tests.

One of the tools that has been developed and extensively used is Assembly Language Program Analysis Tool (ALPS). ALPS performs control flow analysis of assembly programs for Intel 8085, 8086 and Motorola 68000 processors in some assembly language variants [4]. The tool has capability to perform many checks as per IEC 880 standard [3] which has many recommendations on code design. The tool is supported on IBM compatible PC platform. This tool has been applied to analyze software of PDC system for KAPP, Reactor Regulating Systems of NAPS and KAMINI reactors. A reverse engineering tool called C Language Program Analysis System (CLAS) has been developed to handle code written in ANSI C language [5,6]. CLAS supports control flow analysis and entity-relationship analysis. It provides a powerful graphical user interface for analysis and subsequent browsing of results and report generation. The CLAS is supported on SPARC workstation.

Specifying the behavioral aspects of real-time systems and analyzing these before design is undertaken can greatly reduce development time and result into validated requirements. In order to achieve this a Programming Environment for Real-Time Systems (PERTS) is under development using the Statecharts formalism.

4. Computerized Safety Systems

4.1 Programmable Digital Comparator System

Programmable Digital Comparator System (PDCS) is a microcomputer based facility installed at Kakrapar Atomic Power Project (KAPP). It compares plant signals against set points to generate outputs which are used to perform the function of reactor trip, reactor setback, interlocks for process equipment and alarm annunciation. Earlier these functions were performed by about 360 Indicating Alarm Meters (IAM). The PDCS is configured (Fig. 1) into two physically and functionally independent units; the triplicated Alarm Unit (AU) and dual redundant Display Unit (DU) coupled by isolated serial communication links. The AUs perform the protective functions while DUs provide MMI functions.

Each AU handles 120 analog input signals which are filtered, checked for rationality and then compared against set-points. It periodically communicates status and signal values to DU. It implements set-point changes, signal enable/disable, set-point values and diagnostic messages when requested by DU. All hardware and software modules are periodically checked for their healthiness and a fault tolerant and fail safe approach is taken on the system outputs.

The DU receives periodic information sent by AU and provides the operator interface functions. The DU screen provides signal status information in 23 tables grouped by subsystems, displays all alarm messages and provides the record of all set-point changes and enable/disable operations performed under authorization.

An improved version of PDCS with enhanced facilities is being developed for PHWRs under construction.

4.2 Shut down System Performance Monitors

Monitoring the performance and availability of shut down devices is an important function from the point of reactor safety. A triplicated microprocessor based system namely Primary Shutdown System Failure Sensing Units (PSSFSU) for monitoring the scram performance of fourteen mechanical shut-off rods has been incorporated. Similarly, another microprocessor based system to monitor the performance of solenoids during surveillance checks of Secondary Shutdown System and also log the performance, namely, Bank in Times during scram, has been designed and put into operation in KAPS. The timing information from these systems help station operations to asses the health of the system and assist in maintenance.

5. Computerized Safety Related Systems

5.1 Fault Tolerant Systems for Process Control and Reactor Regulation

Two distributed fault tolerant computer control systems based on Dual Processor Hot-Standby (DPHS) configuration have been developed for NPPs under construction. The DPHS-RRS is designed for reactor power regulation and DPHS-PCS regulates five critical processes in the reactor heat removal systems, namely, primary coolant pressure, steam generator pressure, bleed condenser pressure and level and the fueling machine heavy water supply pressure.

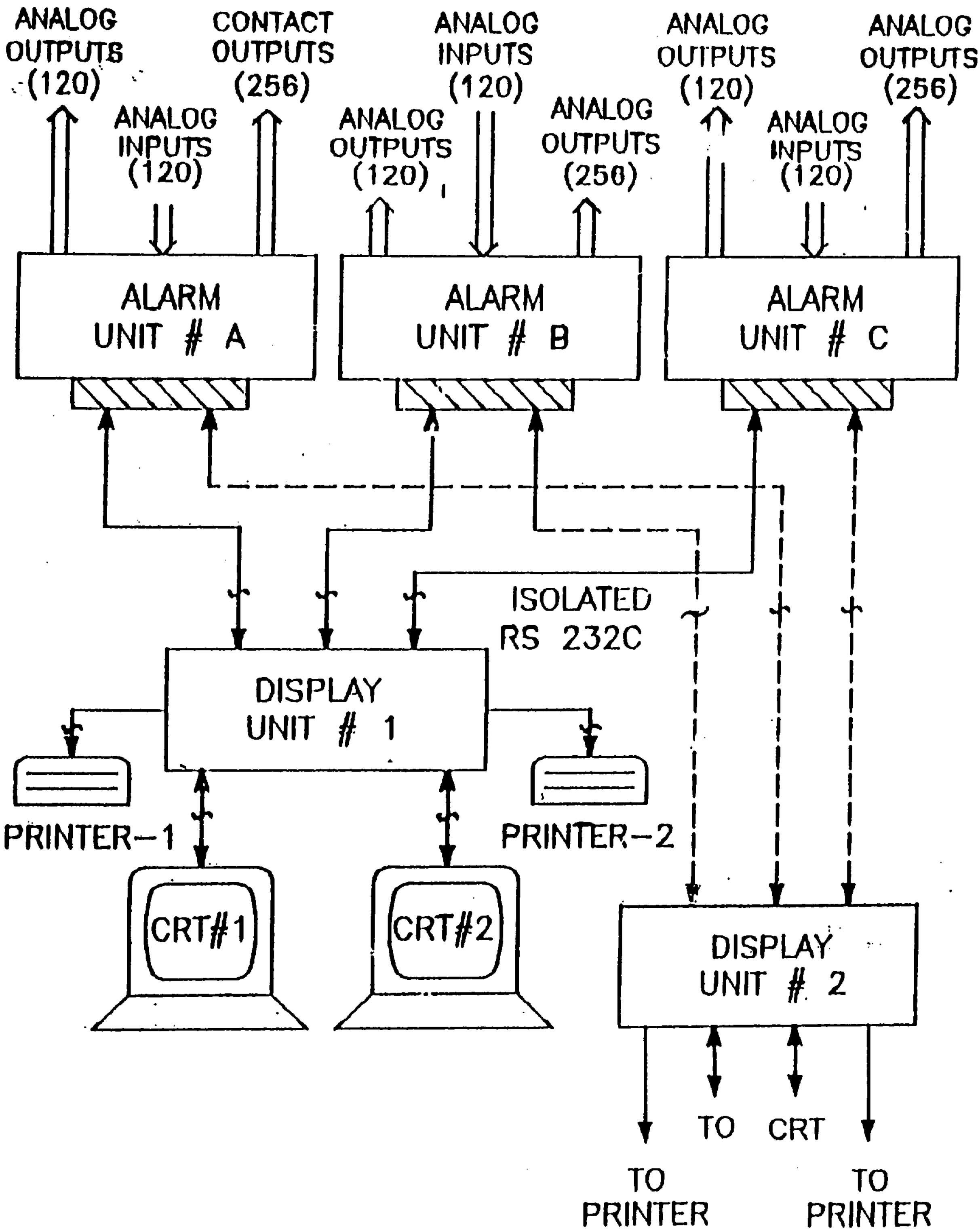

FIG. 1 : KAPP PDCS CONFIGURATION

Both the reactor regulating system (RRS) and the process control system (PCS) receive a large number of analog inputs from process measurement signals most of which are triplicated. In addition, a sizable number of contact inputs, from operator panels and other plant control systems are fed to each DPHS system. The control outputs are sent to the final actuators.

In DPHS configuration (Fig.-2) high reliability is achieved by having a `main' computer system (system-A) and an identical `stand-by' computer system (system-B), each one generating all needed control outputs. Further, both computer systems employ two identical processors, each with its own local memory and watchdog timer. These processors perform essentially the same processing jobs on same inputs, in parallel but with some time shift, enabling prompt detection of faults which manifest as mismatches between the two processor outputs. In addition, these processors perform exhaustive on-line diagnostic checks. As both the systems generate all the control outputs, a failure detected in one system leads to automatic switch-over of control output connections to the other healthy system through the action of a simple and rugged Control Transfer Unit, which is based on electromagnetic relays.

To reduce the impact of control signal interruption during a switchover after one computer system failure, the control signals for multiple actuators acting in parallel on the process (e.g. multiple control rods, multiple control valves on the same line etc.) are distributed among the two computer systems in a balanced manner. There is also a provision for the operator to transfer the entire control duty to any one system. This would enable him to perform off-line diagnostics and maintenance on the other system.

To minimize the complexity of software in the control computers, a display computer system has been interfaced to the control computers. The display computer, in communication with the control computers, drives panel mounted display devices. The control computers, thus freed from the display driving duty, have simpler software structure and provide better response for control functions. A PC based operator information system also forms a part of the control system. This screen based operator information system provides an improved man-machine interface giving the operator necessary information for process monitoring, system diagnostics and interactive facility for permissible on-line parametric changes.

All the computers in the distributed configuration are connected through a high speed Token Bus LAN which ensures that the timing specifications for updating information to the operator are met. The system also provides facility to periodically update the data in the plant-wide Computerized Operator Information System through special network connections.

5.2 Controls for On-Power Fuel handling System

The fuel handling system handles radioactive spent fuel bundles from the reactor and all operations are required to be done safely and remotely, as the equipment is inaccessible. For the reactors at Rajasthan (RAPS) and Madras (MAPS), the control system is hard wired. For subsequent reactors at Narora (NAPS) and Kakrapar (KAPP), on-power fuelling is being carried out using a computerized control system. The control system has been designed to provide maximum operation

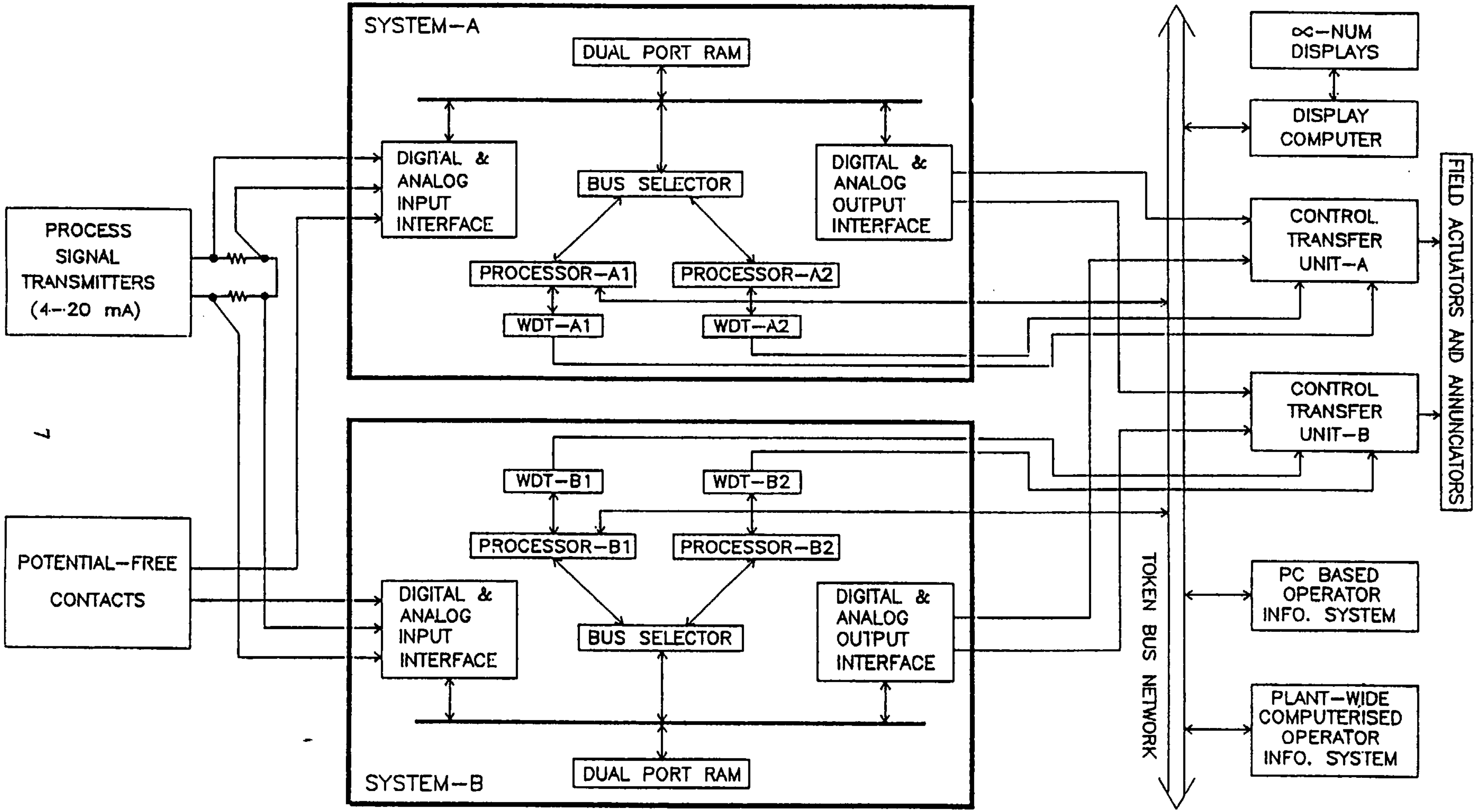

FIG. 2 : DUAL PROCESSOR HOT STANDBY CONTROL SYSTEM CONFIGURATION

flexibility. Three modes of operation Auto, Semi-auto, and Manual modes are provided. The control system is a dedicated network of one minicomputer as a master and two microcomputers as slaves in a distributed control system configuration. The salient features of the on line software are the cycle based executive and the customization of application software. Easy to use structured programming real time process control language (PCL) interface, tailored to application requirements, provides flexibility and maintainability with high reliability.

Based on the experience gained during development and operation of the control system at NAPS and KAPP, a control system with enhanced features is being designed for the four reactors at RAPP-3&4 and Kaiga-1&2. The control system under development enhances the existing configuration to include advanced operator information features to increase ease of operation and to provide storage and processing of operation logs on disks with automated extraction of performance data of various components and subsystem as an aid to maintenance. The software strategy adopted ensures transportability of software during hardware upgradation and follows modular approach such that the enhanced features can be incorporated as back-fits in the control systems of existing power plants. A PC based dynamic simulator is being developed for PHWR fuel handling system, for testing the prototype control system and also for validation of on-site software upgrades.

The software is tested using static simulator and dynamic simulator. A PC based dynamic simulator has been developed for PHWR fuel handling system, for testing the prototype control system and also for validation of on-site software upgrades. A PC screen based soft console has also been developed for carrying out all operations in manual mode.

5.3 Channel Temperature Monitoring Systems

The Channel Temperature Monitoring System measures coolant temperatures at the outlet of all 306 fuel channels using RTDs. The system calculates various temperatures, power output for each channel and also bulk reactor power output and displays and prints records of channel outlet and zone mean temperatures, channel power, etc., and generates alarms whenever any parameter exceeds a set value. For NAPS-2, two computer systems, each catering to one RTD installation, were built and commissioned in 1990. For KAPP, the two RTD installations are multiplexed onto one computer; the KAPP-1 system was commissioned in 1991 and the KAPP-2 system in 1993.

An advanced Channel Temperature Monitoring System for NPPs under construction is being developed such that in addition to the normal monitoring functions, the system software generates setback when it detects temperatures beyond limits at any channel outlet; it also generates outputs for flux tilt control.

The CTM system consists of two installations each of three microcomputer units. The Alarm Unit (AU) acquires all COT inputs, processes these values, and sends their status to the AU of the second installation; it also generates setback upon alarm coincidence to cause reactor power to be reduced. The logging unit (LU) consists of operator interface unit and control unit. The operator interface unit (OIU) collects all COT values from AU, and displays these and several other parameters in various formats including chromatic profile etc. The control unit acquires information on boiler differential temperature, sends it to OIU, and generates alarm outputs for flux tilt control.

5.4 PLC Based Interlocks

A programmable logic control system is used for process logics in RAPP-3&4 and Kaiga-1&2. The system replaces relay based non-triplicated logics used in the earlier plants. The programmable logic controllers (PLCs) are divided in two groups for implementing logics of the redundant equipments in the respective groups. Each PLC has 512 I/O capacity and is based on two processors in hot-standby configuration. The PLCs are interconnected with dual redundant LAN. The system has a control console for monitoring and configuring the logics.

6. Operator Information Systems

6.1 Disturbance Recording System

A Disturbance Recording system has been developed to perform three major functions. It acquires and records 64 parameters every one second for five days for the purpose of replacing strip chart recorders, and, every 200 msec. for five minutes (-2, +3) for the purpose of recording any disturbance and 8 parameters every 5 msec. for 5 sec. for replacing a Visicorder. The system enables an operator to view past and current information in a variety of formats and also obtain hardcopy. The system also provides the data in a machine readable form for analysis at another site. Several general utility software modules including a real time operating system, man-machine interaction package etc. were developed.

The system consists of an Acquisition Unit and a Recording Unit. The Acquisition Unit based on a microprocessor, scans and processes all signals and sends the data to the Recording Unit through a high speed synchronous link for storing in a non volatile memory and providing to the operator on demand. The Recording unit comprises two PCs for redundancy. Six such systems have been fabricated; three systems were commissioned at KAPP-1 and MAPS-1 & 2 in 1992, one at KAPP-2 in 1993 and two at NAPP-1 & 2 in 1994. An advanced version of this system is now being developed for NPPs under construction to incorporate event sequence recording of 448 parameters with a resolution of 1 msec.

6.2 Computer based Operator Information System

The computer based operator information system (COIS) monitors overall plant status. The system is primarily designed as information system, with a limited number of control features. It has resulted in reduction in number of recorders, meters and annunciation windows. For important safety related parameters window-annunciation, dedicated indicators and recorders are retained, with the COIS offering a back-up.

The COIS system consists of main and standby computers, display stations and input-output subsystems as nodes, connected by high speed Local Area Networks (Fig. 4). The network topology is fault tolerant, permitting a single break anywhere in the network. This is achieved by having separate LAN for input output subsystems and the

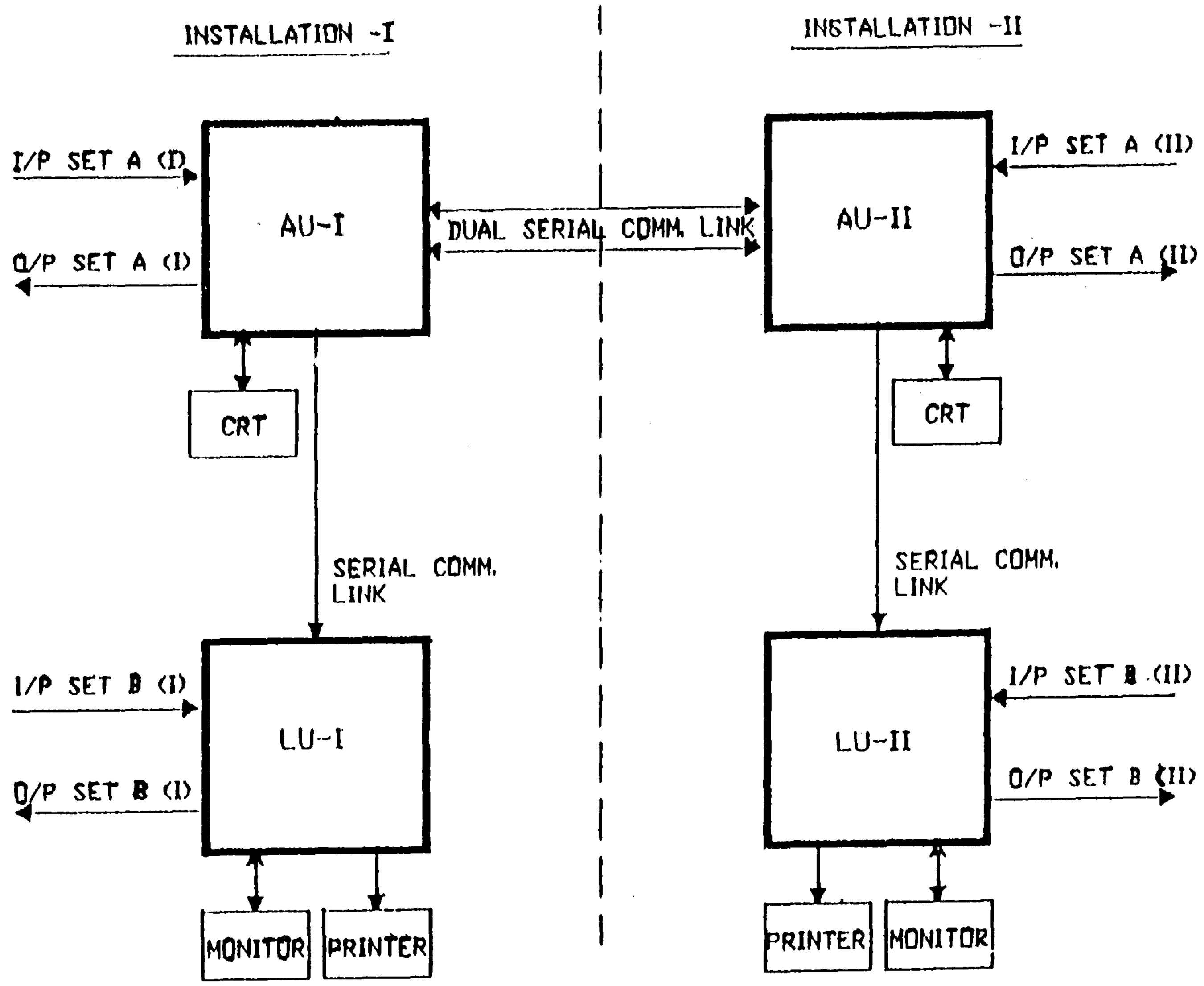

FIG. 3 : CHANNEL TEMPERATURE MONITORING SYSTEM

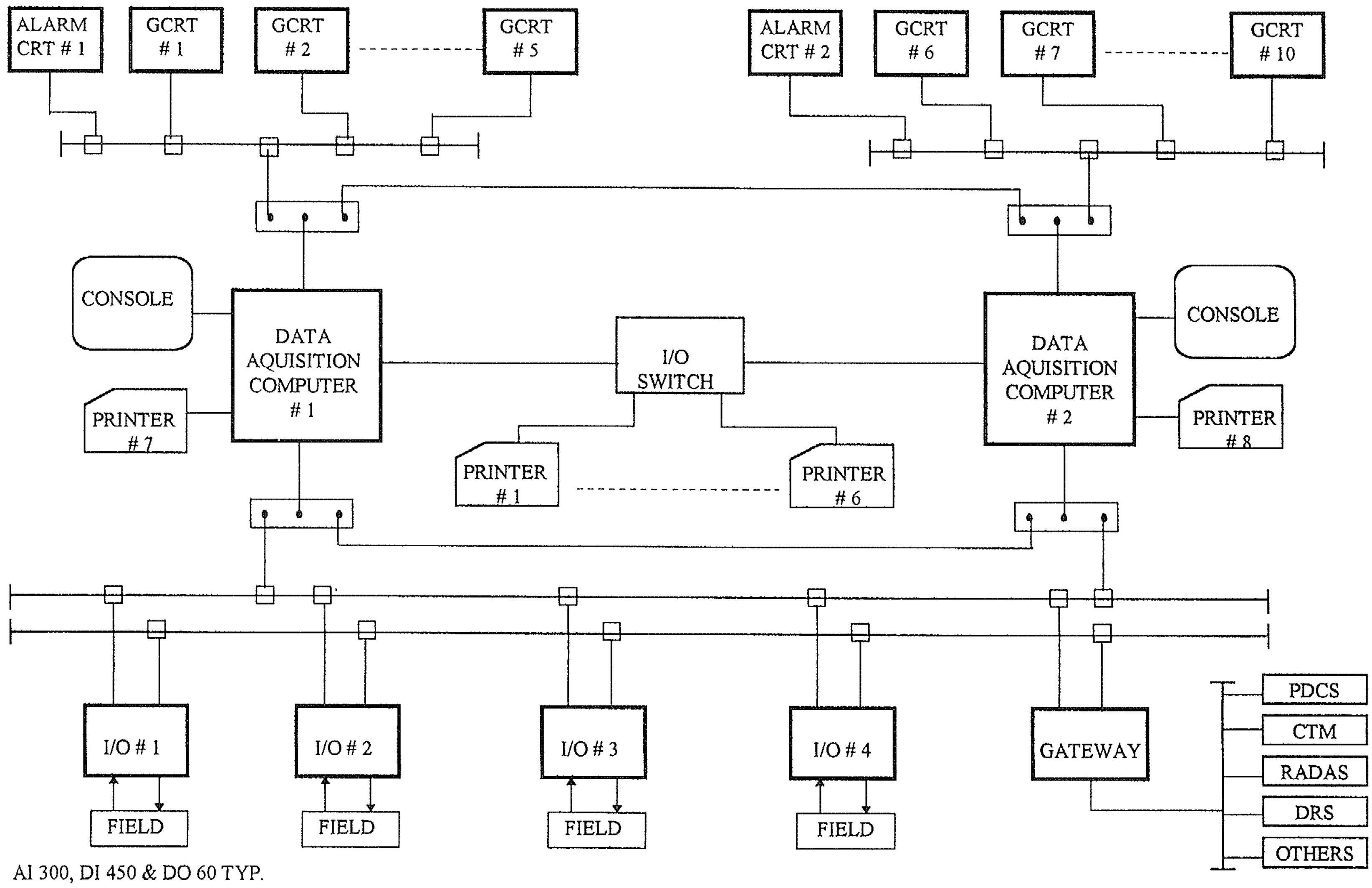

FIG. 4 - CONFIGURATION OF COIS

display terminals. The functions are taken over by the standby computer in the event of failure of the main computer.

There are about 1200 analog inputs for alarm generation and about 1800 digital inputs for alarm and status reporting. Visual display units (CRTs)provide formatted tabular displays, bar-graphs, trends, mimics, etc. Alarms are displayed on two dedicated alarm CRTs located on main control panel and operator console. A dedicated alarm printer gives automatic log of alarms.

For the plants under construction the COIS will also be interfaced to the computer systems like EDAS, RADAS, PDCS, DPHS, PLC, etc., to receive data from these systems for presentation to operator through sophisticated MMI features of COIS, and for on line performance analysis.

Besides performing operator information functions, COIS also provides archival of surveillance carried out on various systems important to safety like emergency core cooling system (ECCS). On demand, COIS also scans delayed neutrons in the coolant samples from each coolant channel and reports channels suspected to contain failed fuels.

6.3 Supervisory Control and Data Acquisition System

In KAPP, a centralized electrical data acquisition system (EDAS) is used for monitoring of all electrical data such as voltage, current, power, frequency, circuit breaker status, etc. A supervisory control and data acquisition system (SCADA) having distributed architecture is under implementation for RAPP-3&4 and Kaiga-1&2. Since the signal inputs are spread over geographically distributed areas like switchyard, turbine building and control equipment room, the system is built around intelligent Remote Terminal Units (RTUs) at each location. The RTUs are connected to the centralized operator information system on a dual redundant data highway. Separate operator station is proposed in Switchyard Control Room. For controls, each node can process data and issue control command to open or close breakers, etc. The system accepts approximately 225 analog and 2500 digital inputs.

6.4 Radiation Data Acquisition System (RADAS)

The radiation data acquisition system performs radiation monitoring at the reactor site. The system has three microprocessor-controlled interface units, one covering inputs from unit-1, second for inputs from common areas of unit-1 and 2 and the third covering inputs from unit-2. The interface units are connected to the rack mounted PC based host system through serial communication links. Area radiation monitors, ventilation duct monitors, environmental monitors, accident monitors in reactor building, stack activity and heavy water loss monitoring system, process water discharge activity monitors, tritium in air monitoring system are all covered by this system. It has provision of four color monitors for display and a printer for hard copy. One monitor is located in health physicist's office for quick assessment of plant radioactivity levels and issue of work permits, etc. The system also gives contact outputs for actuation of local flasher and hooter warning in areas having high radiation fields.

6.5 Integrated Distributed Control and Process Information System for 500 MWe

In 500 MWe PHWR, an integrated approach is being adopted to use microcomputer based system. Distributed control and process information system (DCPIS) is planned for operational C & I. DCPIS is divided in two levels, distributed control system (DCS) level for process control and process information system (PIS) level for operator interface (Fig. 5). DCS nodes are functionally distributed like primary heat transport, moderator system, etc. Control tiles having push buttons are planned so that control from either key-board or control tiles can be done.

Standalone systems are designed for RRS, CTM, RADAS, PDCS, SCADA, etc., which can transfer data to PIS for centralized information.

6.6 Emergency Operating Procedures System

A prototype Emergency Operating Procedures System (EOPS) system based on PCs has been developed to provide procedural support to the operator on a computer screen in case of an emergency. Without such a system, the operator has to monitor a large number of process parameters on CRTs and on panel instruments spread all over the control room. He is also expected to follow emergency operating procedures prescribed for various anticipated emergency situations. A computer based EOPS system offers the advantages of monitoring all instruments quasi-instantaneously and comparing measurements with fixed or dynamic criterion and also responds instantaneously to the changes in situation through parallel monitoring. The system is designed such that it can be programmed with the specific procedures at each site.

6.7 Vibration monitoring system for Turbine-generator

The turbine-generator unit is subjected to complex dynamic forces, some of these acting, at times, simultaneously. Such dynamic effects get reflected in the vibration signature of the rotor and/or supporting structures and proper recording of these and their analysis can give a clear picture of the health of the machine. Using these data and their trends, it is even possible to predict an impending trouble in the machine so that preventive action can be taken in time and catastrophic failures can be avoided. A continuous condition monitoring and analysis can give quick warning and enable operator to take preventive measures. A Fast Data Acquisition System capable of monitoring 30 channels at 5 K samples/sec. per channel has been developed and planned for installation at KAPS.

6.8 On-Line Fatigue Monitoring System

An on-line Fatigue Monitoring System is being developed to assess the structural degradation of components. This assists in planning in-service inspection and maintenance and may also support future life extension program of a plant. The system acquires the relevant plant process parameters for all the components every 5 seconds and stores it on hard disk. Once in every 24 hours, OLFM system software on the PC executes fatigue computational module based on finite element techniques for each of the configured parameters. The module generates fatigue related information, and updates the fatigue data base.

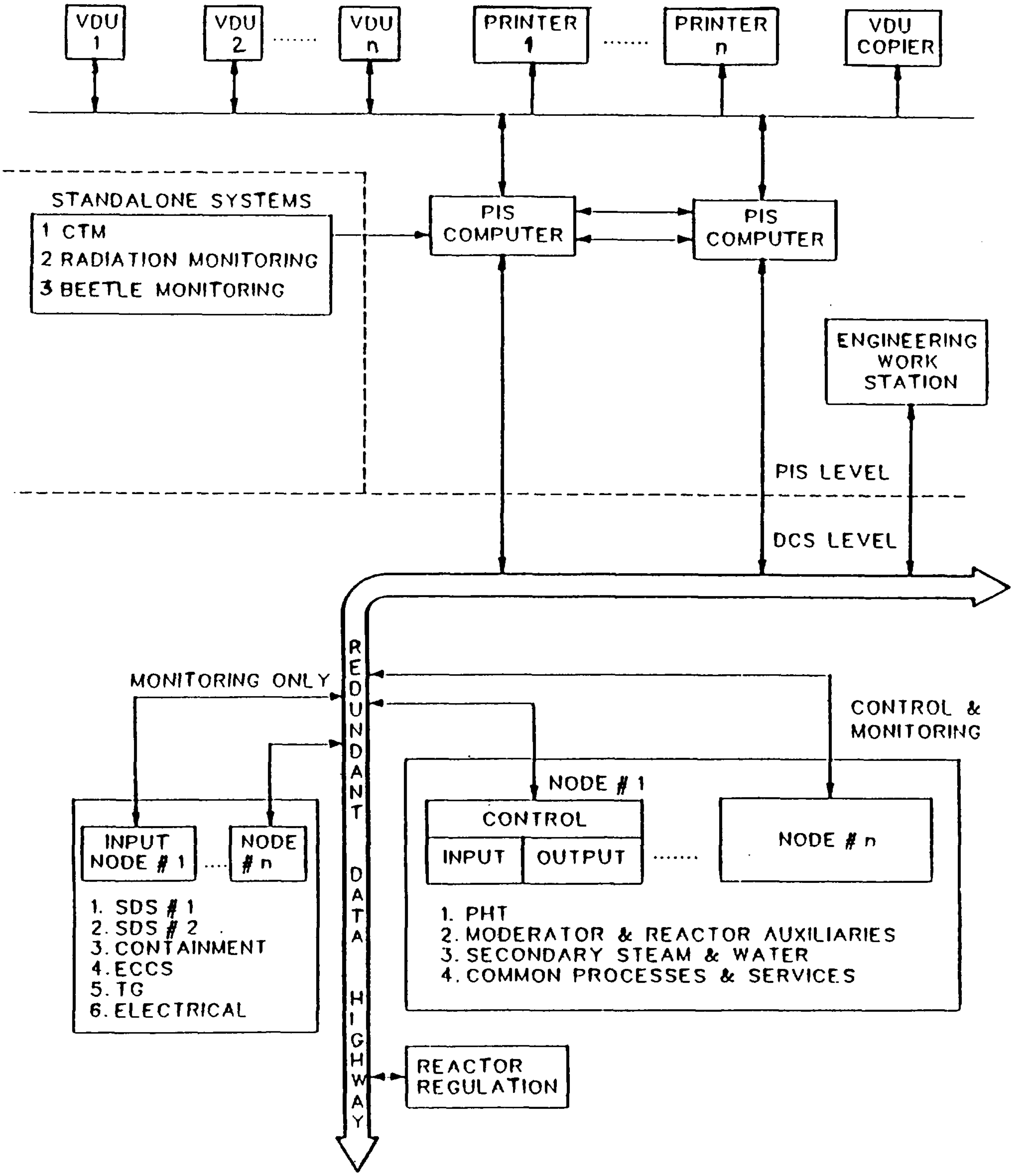

FIG. 5 : ARCHITECTURE OF COMPUTER - BASED C&I FOR 500 MWe PHWR

The OLFM system facilitates user friendly MMI to enable the user to view the component details i.e. geometry , finite element mesh, material properties etc. The system also provides display of fatigue history curves and contours for, temperature, pressure, fatigue usage factor, in elegant formats. The system also provides facility to transfer the data on to a removable media for analysis. In addition the system also provides most of the displays in DRS like trend, tables, bar graphs and history.

6.9 Diagnostics of motor operated valves

Degradation and failure of motor-operated valves (MOV) compromise operational readiness of the safety related systems of a nuclear power plant (NPP). Motor current signature analysis (MCSA) has been found to be a selective and early indicator of any mechanical or electrical fault related to the valve actuator motor (VAM) and the valve. Any knowledge of such faults enables proper preventive maintenance. MCSA allows remote, unobtrusive sensing which is necessary as the valves could be at inaccessible locations in the plant.

Development work on automatic extraction of the motor current signature and its classification into fault classes using a statistical pattern recognition methods is being carried out. Discriminant analysis is being used to classify the feature vectors obtained from current signatures by fitting splines of suitable order into different fault classes. The development of a software package for heath monitoring and fault diagnosis of MOVs has also advanced considerably and work towards installing a diagnostic system in one of the NPPs is in progress.

7. Summary

Use of computers in C & I systems has been of great help. It facilitated in providing instant values of related data like CTM readings, triplicated readings and multipoint trends and records. It has also helped in post analysis of the initiating events and identification of causes of failures. This has resulted in improved analysis and system performance. Introduction of computerized systems for process control, equipment diagnostics, etc., will improve overall availability.

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THE HIGHER ORDER FLUX MAPPING METHOD IN LARGE SIZE PHWRs

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Abstract

A new higher order method is proposed for obtaining flux map using single set of expansion mode. In this procedure, one can make use of the difference between predicted value of detector reading and their actual values for determining the strength of local fluxes around detector site. The local fluxes are arising due to constant perturbation changes (both extrinsic and intrinsic) taking place in the rector.

1. INTRODUCTION.

Large reactor systems, such as Indian 500 MW(e) PHWRs, are known to sustain power tilts following minor local perturbations. If these tilts are not controlled, they can lead to unacceptable power distribution. This can result in an economic penalty. Therefore it is essential to have in-core flux measurements, and to process these measurements to get accurate knowledge of the operating state of the reactor. With the help of the On-line Flux Mapping System (OFMS), a virtually continuous power regulating system, then, can maintain the core power distribution closer to the design values [1].

The Self powered Neutron Detectors (SPNDs) are employed to measure the point thermal fluxes in the reactors. The electrical signals generated by these detectors, on absorption of the neutron and followed by β^- decay, are fed as an input to the flux mapping procedure. In flux mapping, the instantaneous flux distribution is expressed in terms of finite number of λ modes (corresponding to the nominal operating state of the reactor), while the modal amplitudes are determined in such a way that, the least square error between the actual detector readings and synthesized flux values at detector sites is minimum.

It is apparent that there would be inherent errors present in the flux map, generated by OFMS due to following reasons : (1) Flux synthesis errors, as limited number of flux modes are used in obtaining the flux map. (2) The numerical truncation + round off errors in calculating the higher harmonics of the diffusion equation itself.(3) Random errors accompanying the online flux measurements, and their propagation to the estimated flux map.

In order to reduce the synthesis errors in the predicted flux map during the reactor operation under the perturbed conditions, the basis set (i.e. set of predetermined λ modes) is expanded by adding appropriate perturbation modes such as adjuster rod shim modes, control absorber modes, start up modes. A perturbed mode basis is said to have formed, when perturbation modes are added to standard mode basis [2].

Since each mode basis is stored on the online computer and depending upon the operating state of the reactor proper basis set is to be selected for getting the flux map, this procedure of flux mapping needs the operator intervention for selection of the particular basis set. This will definitely increase the administrative efforts. Furthermore in this procedure, only limited number of device perturbations modes corresponding to the predetermined patterns of devices are available for flux mapping. It is possible that this perturbed mode set might be grossly insufficient for determining the flux map in some prevailing operating conditions of the reactor. Therefore the best alternative is to carry out the flux mapping operations with the help of a single basis set. It is apparent that some sort of auxiliary procedure will be needed to improve the accuracy of the flux map for those core configurations which are having strong perturbations.

In this paper we are going to discuss the higher order method for obtaining the flux map using the single basic set for expansion with newly developed auxiliary method. We are also going to present analysis of this procedure.

2. LEAST SQUARE PROCEDURE FOR FLUX MAPPING

The basic assumption of the flux mapping software is that the instantaneous thermal flux inside the reactor can be represented by the linear combination of a set of predetermined modes.

$$\Phi(\bar{r}) = \sum_{j=1}^{S} A_j \Psi_j(\bar{r}) \tag{1}$$

where,

$\Phi(\bar{r})$ is the instantaneous thermal flux, and, $\Psi_j(\bar{r})$ higher modes / flux distributions which are normally encountered in an operating reactor.

The aim of the flux mapping procedure is to obtain the combining coefficients A_j (and therefore the flux profile) from the measured flux levels. Since the equation (14) is valid at all points inside the reactor, it is also valid at the detector sites, we can rewrite it in matrix form at the detector sites as,

$$\left[D_d \right]_{Rx1} = \left[M_{dj} \right]_{RxS} \bullet \left[A_j \right]_{Sx1} \tag{2}$$

The flux equation at the selected points can also be written in a matrix form as follows,

$$\left[F_n \right]_{Kx1} = \left[N_{nj} \right]_{KxS} \bullet \left[A_j \right]_{Sx1} \tag{3}$$

where,

- R Number of flux mapping detectors.
- S Number of modes / flux profiles chosen for the expansion.
- K Number of points at which flux map is required.
- D_d detector reading of the d^{th} flux mapping detector.
- M_{dj} j^{th} mode / flux distribution value at d^{th} flux mapping detector site.
- A_j combining coefficient for j^{th} mode.
- F_n flux map estimated at n^{th} flux mapping point.
- N_{nj} j^{th} mode / flux distribution value at n^{th} flux mapping point.

Since in general the number of detectors are more than the number of modes chosen for the instantaneous flux expansion,the equation (15) is a set of inconsistent equations Therefore to solve these equations a Least Square (LS) approximation is employed. Defining

$$\left[P_{jd} \right]_{SxR} = \left[M_{dj} \right]^{-1}_{RxS}$$

$$\left[P_{jd} \right]_{SxR} = \left[\left[M_{jd} \right]^{T}_{SxR} * \left[M_{dj} \right]_{RxS} \right]^{-1} * \left[M_{jd} \right]^{T}_{R} \tag{4}$$

$\left[P_{jd} \right]_{SxR}$ is called as generalized inverse of $\left[M_{dj} \right]_{RxS}$

Using equation (17), the equations for $\left[A_j \right]_{Sx1}$ and $\left[F_n \right]_{Kx1}$, become

$$\left[A_J \right]_{Sx1} = \left[P_{Jd} \right]_{SxR} \bullet \left[D_d \right]_{Rx1} \tag{5a}$$

$$\left[F_n \right]_{Kx1} = \left[N_{nJ} \right]_{KxS} \bullet \left[A_J \right]_{Sx1} \tag{5b}$$

Equations (5a) and (5b) are the working equations for flux mapping procedure.

3. HIGHER ORDER METHOD.

A novel refinement of the modal expansion method of flux mapping involves use of local flux correction functions. The total flux at any flux mapping site is assumed to separable into global component and a local component. The global component of the perturbed flux describes the propagation of perturbed flux due to neutron multiplication effects in the core, while the local flux describes the propagation of the perturbed flux due to the cross-sectional changes in the core. The local effect thus depends only on the cross-section at the site of perturbation and is attenuated by the absorption outside the perturbation region without propagation due to the neutron multiplication. Therefore the local flux effect in not present in the modes generated, since these modes have been generated for the unperturbed reference core. The local flux effect are modeled from an approximation to the solution of the local flux changes in source / sink model [3]. They appear as incremental flux change at the site of perturbation through the local multiplication factor. This increment flux extends beyond the region of perturbation to the surrounding region. It is assumed that any perturbation in the core is simulated by affecting the proper changes in the absorption cross-section of the thermal group in the perturbation zone. This will lead to the changes in flux level at the perturbation site. The diffusion equation for the perturbation site is

$$\begin{bmatrix} \nu\Sigma_{f1} & \nu\Sigma_{f2} \\ o & o \end{bmatrix} \begin{bmatrix} \phi_1 + \Delta\Phi_1 \\ \Phi_2 + \Delta\Phi_2 \end{bmatrix} - \begin{bmatrix} -D_1\nabla^2+\Sigma_{r1} & o \\ -\Sigma_{12} & -D_2\nabla^2+(\Sigma_{a2}+\Delta\Sigma) \end{bmatrix} \begin{bmatrix} \phi_1 + \Delta\Phi_1 \\ \Phi_2 + \Delta\Phi_2 \end{bmatrix}$$

$$= \begin{bmatrix} \nu\Sigma_{f1} & \nu\Sigma_{f2} \\ o & o \end{bmatrix} \begin{bmatrix} \Delta\Phi_1 \\ \Delta\Phi_2 \end{bmatrix} \tag{6}$$

After certain rearrangement of terms, we get

$$\begin{bmatrix} \Sigma_{r1} & o \\ -\Sigma_{12} & \Sigma_{a2} + \Delta\Sigma \end{bmatrix} \begin{bmatrix} \Delta\Phi_1 \\ \Delta\Phi_2 \end{bmatrix} = - \begin{bmatrix} o & o \\ o & \Delta\Sigma \end{bmatrix} \begin{bmatrix} \Phi_1 \\ \Phi_2 \end{bmatrix} \tag{7}$$

which gives the incremental fast flux $\Delta\Phi_1 = o$, and incremental thermal flux as,

$$\Delta\Phi_2 = \frac{\Delta\Sigma}{\Sigma_a + \Delta\Sigma} \Phi_2 \tag{8}$$

The attenuation of this incremental flux to other region is determined by the source calculation.

Thus the local flux effect functions are used for accounting the local flux changes induced due to the material configuration changes around the detector locations. As seen above the local flux effects are dependent only on the cross-sectional changes and are attenuated by absorption without propagation through the neutron multiplication. These changes are obtained from the flux distribution prevalent in the reactor due to fictitious neutron source kept around the detector [4]. The local flux change at the detector site is $\Delta\Phi_2(r_d)$ is dependent on the strength of the fictitious source around the detector site and the strength of the source is determined from the error between the actual detector reading and the predicted flux map at the detector site. It is further assumed that the local flux effect from all detector readings can be superimposed. Therefore this method of flux mapping would be as follows :-

The preliminary flux map is obtained using only the standard mode set (set of λ modes) for any operating condition of the reactor. The errors between actual and predicted values of detector readings are determined. These errors are used to estimate the local flux effects and further the corrections in the flux map.

4. GENERATION OF MODE DATA.

Higher modes are generated by the code MONICA [5] for their use in flux mapping procedure. These modes are corresponding to the nominal operating state of the reactors which means :

1) Reactor is in equilibrium burnup state.
2) No poison in moderator.
3) Adjuster rods fully in.
4) Zone controller are filled on an average 42 % .
5) All shut off rods are out.

In Table 1 we are giving the modes and their eigenvalue as obtained by code MONICA. We found that these modes are adequate for the purpose of flux map simulations.

5. GENERATION OF DETECTOR READINGS.

The most important input component of the flux mapping procedure is the FM detector readings. Since the online detector readings would become available only from an operating reactor, at present these readings have been estimated numerically from the power profile obtained during the simulation of our fuel management code TRIVENI [6]. The procedure adopted for obtaining the detector readings is discussed below in brief.

TRIVENI simulation is done for the desired Reactivity Device (RD) and burnup configuration for which the flux map is sought. TRIVENI calculates the average fluxes in uniform parallelepiped meshes. The detector tubes are inserted vertically. Since their distance from axial midplane is not normally a integral multiple of the fuel bundle length, the detector readings are derived from TRIVENI fluxes in two steps. Mean fluxes in three consecutive axial points adjoining the given detector location is found first as the average of the surrounding four radial meshes. Then the flux at the detector location is obtained by a parabolic interpolation [7].

6. RESULTS & DISCUSSION.

The detector readings were obtained numerically for the following core configurations, without superimposing the random errors. :

(A) Fresh fuel core configuration with burnup distribution generated from standard Pattern Age approximation and the RD positions are as follows,

1) No poison in moderator.
2) Adjuster rods fully in.
3) Zone controller are filled on an average 42 % .
4) All shut off rods are out.

(B) The core configuration as in case (A) plus Bank-1 of adjuster rods is withdrawn.

(C) The core configuration as in case (A) plus Bank-1 and Bank-2 of adjuster rods are withdrawn.

(D) The core configuration as in case (A) plus Bank-1 Bank-2 and Bank-3 of adjuster rods are withdrawn.

(E) The core configuration as in case (A) plus all banks adjuster rods are withdrawn.

The flux map is obtained for all above configurations using normal as well as higher order flux mapping procedure.

For assessing the predictions of the flux map, 500 flux points distributed uniformly throughout the core were chosen. Table 2 gives the comparisons of predicted flux map and TRIVENI fluxes in terms of maximum and rms errors for 102 detector sites as well as the 500 selected flux sites. It was found that the RMS errors in all 500 points was found to be 2.6 % for normal procedure and 2.3 % for higher order method. It is clear that since the detector readings are exactly matched in higher order procedure, there would be no spread in their predicted value of the flux map. The Maximum error value remains same indicates that the local flux effect dies down from all detectors before reaching at that point.

Table 3 give comparisons of predicted flux map with moderately strong perturbations such withdrawal of number of adjuster banks. As in case of nominal equilibrium case the maximum errors (both under predicted and over predicted) remain unchanged due the the fact that the local flux does not affect their prediction in the present model.

Table 4 gives the flux map comparisons of core follow up studies from fresh core up to 140 Full Power Days (FPDs) using normal and higher order flux mapping method. It can be seen the flux map prediction are better in case of higher order method.

7. CONCLUSIONS.

From present analysis, it can be concluded that the fundamental + 18 higher λ modes are sufficient for obtaining the flux map in case of small perturbations. In case of larger perturbations like movement of adjuster banks, the predictability of the flux map can be improved by making use of higher order flux mapping method. It can be noted that the basis set is not expanded by adding the perturbation modes correspond to the movement of adjuster rod banks. It can be possible to refine this technique to predict the flux map within the tolerated accuracy with this expansion mode set.

Table 1

Modes used for expansion of flux & precursor concentration

NO	NAME OF THE MODE	EIGENVALUE
1	FUNDAMENTAL MODE	1.00537
2	FIRST AZIMUTHAL (1)	0.99036
3	FIRST AZIMUTHAL (2)	0.98934
4	SECOND AZIMUTHAL (1)	0.96671
5	SECOND AZIMUTHAL (2)	0.96566
6	THIRD AZIMUTHAL (1)	0.93782
7	THIRD AZIMUTHAL (2)	0.93334
8	SECOND AXIAL	0.95154
9	FIRST RADIAL	0.93831
10	SECOND AXIAL * FIRST AZIMUTHAL (1)	0.92742
11	SECOND AXIAL * FIRST AZIMUTHAL (2)	0.92573
12	FIRST AXIAL	0.98364
13	FIRST AXIAL * FIRST AZIMUTHAL (1)	0.96705
14	FIRST AXIAL * FIRST AZIMUTHAL (2)	0.96586
15	FIRST AXIAL * SECOND AZIMUTHAL (1)	0.94222
16	FIRST AXIAL * SECOND AZIMUTHAL (2)	0.94154
17	FIRST AXIAL * THIRD AZIMUTHAL (1)	0.91377
18	FIRST AXIAL * THIRD AZIMUTHAL (2)	0.90860
19	FIRST AXIAL * FIRST RADIAL	0.92914

Table 2

Flux shapes comparisons between TRIVENI fluxes and predicted flux map for the equilibrium case.

	102 Detector sites		500 flux map sites	
	Normal	Higher	Normal	Higher
1). % rms error.	1.8	---	2.6	2.3
2). % maximum under prediction error.	-6.9	---	-10.4	-10.4
3). % maximum over prediction error.	2.5	---	2.5	2.5
4). Number of points having errors between ± 2 %	80	102	334	372

$$\% \text{ error} = \frac{(\text{Predicted}) - (\text{TRIVENI})}{(\text{TRIVENI})} \times 100$$

Table 3

Flux shapes comparisons between TRIVENI fluxes and predicted flux map

	Case B	Case C	Case D	Case E
A) 102 Detector sites				
% rms error	1.7	3.9	3.6	5.9
% max error (under prediction)	-7.2	-14.9	-17.2	-23.2
% max error (over prediction)	3.1	6.3	6.1	14.2
number of points (closely predicted)	89	71	45	44
B) 500 Flux map sites				
% rms error	2.7	4.0	4.8	7.8
% max error (under prediction)	-8.8	-15.4	-14.2	-22.7
% max error (over prediction)	8.1	8.6	9.4	17.2
number of points (closely predicted)	330	252	188	114
C) 500 Flux map sites				
% rms error	2.4	3.8	4.5	7.2
% max error (under prediction)	-8.8	-15.4	-14.2	-22.7
% max error (over prediction)	8.1	8.6	9.3	16.8
number of points (closely predicted)	355	295	226	152

A Normal procedure.

B Normal procedure

C Higher order procedure.

Table 4

Flux Map Comparisons during Core follow up

FPDs	Flux Map Comparisons Normal procedure			Flux Map Companions Higher Order procedure		
	% rms err	% max err	pts	% rms err	% max err	pts
0	2.58	10.4	334	2.36	10.4	371
10	2.49	9.4	333	2.27	9.4	375
20	2.68	10.5	323	2.45	10.5	364
30	2.72	10.8	338	2.47	10.8	364
40	2.44	9.5	350	2.21	9.5	384
50	2.22	8.3	368	2.01	8.3	397
60	1.99	7.0	380	1.80	7.0	406
70	1.67	5.3	413	1.51	5.3	428
80	1.42	5.2	428	1.29	5.3	443
90	1.29	5.1	449	1.18	5.1	456
100	1.33	5.0	440	1.23	5.0	453
110	1.59	4.9	401	1.43	4.9	425
120	1.93	5.2	347	1.71	5.2	386
130	2.32	6.0	301	2.04	6.0	350
140	2.74	7.6	253	2.39	7.6	317

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CANDU DESIGN OPTIONS WITH DETRITIATION

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Abstract

CANDU® reactors include a number of auxiliary systems to manage the inventory, purification, clean-up and isotopic purity of the heavy water used in the moderator and heat transport system. These systems are designed and installed to treat the moderator and heat transport water in separate parallel systems. One of the reasons for this parallel approach to heavy water management is the tritium inventory in the heavy water. Different levels of tritium accumulate in the moderator and heat transport system during reactor operation, with the moderator water having a much higher tritium concentration. Strict separation of the high-tritium-concentration moderator water from the low-tritium-concentration heat transport system water is an integral component of the CANDU design and operating strategy to limit potential releases of tritium to the containment building atmosphere.

AECL is developing a new cost-effective technology for the detritiation of heavy water based on the Combined Electrolysis and Catalytic Exchange (CECE) process. This detritiation technology has the potential to be integrated into the heavy water management systems of a CANDU reactor. On-line detritiation could be used to limit the concentration of tritium in the moderator and also to detritiate any water collected within the containment building from other sources.

The availability of economic detritiation technology would provide a flexibility to redesign some of the auxiliary heavy water management systems. In particular, there is potential to eliminate some of the duplication in the current management systems and also reduce costs by reclassifying some reactor systems that would have lower maximum tritium concentrations. This paper discusses some of the advantages of detritiation and some of the conceptual design options that detritiation would provide. The goal would be to lower the overall reactor cost with detritiation, but it is premature to assess whether this goal can be achieved.

1. INTRODUCTION

CANDU® reactors include a number of auxiliary systems to manage the inventory, purification, and clean-up of the heavy water used in the moderator and heat transport systems. In the current generation of CANDU reactor designs, the auxiliary systems are constructed in parallel to treat the moderator and heat transport water separately. There are two reasons for this parallel approach to heavy water management. The first reason is the different chemistry control requirements of the heavy water in the two systems. The second reason is different tritium levels in the heavy water of the two systems. Strict separation of the high-tritium-concentration moderator heavy water from the low-tritium-concentration heat transport system heavy water is an integral component of the CANDU design strategy to strictly control potential releases of tritium to the containment building atmosphere.

® <u>CAN</u>ada <u>D</u>euterium <u>U</u>ranium, registered trademark

A constant target in the design of future CANDU reactors is cost reduction. One potential mechanism for reducing costs is to decrease the number of parallel systems heavy water management systems. This may be possible if a cost-effective detritiation technology is available to provide design options for tritium management.

Tritium is produced in heavy-water reactors as a normal consequence of neutron absorption by the deuterium atoms of the heavy water in the moderator and heat transport system. Tritium also undergoes β decay with a half-life of 12.3 years to balance the rate of tritium production. In a CANDU reactor, the tritium inventory accumulates more rapidly in the moderator than in the heat transport water because the moderator water is continuously exposed to a neutron flux during operation while the heat transport water spends a large fraction of its time outside of the core and absorbs a lower neutron dose. As a consequence, the tritium concentrations slowly increase during normal operation with the tritium concentration always being about 40 times higher in the moderator than in the heat transport system.

CANDU reactors are designed for safe operation with tritium in the moderator and heat transport system up to the end-of-life inventories. The design is based on a defense-in-depth approach supported by management of the heavy water inventories in the reactor based on design principles that include:

- minimization of heavy water leakage from all systems,
- collection of heavy water from known potential leak points (e.g., heat transport pump seals),
- collection of vapour heavy water within the containment building, and
- segregation of the moderator water with its relatively high tritium concentration from the heat transport water with its lower tritium concentration.

The first tree design principles also minimize the potential losses of expensive heavy water from the reactor and contribute to control of the operating costs of the reactor. The design features to minimize heavy water (and tritium) losses from the reactor systems are supported by leak management monitors for tritium in containment air and monitors for heavy water/tritium in process water. Tritium management arises as a natural consequence of good cost-effective management of the heavy water inventory of the reactor.

Segregation of the high-tritium-concentration moderator heavy water from the low-tritium-concentration heat transport heavy water is important because of the potential for leakage of heat transport water is much higher. The heat transport system operates at a much higher temperature and pressure than the moderator and thus has a greater driving force for leakage to containment. In addition, the heat transport system is considerably more complex and has many more locations at which heavy water losses can occur (e.g., the fuelling machine and spent fuel handling system). Maintaining the tritium level in the heat transport system heavy water as low as possible helps to minimize tritium release from these potential leak points.

Strict segregation of high and low tritium level water contributes to reactor construction and operating costs and there is potential for cost savings in eliminating the duplication that segregation requires. The intent of this paper is to outline some of the opportunities for cost reductions that can arise with the availability of detritiation.

2. DETRITIATION

Cost savings through the redesign of heavy water management systems can only be achieved if there is a cost effective process available for tritium removal from the heavy water inventories. The cost savings possible with new heavy water management systems may not be the only consideration in offsetting the cost of detritiation. Reactor owners may also decide to invest in detritiation for the following reasons.

(1) Public safety perceptions. While current reactor stations operate well below regulatory limits, tritium releases contribute to the total station emissions. The release of any radioactive material has a negative public perception burden and there are pressures from the public (and the regulators) to implement ALARA steps to further reduce emissions.

(2) Dose reduction to workers. Tritium in the containment building air contributes to the radiation dose to plant workers and is a factor in the maintenance cost of a reactor. Elimination of tritium at the source (reducing the tritium concentrations in heavy water systems) is one of the health physics strategies that can contribute to reduced worker dose. However, the effectiveness of this strategy is dependent on other workplace safety practices.

The potential contribution of detritiation to tritium management will be tempered by the fact that this can be only one element of an overall strategy to limit potential tritium emissions. It is by no means certain that reduction of the tritium inventories in reactor water will lead to either lower tritium emissions or lower plant worker doses. Both of these objectives also depend on plant management and maintenance standards and practices. Nevertheless, a reduction in the tritium inventories in reactor systems would, in principle, give plant managers the ability to lower emissions and doses and greater flexibility to manage within safety and regulatory emissions limits and dose limits.

3. DETRITIATION TECHNOLOGY

Detritiation of heavy water can be accomplished using several different techniques. The technology chosen will depend on a number of economic and logistical factors. Among these, the quantity of water to be treated and the target level for detritiation loom the largest.

Two different approaches can be adopted for the detritiation of heavy water reactors. The first is establishment of a large central extraction facility with transportation of tritiated heavy water from several nuclear power stations. The second is the establishment of much smaller extraction facilities that are integrated into individual power stations. A significant difference between the approaches is the size and timing of the capital investment that is required.

One advantage of localized, detritiation at individual nuclear stations is the design flexibility that this allows. The tritium extraction facility could be directly integrated into the station heavy-water management systems minimizing the need for storage tanks and heavy water inventories in storage or in transit. Location of the extraction facility within the reactor containment building also can take advantage of the controlled containment atmosphere and tritium monitoring systems to manage potential tritium leakages from the detritiation process.

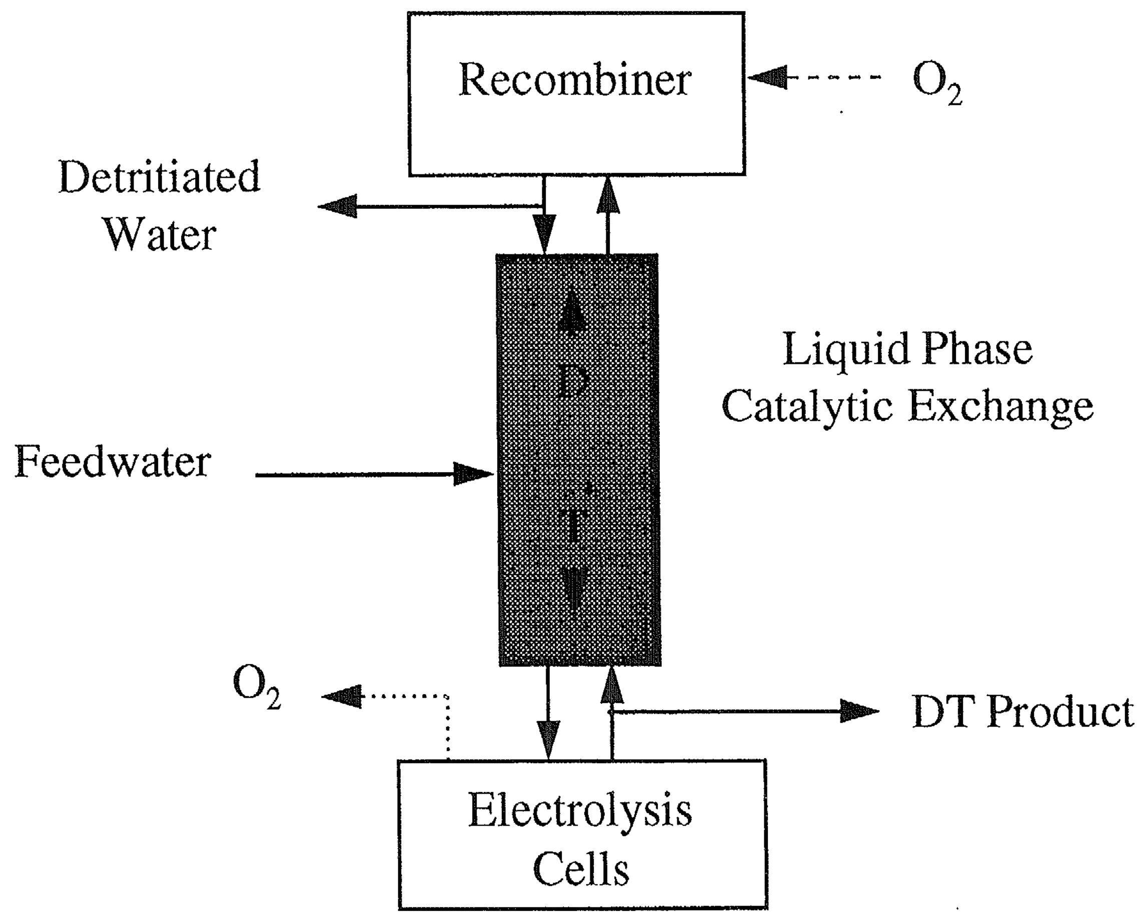

FIGURE 1 Combined Electrolysis and Catalytic Exchange Process (CECE)

4. AECL DETRITIATION TECHNOLOGY

AECL has developed a technology for the separation of the isotopes of hydrogen based on a catalytic exchange process. This technology is known as the Combined Electrolysis and Catalytic Exchange process (CECE)[[1]. The CECE process was originally developed for the separation of deuterium from normal hydrogen in light water, but it can also be used to separate trace levels of tritium from deuterium in heavy water.

The CECE process is illustrated schematically in Figure 1. A feedstock of tritiated heavy water is separated into a product stream of heavy water with a reduced tritium concentration and a product stream of deuterium gas enriched in DT. The degree of detritiation of the heavy water and the tritium concentration in the product gas stream can determined by the design of the system. The cost will increase with the degree of detritiation desired and will also be affected by the concentration of tritium in the deuterium gas stream.

The CECE technology provides an alternative to the conventional detritiation systems which include cryogenic distillation. Cryogenic distillation is a capital intensive process that is best suited to the treatment of large quantities of tritiated heavy water. In contrast, the CECE technology can be modularized to the size appropriate to an individual reactor with no capital cost penalty.

The deuterium gas product of a CECE detritiation facility requires further processing. In a large-scale central detritiation facility, the tritium could be further concentrated in this gas by cryogenic distillation. The concentrated tritium would then be stored for sale or disposal (through radioactive decay). For a CECE facility located at a reactor station, the tritium-containing deuterium gas can be stored on titanium beds directly. These beds can then be transported for further tritium concentration at a different facility elsewhere, if desired, or stored to allow tritium decay and the eventual recovery of the deuterium. A large fraction of the operating cost of a CECE facility would be the cost of the titanium beds and the sequestered deuterium. This cost is dependent on the concentration of the tritium in the deuterium gas that can safely be handled and stored. AECL is in the process of developing and proving the CECE detritiation technology.

5. CANDU DESIGN OPTIONS WITH DETRITIATION

With the availability of cost effective detritiation, there are opportunities for more flexibility in the design of the heavy water systems. The flexibility and range of options available will depend on the degree of detritiation that can be achieved. Cost savings can arise in three ways.

(1) Lower system design and construction costs.

One factor in the classification of nuclear systems for engineering design and construction is the maximum radioactivity content of the heavy water (essentially the tritium concentration for most systems). In Canada, systems which will contain less than 10 Ci/kg (370 GBq/kg)of tritium may be designed and constructed to the Canadian Standards Association nuclear code requirements of CSA-N285 Class 6. Systems which may contain more than 10 Ci/kg of tritium must meet the more stringent and more expensive requirements of CSA-N285 Class 3.

(2) Integration of parallel systems.

The availability of detritiation removes a key barrier to the transfer of water between the moderator and heavy water systems. This is particularly true if the detritiation facility is integrated into the plant heavy water management systems.

(3) Improved emissions control.

Reduction of tritium levels in heavy water systems is one element of an overall emissions control strategy. It may allow design changes and cost savings in other areas while achieving improved levels of emissions control.

Clearly the greatest potential for cost reduction would be provided by a detritiation facility with a capacity sufficient to keep the moderator tritium level below 10 Ci/kg and a detritiation efficiency sufficient to produce heavy water with a tritium concentration below 0.1 Ci/kg. Meeting the first target would open up the possibility of designing moderator auxiliary systems to a lower code classification. In practice, a detritiation facility would need the capacity to maintain the moderator at a level somewhat below 10 Ci/kg to provide a margin for plant operation. Achieving this would essentially require that the detritiation facility be sized to remove the annual production of moderator tritium (about 5 Ci/kg per annum in a CANDU 6). Meeting the second target would ensure that any detritiated water which is added to the heat transport system will not increase the tritium concentration in the heat transport water.

The following sections outline in the potential impact of detritiation on several of the heavy water systems in the CANDU reactor where changes could be made and cost savings found. The feasibility of these design changes have not been fully evaluated and hence the options discussed are speculative at present.

5.1 Heavy Water Supply System

The Heavy Water Supply System in the current CANDU reactor designs consists of three subsystems: the D_2O Storage System, the D_2O Cleanup System and the D_2O Upgrader. The D_2O Storage System provides storage tanks and associated equipment for the management of the reactor D_2O inventory. The D_2O Cleanup System provides ion-exchange columns, filters, strainers and associated equipment to remove contamination from heavy water recovered from spills or other sources. The D_2O Upgrader consists of distillation columns and associated equipment remove light water from recovered heavy water and maintain the heat transport and moderator heavy water at the high deuterium isotopic purities required for cost-effective reactor operation.

There are two parallel sets of equipment in all three subsystems (Figure 2 illustrates this with a schematic for the D_2O Storage System). One set of equipment is used to treat water with a high tritium concentration (for the moderator) and the second set is used to treat water with a low tritium concentration (for the heat transport system). The parallel heavy water cleanup and storage systems are normally kept isolated to prevent contamination of the low tritium water with high tritium water.

Detritiation would make possible a major redesign of the D_2O management systems. One option for integrating a detritiation facility into reactor heavy water management is shown

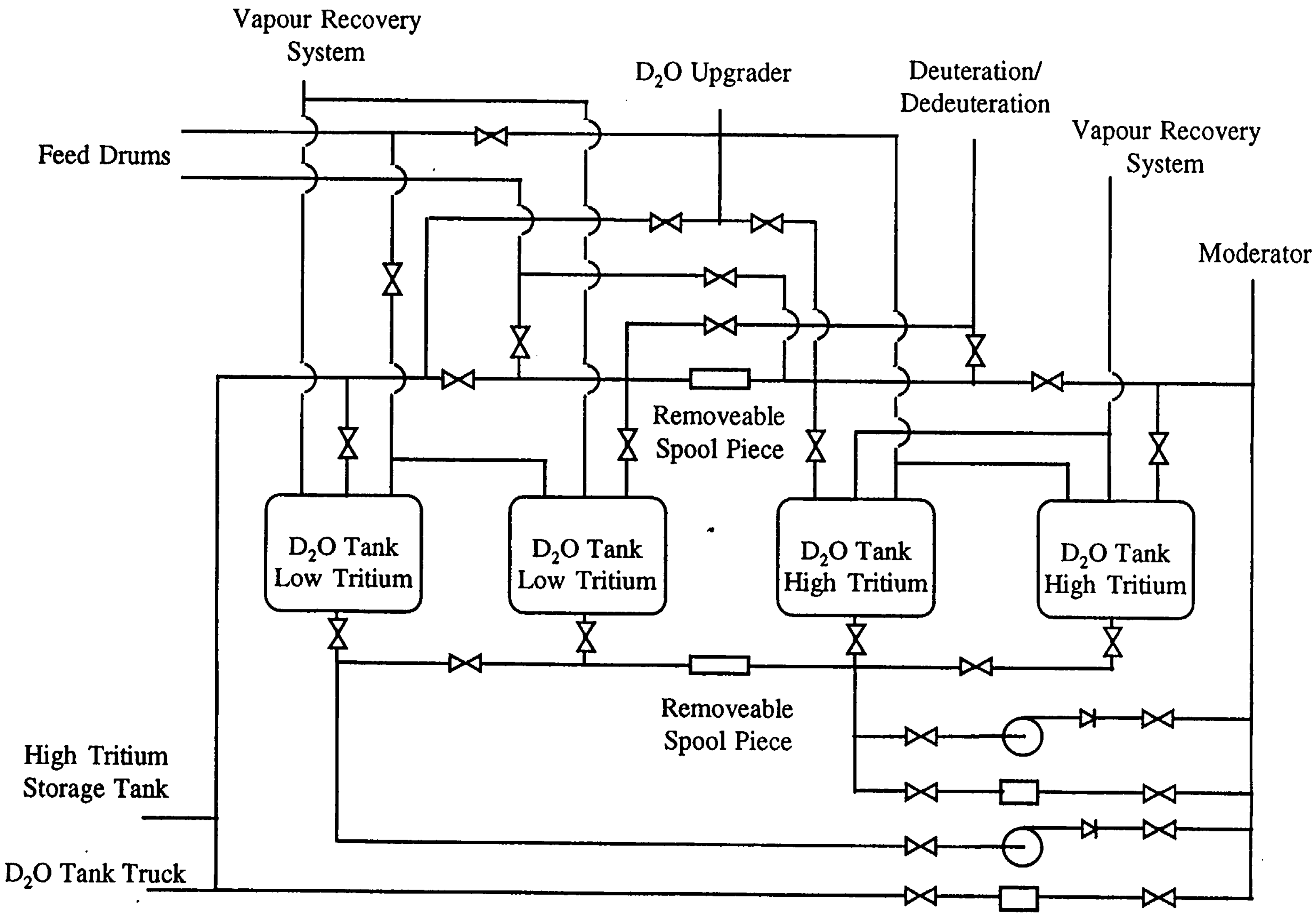

FIGURE 2 Heavy Water Supply System

schematically in Figure 3. Moderator heavy water with a relatively high tritium level would be treated in the detritiation facility and low-tritium water would be cycled back into the moderator and into the heat transport system. Heat transport water with a relatively low tritium level would also be cycled through the moderator after lithium removal.

In addition Figure 4 shows schematically how the other reactor water could be processed. Heavy water from the heat transport system, the leak collection system (and spills) and containment air driers could be processed by a single D_2O Cleanup System to remove lithium and other chemical contaminants. Clean water from this system would be transferred to a single heavy water storage system with three tanks for low-isotopic, high-tritium water, low-isotopic, low-tritium water and high-isotopic high tritium water. Water in the storage system would be processed by the Detritiation Facility and a single D_2O upgrader as required by reactor operations. Water from this the storage system would be used to supply the Deuteration/Dedeuteration System, moderator systems and the heat transport system (with addition of lithium as required to maintain the heat transport chemistry requirements).

This conceptional redesign of the D_2O management system eliminates a considerable amount of equipment. The cost implications of such changes will depend on the full implications of changing the D_2O management philosophy the loads on the remaining management systems and the cost of operating an integrated system. Specific impacts of this conceptual design include the following:

(1) The D_2O Storage System can be constructed to code Class 6 since the tritium concentration in all heavy water would be less than 10 Ci/kg.

(2) The D_2O Upgrader System can be constructed to code Class 6 since the tritium concentration in all heavy water would be less than 10 Ci/kg.

(3) The D_2O Cleanup System would remain code Class 3 to allow for safe handling of events where fission products may be released to the heavy water.

(4) With the elimination of one set of D_2O cleanup equipment, the remaining equipment would need to be redesigned to handle heavy water from all sources and remove lithium.

(5) The single distillation column remaining in the D_2O Upgrader System would need to be redesigned to handle all reactor heavy water.

(6) Elimination of one of the D_2O upgrader columns would save both money and equipment space. The latter impact would particularly help offset the requirements of adding a detritiation facility within the containment building.

5.2 Moderator Purification System

The Moderator Purification System is currently designed to code Class 3 because of the tritium inventory in the moderator water and the accumulation of activation products removed from the moderator water by the ion-exchange resins. Lowering the tritium inventory in the moderator water is unlikely to lead to a change in code classification and cost savings for this system because of the additional radioactivity burden in the ion-exchange resins.

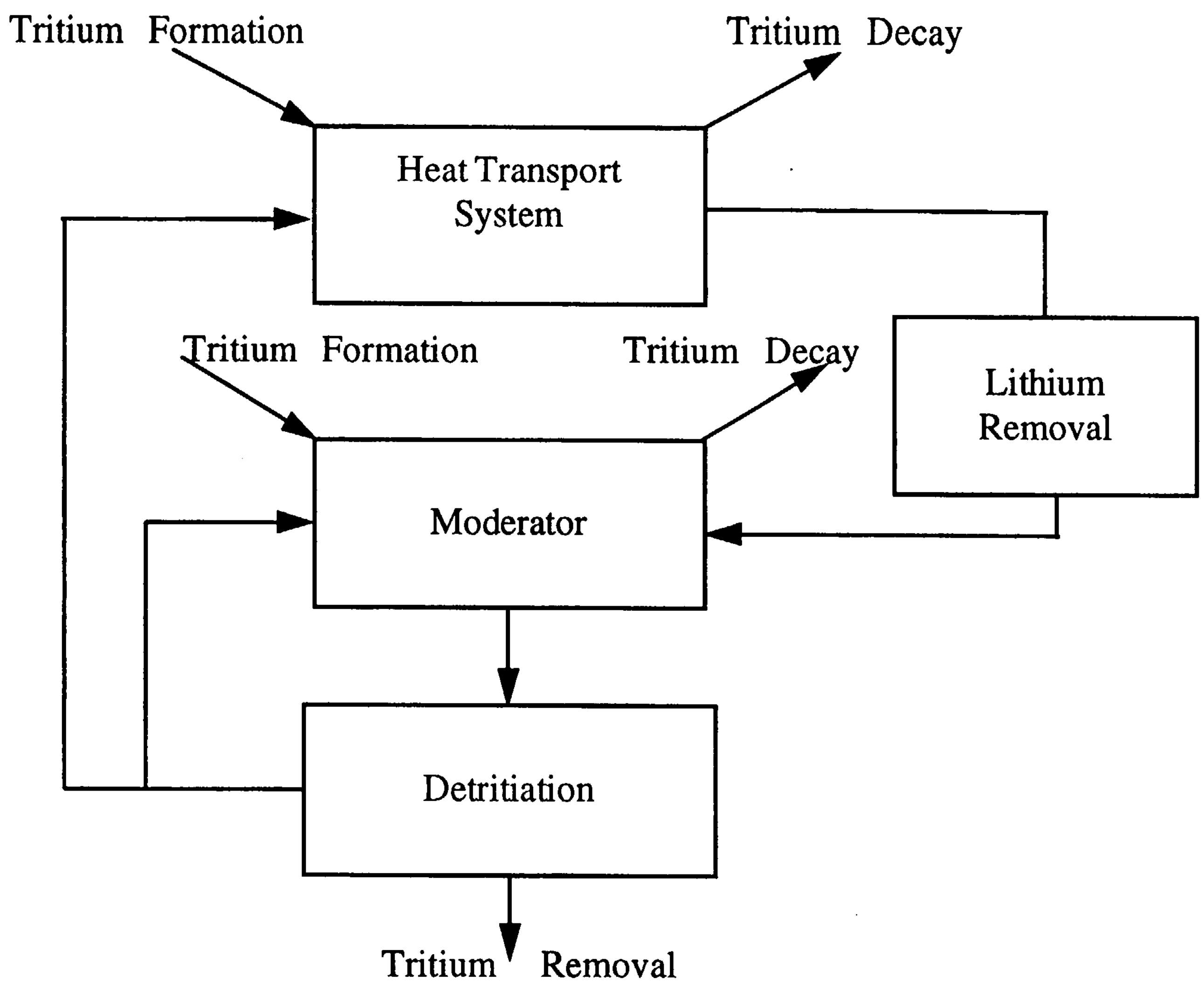
Tritium Formation
Tritium Decay
Heat Transport System
Tritium Formation
Tritium Decay
Lithium Removal
Moderator
Detritiation
Tritium Removal

Figure 3

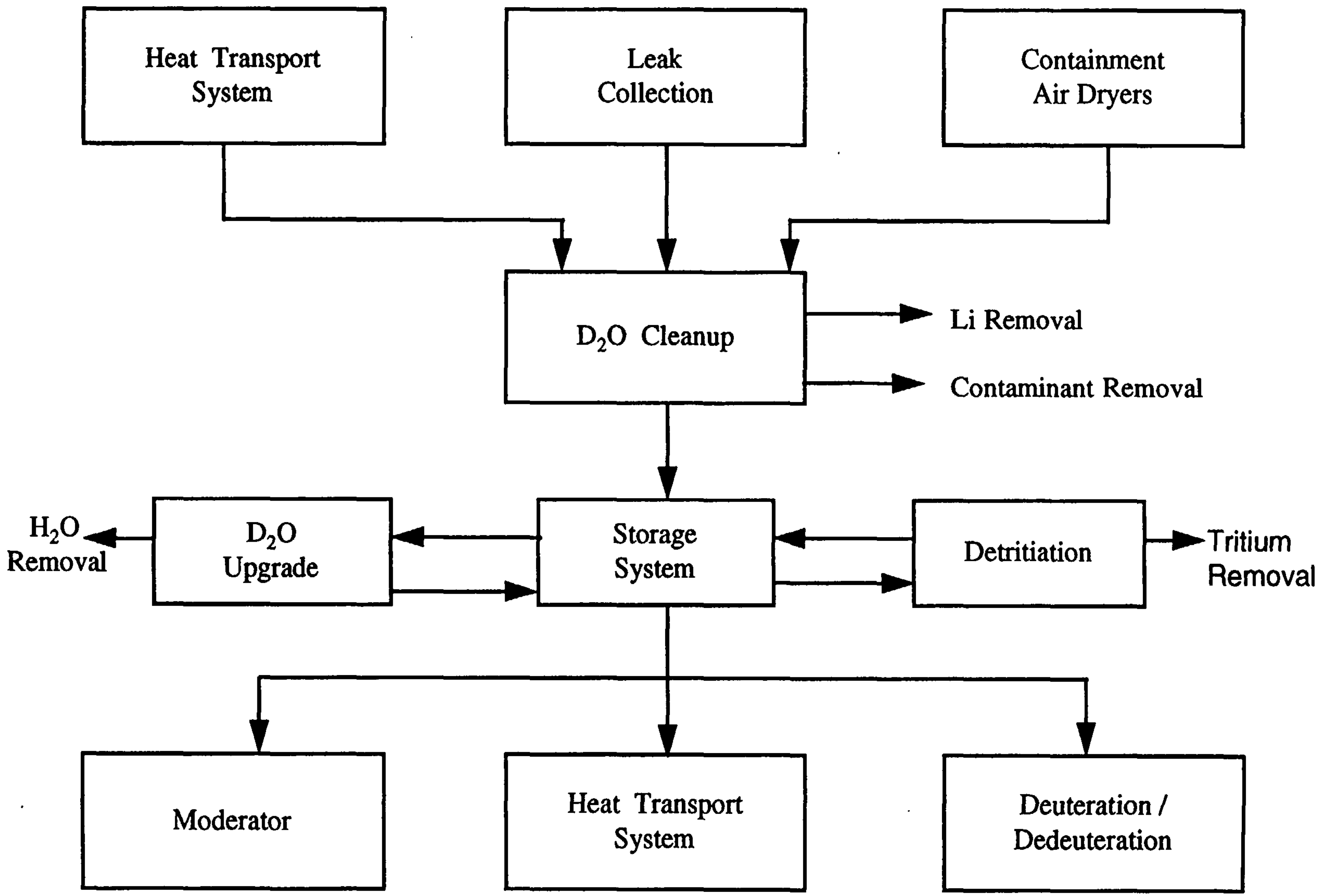

FIGURE 4 Heavy water processed by a single Heavy Water Cleanup System to remove Lithium and other Chemical Containments

5.3 Deuteration/Dedeuteration System

The Deuteration/Dedeuteration System is designed to remove light water from fresh ion-exchange resins prior to their use in the moderator and heat transport water purification systems and then to remove the heavy water from the spent resins. Two parallel duplicate systems are currently provided to handle the moderator and heat transport resins separately and prevent accidental cross-contamination of the water in the two systems. This prevents both tritium addition to the heat transport water and lithium addition to the moderator water.

With the provision of detritiation, it is possible to reduce the Deuteration/Dedeuteration System to a single unit for handling the ion-exchange resins of both the moderator and heat transport purification systems. This becomes possible if the heavy water used for the deuteration step can be taken from the output of the detritiation facility (at pH 7 to ensure the moderator safety systems are not compromised). Heavy water removed from the resins can be transferred to the D_2O Cleanup System and then recycled.

The Deuteration/Dedeuteration System is currently Class 3 because of the tritium level of the heavy water and the inventory of radioactivity on the spent resins being dedeuterated. Since detritiation will not affect the latter activity levels, those elements of the system that will contain the spent resin will need to remain Class 3.

5.4 Moderator Cover Gas System

The Moderator Cover Gas System is a closed gas-recirculating system designed to limit the buildup of deuterium (produced by D_2O radiolysis) in the gas spaces above the moderator. This system is currently designed to code Class 3 based on the tritium concentration in the moderator water and safety analyses of the occupational dose consequences of a system failure. If the moderator tritium concentration can be maintained below 10 Ci/kg it may be possible to redesign this system to code Class 6.

5.5 Heat Transport/Moderator D_2O Collection Systems

The D_2O Collection System consists of two parallel sets of equipment for the collection and transfer of both the moderator and heat transport water from various sources within the reactor building including pump and valve seals. If the D_2O management system is redesigned to manage and detritiate the heavy water collected from all leak points, then half of the current collection tanks and pumps could be eliminated. At present it is possible to directly recirculate leakage water back to the source system. This design change would eliminate direct recirculation of recovered heavy water leaks to either the moderator or heat transport system and would increase the load on the D_2O Cleanup and D_2O Upgrader Systems. Analysis is required to determine whether this would be a cost effective improvement.

5.6 D_2O Vapour Recovery System

CANDU reactors have a D_2O Vapour Recovery System to capture D_2O which escapes to the containment atmosphere from the reactor systems as a result of leakage (during normal operation or maintenance). The recovery system is based on separation of the reactor building into three zones:

(1) The reactor inlet/outlet vaults and steam generator enclosure (accessible only during reactor shutdown),

(2) The accessible area.

(3) The moderator equipment enclosure.

The three zones are designed to provide segregation of the building atmosphere based on tritium content. The segregation is achieved by appropriate air flow barriers and differential pressures to ensure air flow from low-tritium to high-tritium zones. Each zone is provided with ducting, instrumentation and air driers to separately collect D_2O with different tritium contents.

The D_2O vapour recovery system provides two functions. It prevents the escape of leaked D_2O and allows for recycling of a valuable capital asset. It is also a component of the health physics strategy to minimize the dose from airborne tritium to plant workers. For the latter function, the separation of the reactor building into separate zones and the isolation of the moderator enclosure is important because of the high tritium content of water that may leak from moderator systems.

If the moderator tritium level can be maintained below 10 Ci/kg, there may be an opportunity to eliminate the requirement for a separate moderator zone. The reactor containment could be managed with only two zones, only one of which would be normally accessible during reactor operation. This change in ventilation philosophy requires not only low tritium levels in the moderator, but also good maintenance practices on moderator systems and may not be cost effective to implement.

6. CONCLUSION

Detritiation may be considered as one component of the strategy to safely manage the tritium inventories in a heavy water reactor. AECL is working on the development of new technology to remove tritium from heavy water at individual reactors. Detritiation would add to the cost of constructing and operating a reactor. However, there are a number of design changes which on-line detritiation could make possible that could result in substantial cost savings. If the detritiation technology is not too expensive, the advantages gained by redesign could lead to overall savings in the cost of future reactors.

The full benefit of cost savings from detritiation would arise from three major changes in the reactor design philosophy:

(1) Elimination of duplicate parallel equipment for moderator and heat transport water D_2O management system.

(2) Combination of the reactor D_2O vapour and liquid recovery systems.

(3) Reduction in the design requirements (where warranted) to the lower code classification permitted by low tritium levels.

Several conceptual design changes that could be considered with detritiation have been discussed in this paper. The feasibility of these conceptual designs and their cost implications have not been fully evaluated. Until further information becomes available on the costs of improved detritiation technology and a more detailed evaluation of design options is performed, there can be no conclusion regarding the overall cost benefit of detritiation.

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A PASSIVE EMERGENCY HEAT SINK FOR WATER COOLED REACTORS WITH PARTICULAR APPLICATION TO CANDU REACTORS

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Abstract

Water in an overhead pool can serve as a general-purpose passive emergency heat sink for water-cooled reactors. It can be used for containment cooling, for emergency depressurization of the heat transport-system, or to receive any other emergency heat, such as that from the CANDU® moderator.

The passive emergency water system provides in-containment depressurization of steam generators and no other provision is needed for supply of low-pressure emergency water to the steam generators.

For containment cooling, the pool supplies water to the tube side of elevated tube banks inside containment. The elevation with respect to the reactor heat source maximises heat transport, by natural convection, of hot containment gases. This effective heat transport combines with the large heat-transfer coefficients of tube banks, thereby reducing containment overpressure during accidents. Cooled air from the tube banks is directed past the break in the heat-transport system, to facilitate removal of hydrogen using passive catalytic recombiners.

1. INTRODUCTION

New designs of water-cooled reactors as summarized by Ritzman [1] use pools of water for passive emergency heat storage and rejection. In the event of a loss-of-coolant accident (LOCA), the simplified boiling-water reactor (SBWR), by General Electric, uses a pressure-suppression pool within containment to condense steam from the reactor pressure vessel (RPV), and a heat exchanger in another pool, outside containment, to condense steam from the containment atmosphere. Steam from the RPV is condensed to depressurize the RPV, enabling a flow by gravity of emergency coolant. Steam from containment is condensed to limit the increase in pressure and maintain containment integrity.

Similarly, the AP600, by Westinghouse, uses a water pool inside containment to condense steam from the RPV, facilitating gravity-feed of emergency coolant from that pool. Containment cooling is effected by heat transfer through a steel containment vessel to the outside air flowing upwards by natural convection. The air-side heat transfer is augmented by the evaporation of water, which flows from an elevated tank against the upflow of air and over the outside of the containment vessel.

In a LOCA, a passive concept by Spinks [2] uses a water jacket to cool the containment atmosphere and, as in CANDU reactors, the heat-transport system is depressurized by depressurizing the secondary side of the steam generators, rather than by discharging the primary fluid. Depressurization of the CANDU primary heat-transport system is needed to facilitate injection of emergency coolant. Depressurization is done by discharging steam from the steam generators to the external atmosphere, and then water is supplied by gravity from an elevated tank to the steam generators. In the event of a LOCA, with coincident loss of emergency coolant injection (ECI), depressurization of the heat-transport system, together with moderator cooling of the fuel-channels, ensures fuel channel integrity even under these conditions.

A drawback of the AP600 and the approach of Spinks [2] is the elevation of the heat sink, the containment wall, with respect to the heat source. Heat should be transferred from the containment atmosphere at the highest possible elevation to maximize natural circulation and heat transfer from containment gases. A further benefit of enhanced natural circulation would be the availability of a flow of air within containment for passive hydrogen mitigation.

Another drawback is the poor heat-transfer coefficient that is usually encountered with flow tangential to a surface, as would apply both inside and outside the containment wall. The coefficient is typically an order of magnitude larger for flow across a tube bank [Compare Figures 13.3-1 and 13.3-3 of Ref. [3]]. A similar drawback is the limited heat-transfer surface area that is available in these designs.

A drawback of the CANDU reactor approach is the release of steam outside containment in order to depressurise the reactor during an accident. Operator action is required to switch to an alternative heat sink in the event of steam-generator tube failures.

The intent of this report is to describe a passive emergency-water system (PEWS) that overcomes these pressurized water reactor (PWR) and CANDU reactor weaknesses. It uses the SBWR feature of an elevated pool vented outside containment but, for containment cooling, applies the water in a different manner maximizing heat transfer from containment gases and generating a flow of air for hydrogen mitigation. Maximum heat transfer leads to minimum containment over-pressure, which reduces cost and radioactivity releases.

2. PEWS AS A HEAT SINK FOR CANDU

Figure 1 illustrates PEWS as applied to a CANDU reactor.

The vented water pool in the containment dome is a general-purpose emergency heat sink. For a large CANDU reactor, a volume of some 2000 m^3 provides, after boil off, a three-day heat sink. It serves as a heat sink for the steam generators, for containment, and for the moderator acting as an emergency-core-cooling (ECC) system. It could also serve as a heat sink for the ECI system, but this might be better designed to be independent of the moderator system. For redundancy, the PEWS tank can be divided azimuthally.

2.1 Steam Generator Heat Rejection

Given a LOCA or steam-line break, the steam generators would be isolated from the main steam line using valves located close to each steam generator: see the normally-open valves in Figure 1. Following isolation, valves to connect the steam generators to the emergency heat sink would be opened: see the normally-closed valves in Figure 1. The steam generators would be depressurized by condensation of steam in condensers located in the PEWS vented water pool. The condensate would return by gravity to the steam generators. This return flow eliminates the need for a supply of emergency water, commonly fed from an overhead tank.

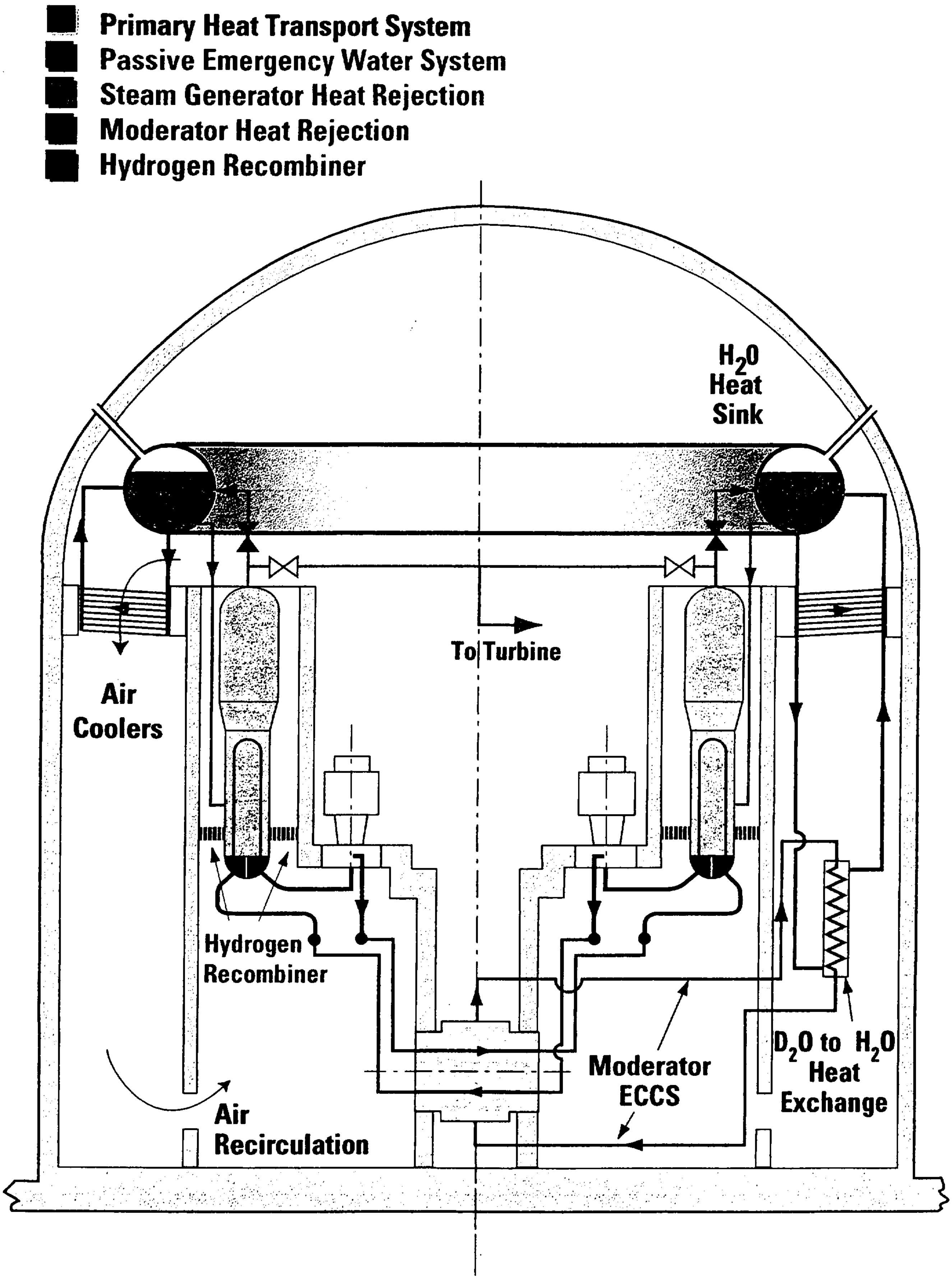

Figure 1: Passive Emergency Water System

In the event of a steam-line break, depending on the speed of reactor trip and speed of closure of the main-steam-line isolation valves, the discharge of steam into containment is reduced. The steam-line break need not be a limiting break from the viewpoint of containment overpressure.

The PEWS tank could be subdivided into a division per steam generator. Then, in the event of a loss of water in one part of the pool, the remaining parts would be available to provide a limited-duration heat sink.

2.2 Containment Heat Rejection

In the SBWR, in a LOCA, containment heat is rejected firstly to a pressure-suppression pool and, in the long term, to a vented pool via condensers in the pool. The tube side of the condenser is open to the containment atmosphere. The pressure-suppression pool is needed because of a limitation in size of condensers in the vented pool. This limitation is overcome in PEWS by employing tube banks located at an upper elevation within containment. Water from the vented pool is supplied to and returns from the tube side of the tube banks by natural circulation via vertical headers.

The tubes are inclined to the horizontal, so that there is a preferred flow direction for the water. Water is supplied from the pool via the header at one end, and heated water is returned to the pool via the header at the other end of the tube bank. In the longer term, a boiling steam/water mixture is returned to the pool.

The containment itself is divided into an inner zone and an outer annular zone, as shown in Figure 1. Such is normally the case in CANDU containments, the outer zone being an accessible area that is not connected to the inner zone. However, a connection is required in this design, at least during accidents, to permit a natural-circulation flow of gases up through the steam-generator enclosure and down through the accessible area. The flow is induced by the difference in density of the hot gases (air, steam and hydrogen) rising from a break in the reactor coolant pipes, and the gases cooled in passing across the elevated tube banks. The elevation difference between the heat source, near the reactor, and the heat sink, at the tube bank, is larger than in other designs, leading to an enhanced natural-circulation flow. The flow is further enhanced by using the internal wall as a baffle, which eliminates restrictive interaction between downflowing and upflowing streams.

A reduced containment pressure follows from a high rate of heat transfer from the containment gases to the banks of water-cooled tubes. A large heat-transfer coefficient follows not only from the enhanced flow velocities, but also from the high heat-transfer coefficient for flow across a tube bank, compared to the coefficient for flow tangential to a surface, such as a vertical containment wall.

2.3 Hydrogen Mitigation

The enhanced natural-circulation flow within containment permits improved hydrogen mitigation. Hydrogen mitigation can be accomplished by directing the recirculating air to the source of the hydrogen and locating catalytic hydrogen recombiners in the stream of mixed air, steam and hydrogen. If the recombiners are located at low elevation, the heat of recombination acts to augment the buoyancy-induced flow.

Figure 1 shows hydrogen recombiners located in the steam-generator enclosure. The intent is to blank off any alternative flow path, so that the entire recirculating air flow is available for hydrogen mitigation. This greatly increases the effectiveness of recombiners, compared to the conventional strategy of dispersing recombiners throughout containment and relying on local convective flows for supply of air to each recombiner. In addition, recombiners placed in a location of known high hydrogen concentration are more effective than recombiners placed at other locations, where the hydrogen concentration would be expected to be diluted. Also, as the recombiners are moved closer to the hydrogen source, the mass of hydrogen between source and sink is reduced, reducing the effects of a sudden deflagration.

Catalytic hydrogen recombiners located in a strong flow of air can have single-pass efficiencies above 80%, leading to small exit hydrogen concentrations.

2.4 Moderator and Emergency Coolant Heat Rejection

PEWS can be a heat sink for the CANDU moderator, which, in surrounding the fuel channels, can act as an ECC system [4]. Moderator emergency heat rejection would be done passively, by transferring heat from a D_2O natural-circulation loop to a H_2O natural-circulation loop, the latter being part of PEWS, as shown in Figure 1.

Full-height, but reduced-scale, testing of a natural-circulation loop, driven by flashing of water to steam as it rises to an elevated heat exchanger [5], has demonstrated the feasibility of this mode of moderator heat rejection.

The PEWS pool could similarly accept heat from the ECI system. However, this would detract from the potential independence of the two ECC systems. Two fully-independent ECC systems can lead to a CANDU core-melt frequency of less than 10^{-7} per unit year [6].

Advantages for CANDU

To summarize, the advantages of PEWS as applied to CANDU reactors are as follows:

a) Steam-generator depressurization inside containment avoids the need for operator action in the event of failed steam-generator tubes.

b) The condensed water is returned to the steam generator, avoiding the need for another source of emergency water.

c) Passive containment heat rejection, which avoids any dependence on emergency power supplies, is effected in an optimal manner: the elevated heat sink maximizes the flow of emergency water and the flows of containment gases. Containment design pressure is reduced, reducing containment cost and radioactivity release.

d) The containment air-flow facilitates hydrogen mitigation.

e) Passive moderator heat rejection, which avoids any dependence on emergency power supplies, is effected in an optimal manner: the elevated heat sink maximises the flows of light water and heavy water.

3. CONCLUSIONS

A passive emergency water system (PEWS) has been described that uses an elevated and vented pool of water that can act as a general-purpose passive heat sink for water-cooled reactors. Containment cooling and hydrogen mitigation are improved compared with other designs of passive water-cooled reactors. The application to and advantages for CANDU reactors have been described.

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CANDU PRESSURE TUBES:
ASSESSMENT AND IMPROVEMENT OF SERVICE LIFE

(Session 6)

Chairman

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India

IMPROVING THE SERVICE LIFE AND PERFORMANCE OF CANDU FUEL CHANNELS

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Abstract

The development objective for CANDU fuel channels is to produce a design that can operate for 40 years at 90% capacity. Steady progress toward this objective is being made. The factors that determine the life of the channel are reviewed and the processes necessary to acheive the objectives identified. Performance of future fuel channels will be enhanced by reduced operating costs, increased safety margins to postulated accident conditions, and reduced retubing costs compared to current channels. The approaches to these issues are discussed briefly in the paper.

1. INTRODUCTION

The fuel channels in a CANDU reactor consist of the pressurized components (pressure tubes, end fittings and closures), the primary system internals (shield plugs) and the support components (spacers, calandria tubes), Fig. 1. The evolution of the CANDU design has been in conjunction with steady improvement in the performance of most of these components in response to design and service demands. The pressure tube determines the life of the channel because it has to withstand the most severe operating conditions, and considerable research and development effort has been expended to understand and improve its service behavior.

Deformation occurring under neutron flux either by creep or growth will ultimately limit the life of a pressure tube. To a degree the deformation can be accommodated by the design of the channel. However, the life of a pressure tube can also be limited by an increased probability for fracture as service life increases. To prevent fracture the designers have to protect against fracture initiation and also establish the applicability of Leak–Before–Break (LBB) for the service life. Research and development efforts have made significant progress in reducing fracture to a very low probability event and thus eliminating LBB as the criterion that limits the service life of pressure tubes.

Improvements in the performance of CANDU reactors can be undertaken to meet a number of objectives such as:

- longer life
- reduced operating costs
- increased power output
- increased safety margins under postulated accident conditions
- reduced retubing costs

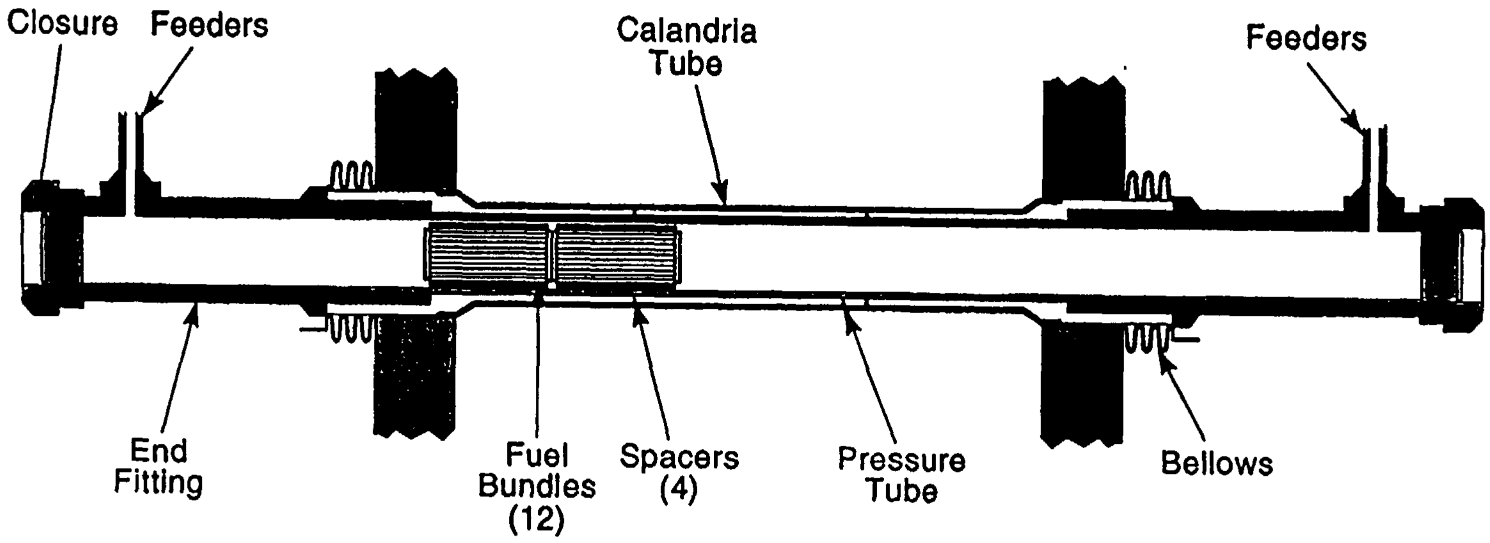


Figure 1 Simplified Illustration of a Fuel Channel

This paper discusses the first item in some detail and outlines the approaches to the others. In these discussions, no pressure tube alloy other than Zr-2.5Nb is considered, although the alloy may be modified to optimize its behavior.

2. LONGER LIFE

The service life of the current CANDU 6 fuel channels is expected to be 30 years at 85% capacity providing 25.5 Effective Full Power Years of operation.(EFPY). However, the designers expect to meet a requirement for 30 years at 90% capacity in the near future and the target is eventually to meet 40 years at 90% capacity (36 EFPY) which represents a 33% increase in life expectancy over the early units.

Pressure tubes have reached the end of their service life when they can no longer perform their design function. The most important factors are changes in dimensions and the increase in the probability of fracture. The forms of deformation that are of greatest concern are elongation and increase in diameter. Elongation may lead to excessive feeder connection stressing, whereas diametral expansion causes flow bypass around the fuel bundles which may reduce the margin between normal operating power and the critical channel power. The probability of fracture and the need to involve LBB depend on flaw development, hydrogen concentration and fracture toughness. From a review of the needs of the design to provide low maintenance over the expected life, the designers have defined the values of the properties that will be required to achieve the target lifetime, Table 1.

To deal with fracture, a strategy of defence in depth is used; crack initiation is prevented and LBB is assured by having adequate time available to detect a through wall crack by its leakage before it becomes unstable. Both of these requirements should be met with adequate margins over operating conditions.

2.1 Deformation

Progress has been made towards reducing elongation in reactor. A modified fabrication route Ref.[1], that decreases the dislocation density by allowing a heat treatment at 500°C for 6 hours has been shown to reverse the direction of irradiation growth in the longitudinal direction. The total in reactor deformation is usually taken as the sum of irradiation creep and irradiation growth, thus the reversal of growth implies a reduction in elongation. This reduction has been demonstrated in tubes installed in Bruce Unit 8 which are elongating about 30% slower than standard cold worked tubes in neighbouring channels.

Evaluation of the probable effects of microstructural features that could reduce the diametral expansion without increasing the elongation has indicated the desirability of retaining the strong transverse texture and reducing the thickness of the plate like α–grains sufficiently for their boundaries to dominate as sinks for mobile point defects, the source of irradiation induced deformation. This will require a modification to the extrusion process, or higher cold reduction of the wall thickness than the current practice, followed by stress relieving to reduce the dislocation density without recrystallising the α–grains, or both, to achieve the required microstructure.

One practical method currently employed to reduce diametral expansion is to match the variation in properties along the length of the pressure tube that result from extrusion, to the temperature and neutron flux variations seen in service. Typical axial temperature and flux profiles in

TABLE 1. END OF LIFE PROPERTIES OF PRESSURE TUBES FOR 30 YEARS AT 90% CAPACITY.

Property	Current tubes		Target for future tubes	Comment
	Pre-1993 Specification	Post-1993 Specification		
Hydrides present? - outlet end	Yes	No	No	Initial concentration at limit of specification. Current maximum ingress rate 0.015 at%/year (3 ppm D/year). Assumes rate of pick-up constant for 30 years.
- inlet end	Yes	Yes	No	Initial concentration at limit of specification. Current maximum ingress rate 0.005 at%/year (1 ppm D/year).
- region of high tensile stress at rolled joints	Yes	Yes	No	
K_{IH} (MPa$\sqrt{m}$)	4.5	4.5	>10	Little further decrease with irradiation.
DHCV at 250°C (m/s)	2×10^{-7}	2×10^{-7}	$<7 \times 10^{-8}$	
CCL at 250°C and 10 MPa (mm)	42	>80	>80	Data from burst tests.[14]
Diameter change (%)	5	5	<3	Linear behaviour, no rupture.
Elongation (mm)	230	230	<100	Current value based on increasing rate.

CANDU reactors are shown in Figure 2. The incremental rise in temperature from inlet to outlet end of the channel is proportional to the flux profile. There are two general types of flux profile, a cosine shape as shown in Fig. 2a and a "flattened-cosine" shape as shown in Figure 2b. The back end of the tube (the end that exits the extrusion press last) has a smaller α-grain size and a higher dislocation density and hence a higher tensile strength and higher in-reactor deformation behaviour than the front end.

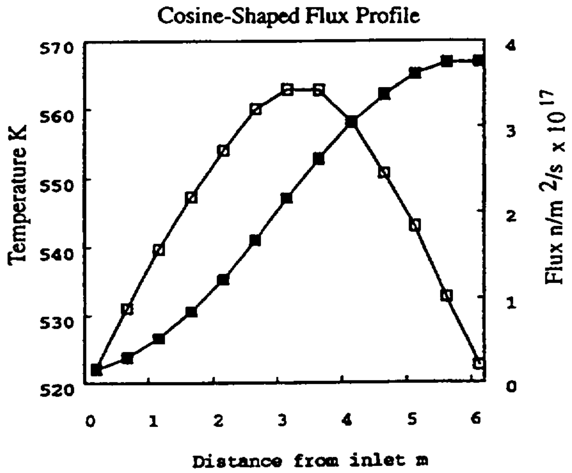
The transverse deformation of the pressure tube varies depending on whether the back end or the front end of the tube is installed at the inlet end of the channel. In channels with a cosine-shaped flux profile, the strain profile when a back-end is installed at the outlet is only displaced slightly towards the outlet compared to the case where the back-end is installed at the inlet and is almost symmetrical, Fig. 3a. Whereas in channels with a flattened-cosine shaped flux distribution, there is a pronounced peak in strain rate near the outlet end of the channel when the back end of the tube is at the outlet, Fig. 3b. A more uniform deformation behaviour and slightly lower maximum rate can be obtained by installing the tubes with their back ends at the inlet end to the channel. This benefit is also present, but to a much lesser degree in channels which have a cosine-shape flux profile. Models have been developed to account for the dependence of diametral strain on the temperature, flux and microstructure of the pressure tubes Ref. [2]. The calculated behaviours, attribute the variation to the end-to-end variations in the crystallographic texture produced during fabrication of the hcp zirconium alloy, the α-phase grain size and shape and the dislocation density.

2.2 Crack Initiation and Propagation

Of the various mechanisms of crack initiation and propagation, the most likely is Delayed Hydride Cracking (DHC). To prevent DHC, the following risk factors must be absent:

- hydrides; the hydrogen concentration must be less than the terminal solid solubility limit (TSS) so that hydrides do not form,
- tensile stress; the total applied tensile stress and its amplification by flaws must be less than the critical value to fracture hydrides, and
- time; duration of exposure to the first two risk factors must be minimized.

The most important defence against DHC is to eliminate hydrides. This is achieved by limiting the total hydrogen concentration, consisting of hydrogen present after fabrication and deuterium absorbed from corrosion, to not more than 0.3 at % anywhere in the pressure tube during operation for the design life. Crack initiation would be minimized at low temperature, if the minimum value of K_{IH}, the threshold stress intensity factor for DHC, was 10 $MPa\sqrt{m}$. With this value, a flaw necessary to start DHC at the most highly stressed section of the tube - the rolled joint, would be easily detected by modern NDE techniques. If a crack does develop and it is not detected by NDE, it should manifest itself as water leakage detected in the annulus gas system surrounding the pressure tube. The principle of LBB, as practiced in CANDU reactors, requires that the time available to detect moisture from a leaking crack before it becomes unstable is much greater than the time required to detect the leak Ref. [3].



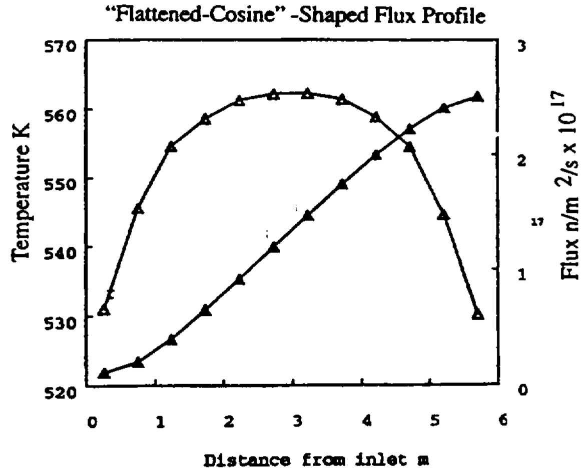
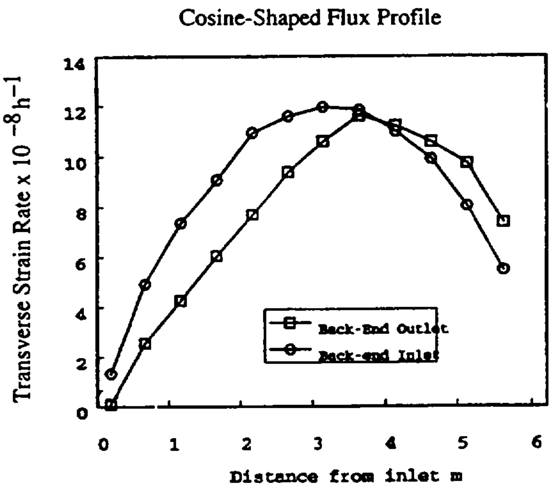


Figure 2a Typical Coolant Temperature and Fast Neutron Flux Profile in CANDU Fuel Channels; Cosine Shaped Flux

Figure 2b Measured Transverse Strain Rate Profiles for Channel with a "Flattened-Cosine" Flux Profile



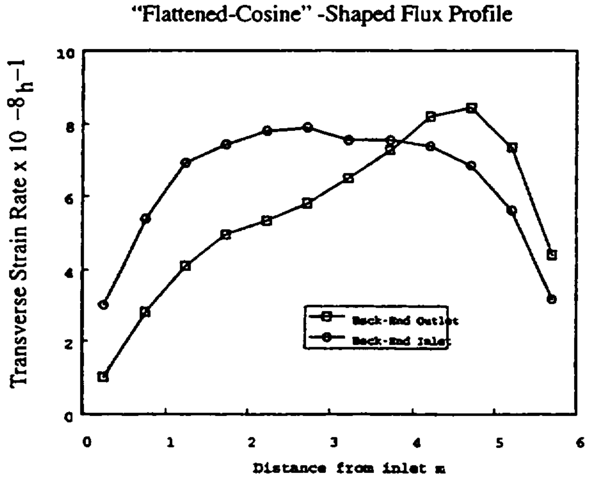


Figure 3 Measured Transverse Strain Rate Profiles for Channel with a "Cosine" Flux Profile

Figure 3b Measured Transverse Strain Rate Profiles for Channel with a "Flattened-Cosine" Flux Profile

2.2.1 Maintaining Hydrogen Concentration Below TSS

The potential methods for maintaining the pressure tubes hydride–free are:

- minimize the initial hydrogen concentration,
- minimize ingress during operation, and
- move the hydrogen to innocuous locations.

2.2.1.1 Initial Hydrogen Concentration

Up to 1993 the specification allowed a finished pressure tube to contain up to 0.23 at% (25 ppm) of hydrogen. In practice, tubes contained an average of 0.095 at% (10.3 ppm) hydrogen, but values as high as 0.16 at% (18 ppm) were observed. With the latter concentration, hydrides would be present at 217°C Ref [4]. If deuterium is picked up at 0.005 at%/year (1 ppm/year) at the inlet end of a channel, (250°C), then hydrides would be present after 26 years of operation in a tube with the high initial hydrogen concentration. Thus there was a large incentive to reduce the initial hydrogen concentration.

The stages and locations at which the hydrogen enters the zirconium during tube fabrication were identified and used in its reduction. The implementation of these improvements Ref. [5], led to a reduction to 0.05 at% (5 ppm) of the initial hydrogen allowed by the current specification and is illustrated for recent production in (Fig. 4).

Since there is higher deuterium ingress at rolled joints, hydrides form early in life at the end of the pressure tube where it is supported by the end fitting. With time, the front of hydrides moves inboard and eventually passes the location where the tensile stresses are at a maximum. The aim is to postpone when the hydride front reaches this location. Using the maximum ingress rate of the inlet end of a Bruce type reactor and the maximum pick up rate of deuterium from corrosion (0.005 at%/year), the predicted position of the hydride front as a function of time (Fig. 5), shows that a tube with an initial hydrogen concentration of 0.15 at% had hydrides present at the critical location after 15 years, but this time was increased to 25 years when the initial hydrogen concentration was reduced to 0.05 at%.

2.2.1.2 Ingress During Operation

Two approaches to reducing ingress during reactor operation are to modify the surface and to control the composition of the alloy. Several surface modifications have been studied and shot peening has been selected as the most promising technique.

Shot peening the surface obliterates the original microstructure with a heavily cold worked layer that may be recrystallised with subsequent heat treatments to produce a microstructure with a reduced concentration of niobium in the α–phase and a uniform dispersion of small β–Nb precipitates, that is almost optimum for corrosion resistance. The results of out reactor tests are very promising Ref [6]; in either the shot peened or shot peened and heat-treated conditions both oxidation and deuterium pick up are approximately halved compared with as received material. If this behaviour can be maintained in reactor, we can have high confidence in having hydride free pressure tubes for 30 years.

It is unlikely that our current composition of Zr–2.5 Nb is optimum for corrosion and deuterium ingress. Small additions of single elements to Zr–2.5 Nb have been made Ref. [7] and the results

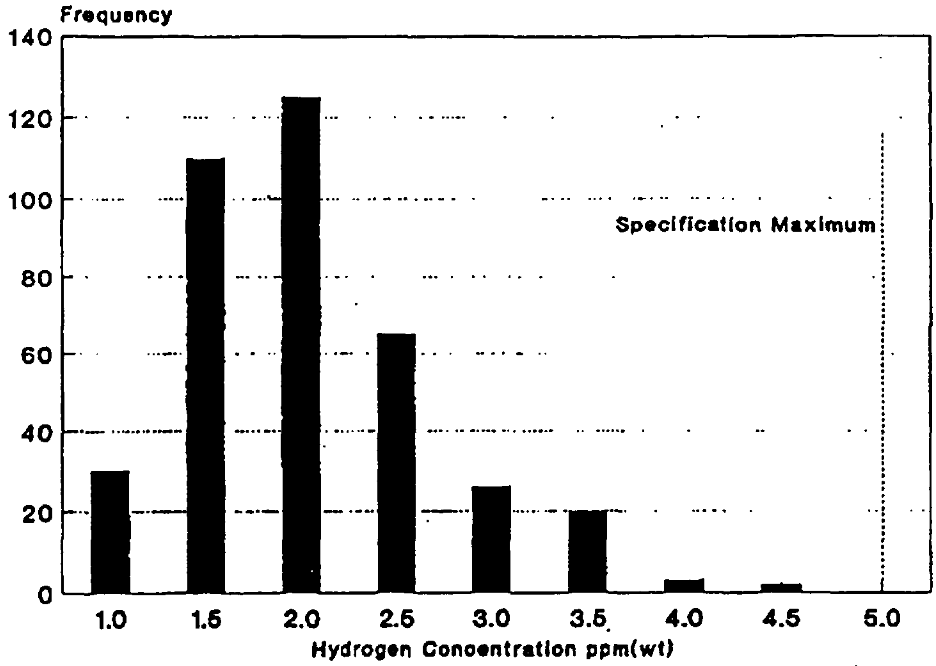


Figure 4 The Measured Hydrogen Concentrations in a Recent Production Order of Pressure Tubes

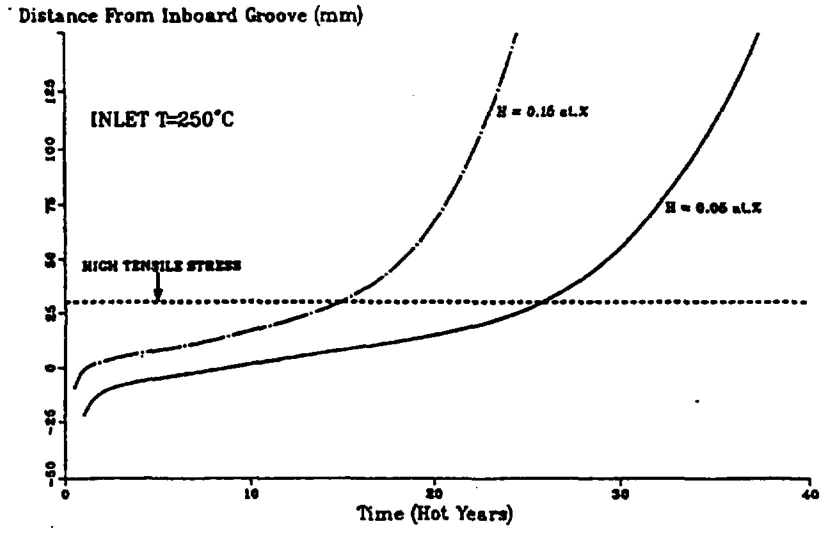


Figure 5 Effect of Initial Hydrogen Concentration on the Time to Precipitate Hydrides at the Inlet end Highest Stress Location

to date show that Ni, Mn and Ti should be avoided while Cr, Fe, Mo and Si, are beneficial. Batches of Zr–2.5 Nb–400 ppm Cr, -1100 ppm Mo and –Fe/Sn/Cr have been made commercially and are showing promise in out–reactor tests.

The extra deuterium absorbed at the ends of the pressure tubes is caused by galvanic and crevice corrosion between the pressure tube and end fitting. A chromium layer between the two components reduces the ingress by minimizing galvanic effects and reducing the transfer of deuterium from the steel into the zirconium (Ref.[8]. Many joints with the chromium layer have been made and are being tested in a D_2O loop. Results indicate a reduction of the extra deuterium ingress by between 50 and 90%. An illustrative calculation for the rolled joint region shows that a reduction in the extra deuterium ingress by 50% would provide an increase of 4 years before hydrides are present at the critical point while a reduction of 80% would provide a margin of over 10 years on the target operating lifetime of 30 years.

2.2.1.3 Relocating the Hydrogen

The principle depends on placing a material such as yttrium, that forms a more stable hydride than zirconium into an innocuous position where it can getter the hydrogen and minimize the hydrogen concentration in the tube Ref. [9,10]. One application is at the ends of the pressure tube where the yttrium can protect the rolled joints. An experiment to demonstrate the efficacy of the getter is in progress. After 200 days of testing the hydrogen distribution in the pressure tube at the rolled joint is following the prediction, (Fig. 6). If this behaviour continued for 30 years, calculations with ingress and allowing for gettering show that no hydrides would be present anywhere in the pressure tube, even in the rolled joint, during reactor operation.

2.2.2 Increasing Fracture Toughness

If a crack does initiate and propagate we need high fracture toughness to support the case for LBB. The fracture toughness of pressure tubes removed from power reactors is very variable. The conclusion from programs to understand this variability were that for high toughness and resistance to degradation by irradiation, we should minimize the concentration of the trace elements chlorine and phosphorus, and control the concentration of carbon Ref. [5,11,12]. Chlorine, present as a residue from the Kroll process, is kept to acceptable concentrations by evaporation during four melts. Phosphorus is present in the starting zircon sand and carbon is introduced in the refining process and from contaminated recycled material. Modification to the purification steps and control of recycled material minimize the phosphorus concentration and maintain the carbon concentration at innocuous values.

The success of the transfer of these developments to tube production is illustrated by comparing the distribution of fracture toughness, characterized by dJ/da, for tubes made from double melted material with that for the latest tubes made from selected sponge that was quadruple melted, Figure 7. This increased toughness will translate into greater critical crack lengths, even after much irradiation, and therefore contribute to improved confidence in LBB.

3. REDUCED OPERATING COSTS

In the early reactors major maintenance costs have been incurred from spacer displacement and delayed hydride cracking. This has necessitated excessive inspection programs that have been costly. For current reactors these problems should not recur and normal levels of inspection (regulatory standard only) will be necessary.

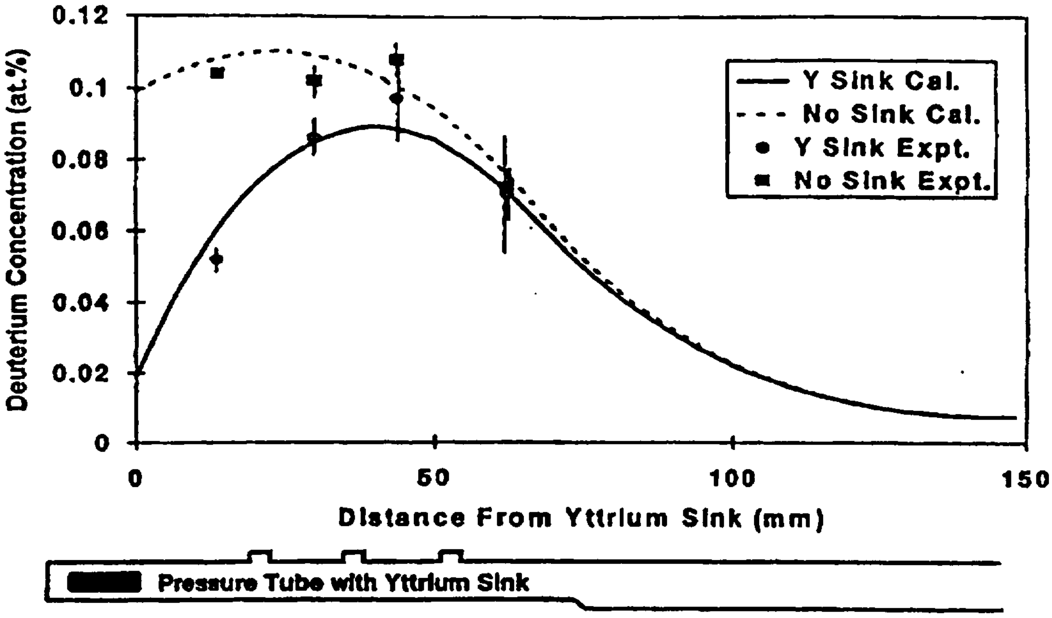


Figure 6 Effect of an Yttrium Inset on the Deuterium Distribution in the Pressure Tube at the Rolled Joint

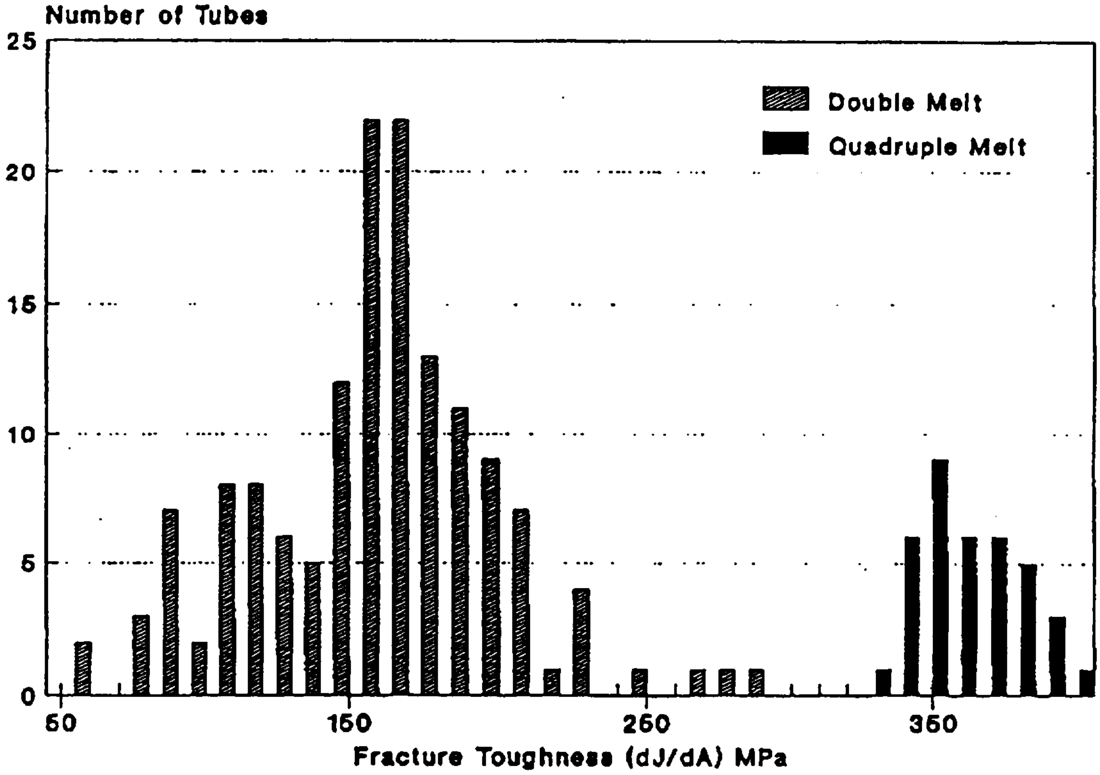


Figure 7 The Effect of Quadruple Melting on the Fracture Toughness of cw Zr-2.5% Nb

A design change to be implemented in the next series of reactors is a welded connection to replace the mechanical "graylok" metal ring seal joint between the end fitting and the feeders. The welded joint will eliminate leaks at such joints and the associated tritium concentrations with no cost penalty compared to the existing joint. The elimination of the large boss on the end fittings that is needed for a mechanical connection, also provides better access to the calandria face for maintenance.

Pieces of debris trapped between the elements of a fuel bundle can vibrate with the flow and erode the pressure tube, creating surface flaws until the debris is worn away. Such flaws if detected must be dispositioned by analysis for their potential to grow which usually requires characterization by detailed inspection. To avoid such flaws, the emphasis must be on the station to operate with debris-free systems.

4. INCREASED POWER OUTPUT

The maximum power output achieved in current CANDU reactors is about 6.5 MWe per channel. This is believed to be an optimum for the 100 mm diameter 6m long channel. To increase the output further would erode critical heat flux margins, increase the neutron flux and increase deformation, which is approximately proportional to flux. However, the overall power output of the reactor can be increased by changing the form factor with the use of enriched fuel so that the outer channels (i.e., those in the intermediate ring) operate with the same output as the inner channels, but none exceed the 6.5 MWt limit. Alternatively more channels can be added to the core.

5. INCREASED SAFETY MARGINS UNDER POSTULATED ACCIDENT CONDITIONS

Under postulated loss of coolant accident conditions, particularly when combined with loss of emergency coolant injection, the pressure tube will heat up to a temperature determined by the heat dissipation to the moderator through the calandria tube. The predicted consequences for such an accident are less severe if the radiative heat transfer from pressure tube to moderator through the calandria tube is increased and the maximum pressure tube temperature reduced. The absorptivity of the calandria tube can be increased three to four times by oxidation to achieve an appreciable increase in heat transfer. This development will almost certainly be applied to the next design of CANDU's.

For a postulated large LOCA plus loss of auxiliary power, the pressure tube may overheat and balloon into contact with the calandria tube. If the calandria tube overheats it may rupture. If high heat conductance results from the contact, dry out will likely occur on the outside surface of the calandria tube and consequently high calandria tube temperatures. For certain surface conditions, an optimum thermal conductance can be produced that achieves nucleate boiling on the outside surface of the calandria tube and keeps the calandria metal temperature low. This device is also being developed for use in future reactors.

6. FASTER RETUBING

Studies have shown that the vault preparation, tube removal and replacement, and recommissioning each contribute about 1/3 to retubing time.Ref. [13]. Channel removal and replacement can be facilitated by material handling in the vault through optimum placement of

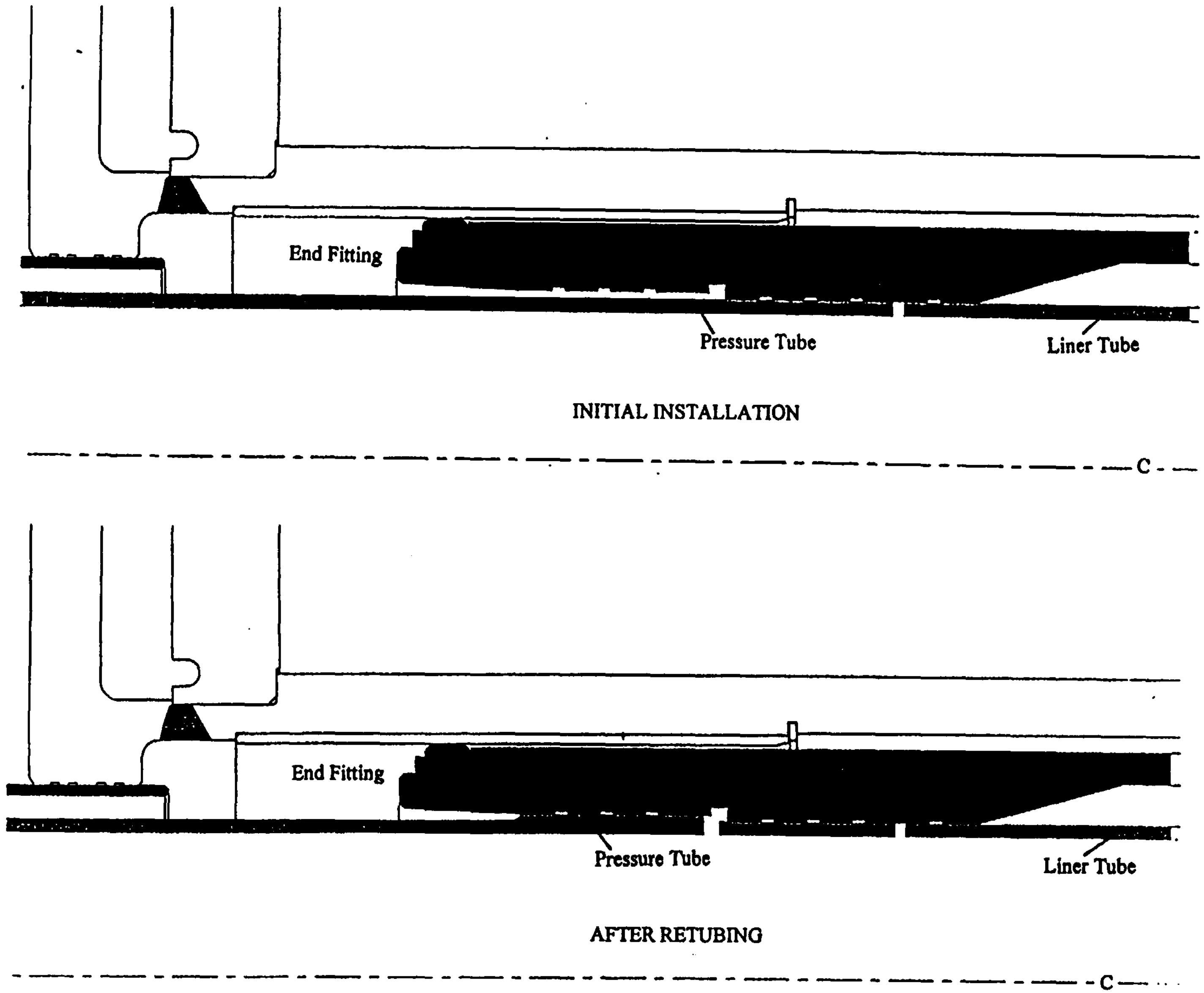


Figure 8 The Double Rolled Joint Design for the "One End Fitting in Place" Retubing Concept

penetrations. However, two channel design features slow up the retubing process. First is the need to reseal the gas annulus bellows and second is the need to position spacers accurately. Both of these activities are currently the subject of development which will enable faster retubing. Another feature that is likely to be incorporated in future designs is a double grooved joint in one end fitting which will enable one end fitting to be left in place during retubing to cut down on the number of components to be replaced, time and person–rem exposure associated with its removal. The design has the pressure tube rolled into the outboard grooves during the initial installation. When the tube has to be replaced the old tube is cut between the two sets of grooves and the new tube rolled into the inboard grooves and the previous rolled section left in place, Fig. 8.

7. CONCLUSION

The CANDU fuel channel is capable of considerable further development to increase life, lower maintenance increase safety margins and be efficiently replaced. This will contribute to the goal of a high capacity reactor operation for a minimum of 60 years with one channel retubing.

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METHODOLOGIES FOR ASSESSMENT OF THE SERVICE LIFE OF PRESSURE TUBES IN INDIAN PHWRs

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Abstract

For estimating safe service life of pressure tubes in Indian PHWRs, analytical methodologies have been developed to evaluate creep deformation, deuterium pick-up rate, blister growth at cold spot, and operating domain required for achieving leak-before-break. The paper provides an overview of these methodologies, and results of some studies carried out towards evolution of proposed fitness-for-service criteria for a pressure tube in contact with its calandria tube.

1. INTRODUCTION

At the current stage of the Indian nuclear power programme eight 200/220 MWe horizontal pressure tube type Pressurised Heavy Water Reactors (PHWRs) are under operation, and a programme of construction of additional PHWRs is under way. The schematic of a typical PHWR coolant channel is given in Fig. 1. The material of the pressure tubes of the first seven Indian PHWRs is 20% cold worked Zircaloy-2, and in these reactors the garter spring spacers are loose-fit around the pressure tube. Subsequently the pressure tube material has been changed to 20% cold worked Zr-2.5% Nb alloy and the garter spring spacers are made tight-fit around the pressure tubes. The pressure tubes are vulnerable to a number of known life limiting mechanisms primarily originating from creep & growth, hydrogen (the term 'hydrogen' used in this paper includes hydrogen equivalent of deuterium) pick-up, and fast neutron irradiation damage.

The loose-fit Garter Spring spacer (GS), used in the early generation Indian PHWRs, has been found to be susceptible to displacement from its installed position, mainly during the construction and commissioning stage of the reactor. This could result in occurence of an early Pressure Tube -Calandria Tube (PT-CT) contact, due to creep deformation of both pressure tube and calandria tube, in those channels where such displacement is significantly large. The failure of a pressure tube in the Canadian Pickering-2 reactor has been attributed to a mechanism initiating from this event [1]. The principal elements of this failure mechanism are: creep deformation of the coolant channel, hydrogen pick-up, growth of hydride blister at contact spot, cracking of blister, sub-critical crack propagation by Delayed Hydride Cracking (DHC), and catastrophic failure. This paper presents an overview of the methodologies developed in India towards modelling of the first three of the afore-mentioned elements of this failure mechanism, and their application for assessment of residual service life of channels with PT-CT contact.

Fast neutron irradiation and hydriding of pressure tubes leads to progressive embrittlement of the pressure tube material. It is necessary to operate the Primary Heat Transport System (PHTS) of the reactor in such a pressure-temperature domain which ensures Leak Before Break (LBB) for a given state of embrittlement of the pressure tube material. The methodology to establish such a domain is also covered in the paper.

2. DETERMINATION OF PRESSURE TUBE-CALANDRIA TUBE (PT-CT) CONTACT TIME DUE TO CREEP DEFORMATION

2.1. Computer Code SCAPCA

During operation of a pressure tube it undergoes dimensional changes due to stress-dependent creep and stress-independent growth. The principal manifestations of such dimensional changes are sag of the channel and PT-CT contact, axial elongation, and increase in tube diameter.

Using a structural mechanics approach, formulations have been developed to simulate the creep and growth behaviour of coolant channels [2]. These formulations are solved using numerical methods and are incorporated in a dedicated computer code SCAPCA (Static and Creep Analysis of Pressure tube and Calandria tube Assembly). Fig. 2 illustrates the analytical model incorporated in this code. Some features of this code are as follows:

a) Axial, hoop, radial and bending deformations have been modelled.

b) Dimensions, material properties, loads, neutron flux, and temperature can be varied along the length of the channel as well as with time.

c) Any number of garter spring spacers can be considered together with variations in their axial positions during different periods of time.

d) Sag profile of pressure tube and calandria tube can be determined prior to, as well as, beyond the occurence of PT-CT contact.

e) Spread of zone of contact between pressure tube and calandria tube is simulated.

2.2. Validation of SCAPCA

During In-Service Inspection (ISI) the radial gap between the pressure tube and calandria tube along the bottom of channel (referred to as `PT-CT gap') is measured along with the axial positions of garter spring spacers. Comparisons between measured PT-CT gap profiles and corresponding SCAPCA predictions based on measured positions of garter springs have been made for a large number of coolant channels of Indian PHWRs subjected to ISI so far. In order to define the quality of fit between the ISI and SCAPCA PT-CT gap profiles for these channels, the individual cases are classified as per the following four classes of fit:

Class-A: Excellent fit over the full span.
Class-B: Good fit; normal scatter in inputs can explain the variations.
Class-C: Fair fit; trends similar but magnitudes differ significantly.
Class-D: Poor fit; trends do not match.

Fig. 3 shows typical examples of different classes of fit. As indicated in the figure, for the 56 inspected channels for which PT-CT contact time is less than ten Full Power Years (FPYs) of reactor operation the percentage of fits in the A, B, C and D classes are 73%, 19%, 5% and 3% respectively. SCAPCA results are conservative with respect to ISI results in 91% of these cases. When the entire population of 144 inspected channels is considered the percentage of fits in the A, B, C and D classes are 43%, 36%, 10% and 11% respectively. It is noted that a good match (Class A & B fits) is obtained between SCAPCA estimates and ISI observations of bottom gap profile of a large majority of the channels subjected to ISI so far.

A parametric study was performed to study the effect of different variables on the creep-sag limited life of the coolant channel [3]. An important deduction from this study is that the bottom gap between the PT-CT as well as the PT-CT Creep Contact Time (CCT), is a strong function of the pressure tube longitudinal creep strain rate and is almost independent of the calandria tube creep strain rate. This implies that a validation of the code with respect to PT-CT gap profile is sufficient to confirm the applicability of the pressure tube longitudinal creep strain law forming a part of SCAPCA; therefore it also serves to validate SCAPCA results for CCT.

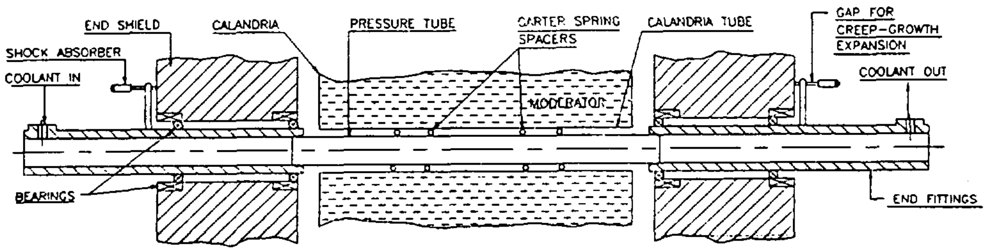

Fig. 1: SCHEMATIC OF COOLANT CHANNEL ASSEMBLY

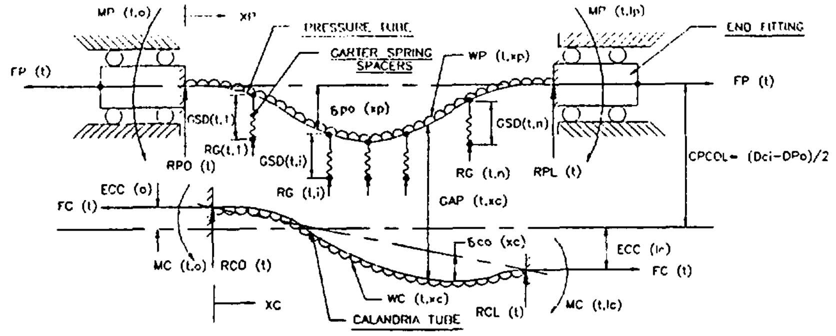

Fig. 2: IDEALISED MODEL OF COOLANT CHANNEL ASSEMBLY FOR CREEP GROWTH ANALYSIS

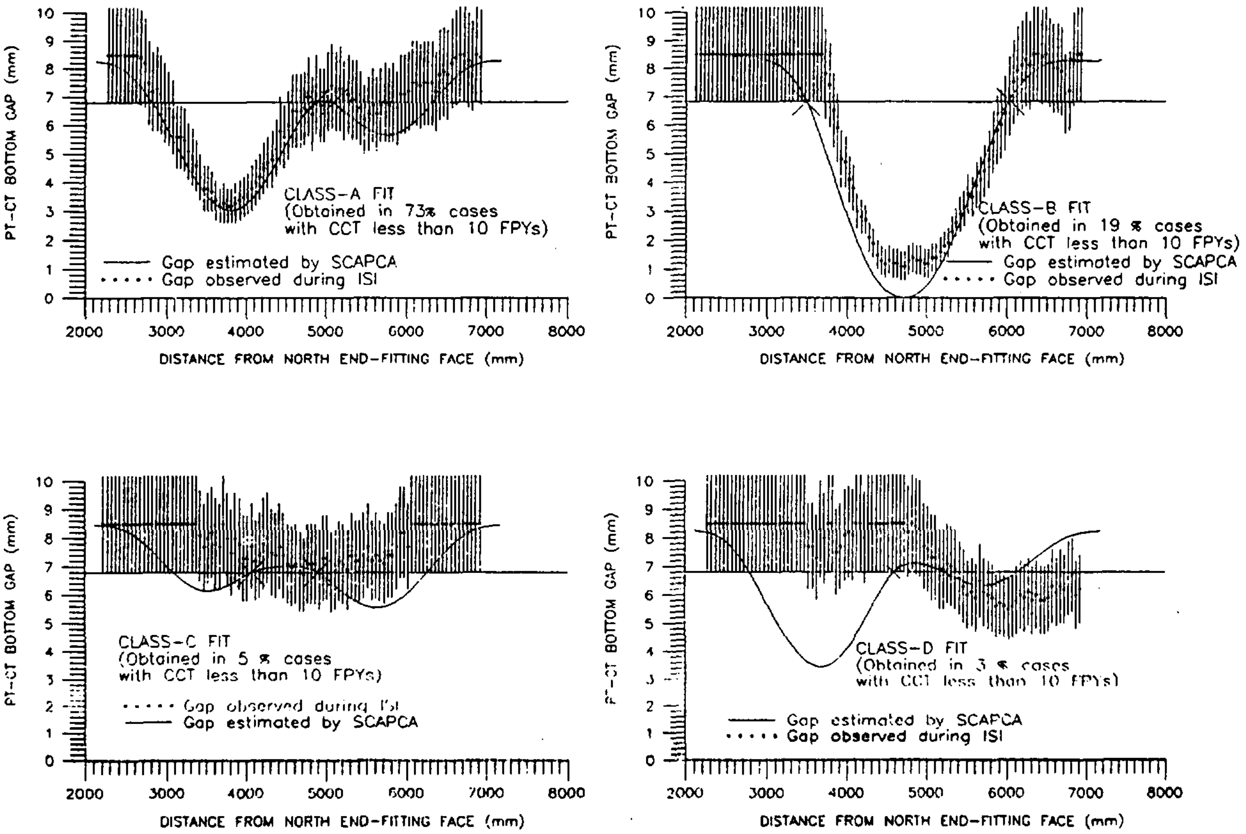

FIG. 3: SCAPCA-ISI GAP COMPARISON SHOWING DIFFERENT CLASSES OF FIT.

2.3. Some studies done using SCAPCA

2.3.1. Feasible domain of PT-CT contact locations

A study has been carried out to estimate the possible region of PT-CT contact by considering a large number of possible shifted positions of garter springs in a high flux coolant channel of a 220 MW(e) PHWR having two garter springs and Zircaloy-2 pressure tube, designated as reference channel case.

Fig. 4 gives a plot of CCT versus the contact location for all the cases yielding a CCT less than 10 FPYs. For a given value of CCT, the horizontal intercept with the envelope of the plotted points identifies the feasible domain of PT-CT contact locations for possible contacts occuring at that time. It has been separately established that spread of the arc of a previously occured contact would remain within the afore-mentioned feasible domain of contact at the given FPYs of operation. Results for channels having a flux distribution different from that taken for the reference case can be obtained by replacing contact time with an adjusted contact time equal to contact time multiplied by the ratio of average fast neutron flux in the given channel and that in the reference channel. Using this procedure some of the cases of MAPS-1 & 2 reactors coolant channels have been plotted in Fig. 4. These points lie within the domain of feasible PT-CT contact.

2.3.2. Simulation of in-service shift of garter spring spacers

On the basis of ISI carried out for pressure tubes of RAPS and MAPS reactors it has been noted that in some of the channels the loose-fit garter springs could have moved under service conditions. In one such channel R2-K07, ISI indicated the presence of only one garter spring just before the pressure tube was removed for Post-Irradiation Examination (PIE), which showed marks on the outer surface of the pressure tube indicating shift of the other garter spring over a period of time. SCAPCA analysis, assuming only one garter spring to be in position since the beginning of reactor operation, resulted in a gap profile which does not match at all with the ISI profile. If, however, it is assumed that the spacers were at their design location to begin with but subsequently moved out at different times to their ISI location, then an excellent correspondence is achieved between the PT-CT gap profiles obtained by ISI and SCAPCA. Fig. 5 shows the results of this study.

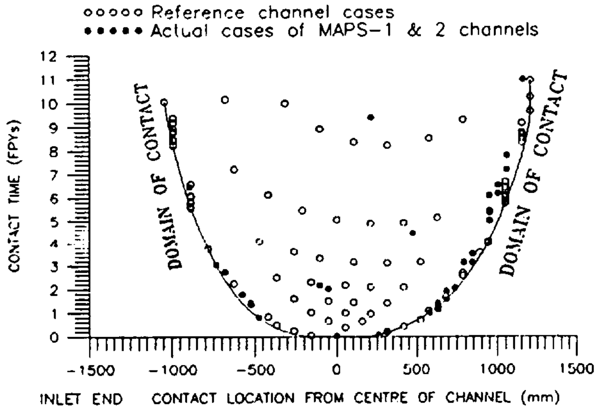

FIG. 4: DOMAIN OF PT-CT CONTACT LOCATION FOR DIFFERENT CONTACT TIME FOR MAXIMUM FLUX CHANNEL.

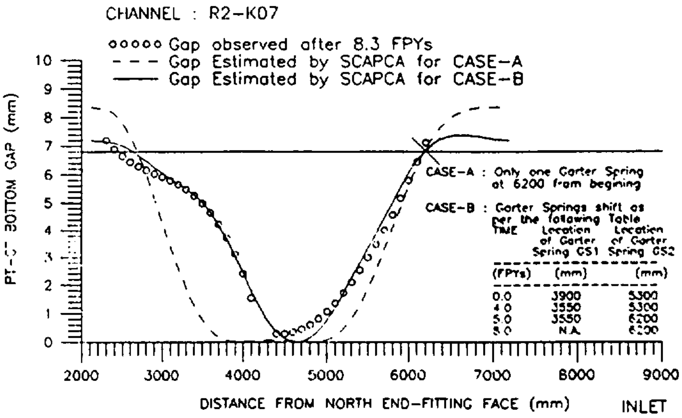

FIG. 5: SCAPCA-ISI GAP PROFILE COMPARISON SHOWING IMPROVEMENT IN FIT QUALITY WHEN IN-SERVICE GARTER SPRING SHIFT IS ACCOUNTED.

2.3.3. *Gap closing rate approach to determine CCT from ISI gap profile*

In order to obtain CCT from ISI gap profile, the first step is to determine the minimum loaded gap, i.e., minimum value of the PT-CT gap calculated after imposition of service loads and operating conditions on ISI PT-CT gap profile. In the next step the Gap Closing Rate (GCR) of the coolant channel assembly is determined. The CCT is then obtained by dividing the minimum of loaded gap by the GCR and adding the result to the FPYs of reactor operation at which the ISI was conducted. The GCR is estimated by SCAPCA at a value of the time variable equal to the ISI time. It has been determined that the GCR does not vary significantly during a further time span of 5 FPYs for ISI time exceeding about 3 FPYs. The previous displacement history of garter spring, if any, does not alter GCR. The GCR approach, therefore is very useful where nominal gap profile (without simulation of possible GS shift) obtained with SCAPCA does not match the ISI gap profile owing to GS shift.

2.4. Methodology for conservative assessment of creep contact time

As already indicated, in a large majority of safety significant cases where the CCT is less than 'current ISI time + 5 FPYs' - in other words where the CCT is not too far away in future and where the relative contribution of creep strains to deformed profile of the channel is significant - the PT-CT gap profile estimated using computer code SCAPCA has been found to be either fitting the ISI measured profile closely or has a minimum gap value lower than that observed during ISI. In view of this the first approach is to obtain CCT using SCAPCA with best available values of input data (including shift of GS). There could be a few exceptional cases where, due to possible GS shift or other reasons, the ISI minimum gap is lower than SCAPCA estimates. In such cases, to maintain conservatism, CCT based on GCR is taken as the basis for arriving at safety related decisions pertaining to a specific coolant channel.

3. ESTIMATION OF HYDROGEN PICK-UP

3.1. Computer Code HYCON-95

For an assessment of integrity of a pressure tube with PT-CT contact a knowledge of hydrogen concentration at the contact location is needed. For the purpose of estimation of hydrogen pick-up in Zircaloy-2 pressure tubes a model, called 'HYCON-95 model', has been developed and incorporated in a computer code, called HYCON-95.

Hydrogen pick-up in pressure tubes mainly results from internal corrosion. Hydrogen diffusion through rolled joints and through a depleted protective oxide film outside the pressure tube may contribute to additional build up of hydrogen concentration in the pressure tube. These two additional contributors may be ignored for carrying out assessment of a channel with PT-CT contact in reactors where the annulii between pressure tube and calandria tube is open to atmosphere. The objective of the HYCON-95 model is to conservatively estimate hydrogen concentration within the domain of axial positions where PT-CT contact is feasible within about ten FPYs of operation. For this purpose only water-side corrosion is assumed to be the source of hydrogen pick-up in the pressure tube.

HYCON-95 is a semi-empirical model derived from the available published models and measured oxidation and hydrogen pick-up data for Zircaloy-2 pressure tubes. This model derives its bases from the following inputs:

a) Hillner's model [4], which provides correlations for calculating oxide film thickness resulting from corrosion of Zircaloy-2 in autoclave environment.

b) UNI-3146 model [5], which was basically developed to explain corrosion and hydriding trends in pressure tubes of N-reactor.

c) Published literature on oxide thickness and deuterium pick-up measurements in pressure tubes removed from Pickering-1 & 2 Reactors [6].

d) PIE results for RAPS-2 pressure tubes O-11 and K-07 [7,8].

The code has the capability to consider the full operating history of the channel. For a typical high flux channel operating under nominal inlet and outlet temperature conditions, Fig. 6 shows computed estimate of hydrogen pick-up trends at the feasible contact locations nearest to the inlet and outlet ends as well as at the middle of the channel.

3.2. Comparison of HYCON-95 results with PIE Data

Fig. 7 shows the plot of data points from the average deuterium profiles of Pickering high flux channels reported along with 'HYCON-95' predictions with multiplication factors 0.6 and 1.5 on the best fit prediction. It is noted that the trend as well as the magnitudes of hydrogen pickup values (excluding the inlet side hump regions) for the five Pickering channels are adequately explained by the 'HYCON-95' model. In this figure, the reported deuterium pick-up values for Pickering-1 have been normalised for the effective full power years of Pickering-2. On the basis of this fit a multiplier of 1.5 on best-fit prediction of HYCON-95 was selected to yield a nominally conservative estimate of hydrogen pick-up to account for channel to channel variability owing to any factor not considered in the model, and uncertainties in input data.

Figures 8 and 9 show the predicted profile of hydrogen concentration and the measured values along the axial length of channels R2-O11 and R2-K07. The best estimate and conservative estimate are shown in the figures. The feasible zone of contact in 8.5 FPYs of operation in RAPS-2 is marked in the figures. It is seen that the HYCON predictions fit well the measured hydrogen data in the feasible contact zone.

3.3. Further work

It is planned to revise the HYCON-95 model to reduce the range between best-fit and nominally conservative estimate, and to improve the fit towards the outlet end. This exercise will be taken up when additional PIE data for oxide thickness profile and deuterium pick-up for Indian pressure tubes becomes available. These limitations do not affect the predictive capability of the model in the feasible zone of PT-CT contact.

4. MODELLING GROWTH OF HYDRIDE BLISTERS

4.1. Computer Codes for Blister Growth Modelling

Diffusion of hydrogen towards the cold contact spot on a pressure tube under PT-CT contact could lead to nucleation and growth of solid hydride blisters on the outside surface of the pressure tube at the contact location.

Determining the temperature field in the pressure tube wall at and near the cold spot is a necessary first step in the analysis for diffusion of hydrogen towards PT-CT contact spot. Results obtained from a 3D analysis performed using Finite Element Method to assess the distribution of temperature in the neighbourhood of PT-CT contact region [9] indicated that the temperature distribution is very close to axisymmetric. Thus a 2-D approach to solution of the problem, in most cases, would be sufficiently accurate. The temperature at cold spot formed in the event of PT-CT contact is governed by the contact condition which, in turn is mainly characterised by the thermal contact conductance, and cold spot diameter. As per studies reported in reference [10], the best estimates of contact conductance and cold spot diameter for Pickering 1&2 channels are found to be 0.01 W/mm^2-°C and 5 mm respectively. For the current studies values of these inputs have been assumed as 0.01 W/mm^2-°C (as in reference [10]) and 4 mm (20 percent lower contact spot diameter for 20 percent lower diameter channel) respectively. In those reactors where the

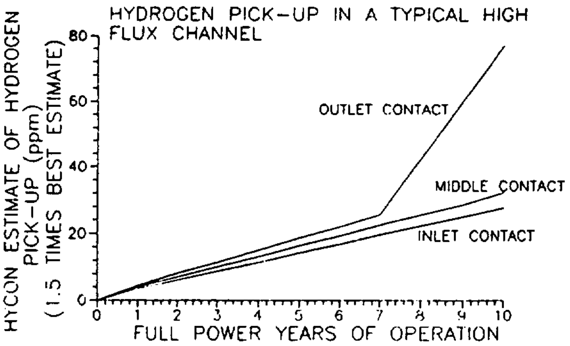

Fig. 6. ESTIMATE OF HYDROGEN PICK-UP USING HYCON CODE.

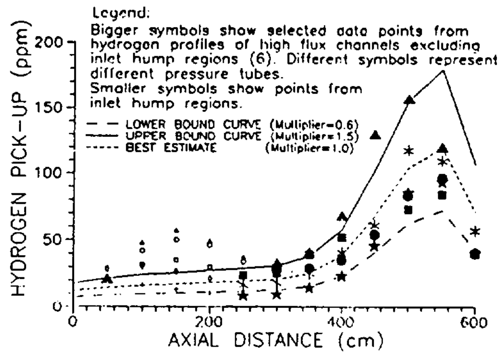

Fig. 7. PREDICTION OF HYDROGEN PICK-UP IN PICKERING CHANNELS USING HYCON CODE

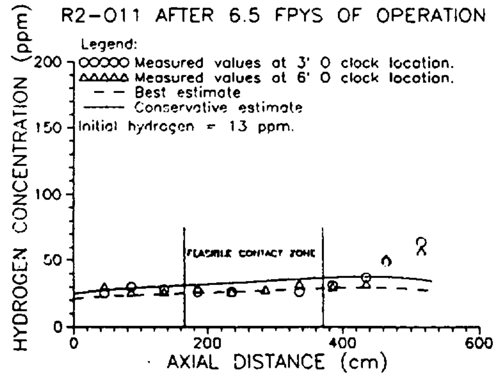

Fig. 8. ESTIMATE OF HYDROGEN IN R2-O11 USING HYCON CODE.

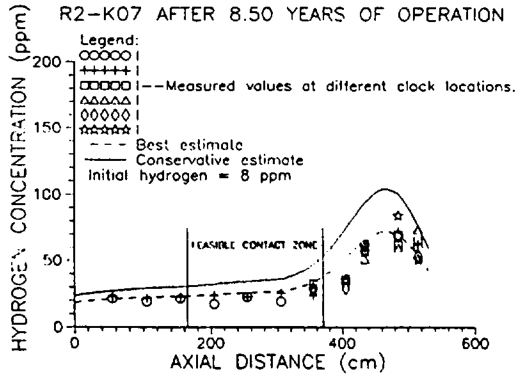

Fig. 9. ESTIMATE OF HYDROGEN IN R2-K07 USING HYCON CODE

annulus between the pressure tube and calandria tube is open to atmosphere, on account of oxide formation on the outside surface, as observed during PIE of R2-O11 and R2-K07 pressure tubes [7], and presence of dust within the calandria tube, as observed during the DRYVIS (DRY channel Visual Inspection System) inspection of R2-O11 calandria tube [11], the actual value of contact conductance could be lower than the afore-mentioned value. A smaller contact spot diameter leads to faster rate of growth of blister, once again leading to conservative results.

Modelling diffusion of hydrogen and growth of hydride blisters involves the solution of differential equation for diffusion of hydrogen under concentration as well as temperature gradients. The migration of hydrogen/deuterium towards the cold spot takes place under temperature gradient and dynamically changes concentration of hydrogen in the solution phase, thus altering the other driving force for diffusion towards or away from the cold spot. The latter is also altered as a result of thermal cycling. Three computer codes, BLIST1D, BLIST2Dand SPARSH, have been developed to simulate the phenomenon of hydrogen migration under the given temperature field and to solve the diffusion problem. BLIST1D is a one dimensional finite difference code, which has been benchmarked against the BLIST2D to obtain approximate but conservative assessments of blister growth. BLIST2D is a 2-D finite difference code which uses rigorous analytical approach to study the phenomenon of hydrogen migration. A third code, SPARSH, has been developed to facilitate hydrogen migration and blister growth assessment using 3D-FEM approach for some special studies. Some details about these codes and the modelling approach are provided in reference [12]. All the three codes can simulate hysteresis effect in dissolution-precipitation of hydrogen in zirconium alloys. Effect of thermal cycling on growth of blister and hydrogen concentration distribution near the cold spot can thus be accounted for. BLIST1D and BLIST2D are the working tools used for assessment of blister growth trends. In the current studies the deuterium picked up in service is substituted by its hydrogen equivalent. Since hydrogen diffuses at a rate faster than deuterium, this assumption introduces a fair amount of conservatism in the blister growth calculation.

4.2. Validation of BLIST1D and BLIST2D

Both these codes have been validated against published experimental results for changes in hydrogen concentration with time for a given temperature gradient [13].

Figs. 10 and 11 show the results of a study carried out to compare the results of BLIST1D and BLIST2D for blister growth assessment for contact at near-inlet and near-outlet feasible contact locations (1.65 m and 3.7 m from inlet end respectively) in a channel with nominal inlet and outlet temperature conditions (249 °C and 293 °C respectively), initial hydrogen concentration 30 ppm, and pick-up rate 3 ppm per year. It is noted that in this case results of BLIST1D are conservative. The computational time typically required to run a case with BLIST2D is large. This necessitates use of coarse mesh, leading to some irregular trend indicated by the BLIST2D curves. In these figures results of a study presented in reference [14] for a similar case are also indicated.

4.3. Further Work

While assuming the input data for blister growth studies, a cautious and conservative approach has been adopted. It is intended to trim the elements of over-conservatism later on the basis of measurements of hydrogen profile near the contact spot in some of the pressure tubes removed from Indian PHWR for PIE.

5. PROPOSED METHODOLOGY FOR SAFETY EVALUATION OF CONTACTING CHANNELS

5.1. Background

For any decision regarding the assessment of safety of a coolant channel under PT-CT contact, it is essential to have a quantitative basis which takes into consideration known channel-specific parameters such as initial hydrogen, hydrogen pick-up rate, contact time, contact condition, temperature at contact point and hot years of operation following contact.

Following the occurence of PT-CT contact, mechanism of growth of blister has two important distinctive stages:

(I) Initiation of blister growth
(ii) Cracking of a blister leading to DHC

Although the time duration between reaching of stage (ii) and the final failure may be several years, on account of limitations on the modelling of this phase of failure mechanism, as a measure of conservatism taking into account current state of knowledge, reaching stage (ii) is proposed to be treated as a criterion for rejection of channel. Appropriate cut-off level and margin of safety would then need to be imposed to ensure that effect of possible uncertainties in input data and approximations in modelling are adequately taken care of.

5.2. Limiting Blister Depth and Acceptable Blister Depth

The Critical Blister Depth (CBD) is expressed in terms of depth of an equivalent blister [10] which would crack at the pressure tube operating stress [15]. For the operating stress values in Zircaloy-2 and Zr-2.5% Nb pressure tubes of Indian PHWRs the lower bound CBD values derived from data given in reference [15] are 0.65 mm and 0.58 mm respectively.

Delayed Hydride Cracking (DHC) in Zirconium alloys can be initiated once the stress intensity factor for a given flaw and stress condition exceeds a critical value, known as KIH. The reported lower bound values of KIH for propagation of radial axial flaws in cold worked Zircaloy-2 and cold worked Zr-2.5% pressure tubes are 5 Mpa√m and 4.5 Mpa√m respectively [16]. For the sake of safety evaluation of a pressure tube it may be postulated that a blister, which is composed

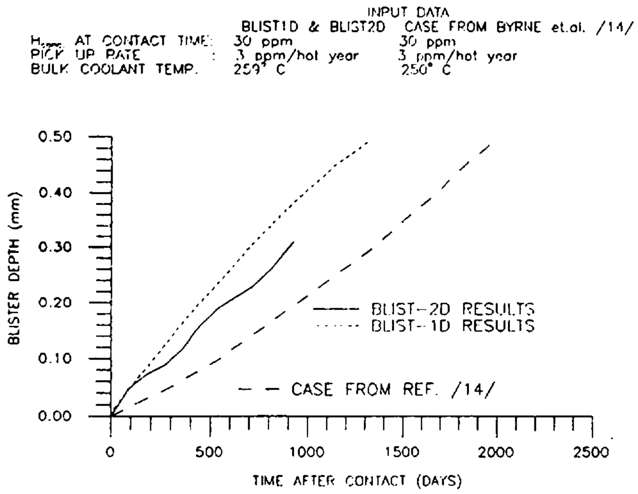

Fig. 10: COMPARISON OF BLISTER GROWTH RESULTS AT INLET CONTACT LOCATIONS OBTAINED FROM BLIST1D AND BLIST2D CODES

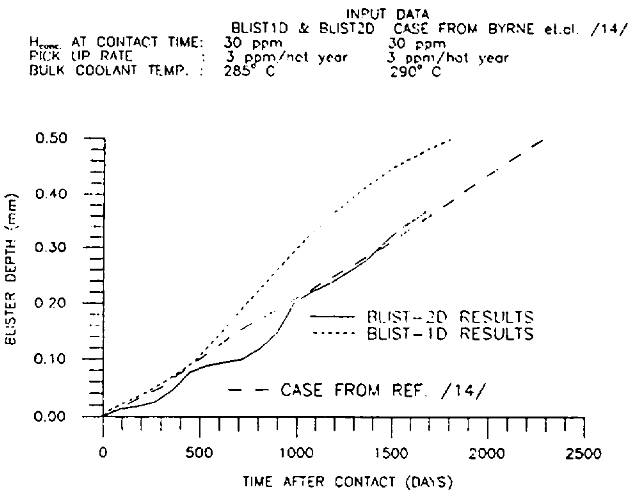

Fig. 11: COMPARISON OF BLISTER GROWTH RESULTS AT OUTLET CONTACT LOCATION OBTAINED FROM BLIST1D AND BLIST2D CODES

of a brittle phase, could crack at lower than CBD under some abnormal loading condition. To ensure that this does not initiate DHC propagation, the dimensions of a blister should not exceed those which could result in a flaw capable of growth by DHC. The values of this upper bound Blister Depth from DHC consideration (BD-DHC) for Zircaloy-2 pressure tubes for blister diameters of 4, 5 and 6 mm are computed as 0.87 mm, 0.70 mm and 0.66 mm respectively. The values of BD-DHC for Zr-2.5% Nb pressure tubes with 4, 5 and 6 mm diameter blisters are 0.30 mm, 0.29 mm and 0.29 mm respectively.

Limiting Blister Depth (LBD) is defined as the blister depth which is lower of the CBD and BD-DHC. Thus LBD has a value of 0.65 mm for Zircaloy-2 pressure tubes, and 0.29 mm for Zr-2.5% Nb pressure tubes (assuming blister diameter of 6 mm). It is proposed to impose a safety factor equal to three, on LBD so as to yield the value of Acceptable Blister Depth (ABD). For Zircaloy-2 and Zr-2.5% Nb pressure tubes this approach yields the values of ABD as 0.21 mm and 0.09 mm respectively.

5.3. Proposed methodology for assessment of fitness-for-service of a coolant channel with PT-CT contact

Keeping in view the various margins of conservatism built in the methodolgies for assessment of Creep Contact Time (CCT), hydrogen pick-up, blister growth, and Acceptable Blister Depth (ABD) together with the possible bands of uncertainties in some of the inputs used in the computations considering size of the available database of measured values, the following methodology has been proposed for assessing the fitness-for-service of a coolant channel with PT-CT contact:

A) Estimate nominally conservative Creep Contact Time (T) using SCAPCA based on both GS position data and PT-CT gap data obtained from ISI.

B) Estimate nominally conservative hydrogen Pick-Up Rate (p) at contact location after multiplying the HYCON best estimate by a factor to take into account channel to channel variability (1.5).

C) Using the values of T and p, determine blister depth (D) at the end of the evaluation period (additional time up to which the channel is sought to be operated) using one of the blister depth codes.

D) Using a value of contact time equal to f*t (f≤1, to take into account all perceived uncertainties in inputs and modelling, a value currently suggested is 0.8) and pick-up rate equal to K*p (K≥1, to take into account all perceived uncertainties in inputs and modelling, a typical value currently suggested is 1.5) determine blister depth (Dmax) at the end of the evaluation period.

E) The pressure tube is fit for continued service during the evaluation period if D≤ABD, and also Dmax≤LBD.

6. METHODOLOGY FOR ASSESSMENT OF LEAK-BEFORE-BREAK CAPABILITY OF PRESSURE TUBES

6.1. Introduction

During in-reactor service, Zircaloy-2 pressure tubes are subjected to embrittlement due to fast neutron irradiation and hydriding. This is evidenced by a reduction in its fracture toughness and Critical Crack Length (CCL) at lower temperatures. A continuous upward shift of the Ductile-Brittle Transition Temperature also occurs. An inside surface crack developed on the pressure tube (during its fabrication or operation) may grow (with velocity V_c) during reactor service due to (DHC). By the time the crack penetrates the wall to start leaking, it would have propagated to a length (L_o) dictated by the aspect ratio of the crack. The crack has to grow to a definite length (A_o) before it provides a detectable leak rate. Once the leak is detected, the operator needs certain time period (t) to identify the source of leak and to shut down the reactor and depressurise the PHT. During this period, the crack would continue to grow axially. In order to avoid catastrophic failure of the pressure tube it is necessary to contain the undetected crack length below the CCL at all operating conditions.

6.2. Determination of operating domain to achieve LBB

The fitness of pressure tube for service requires a demonstration of leak-before-break (LBB) capability. For pressure tubes of PHWRs, the LBB criterion is defined by the following expression:

$$A_o + L_o + 2\,V_c * t < A_{max}$$

where:

A_o is the through wall axial crack length which leads to minimum detectable leak rate.

L_o is the length of the crack on the inside surface of the tube just as the crack front reaches the outside surface of the tube.

V_c is the delayed hydride crack propagation velocity.

t is the operator response time.

A_{max} is the maximum allowable crack length.

To achieve LBB, 'A_{max}' must be less than the minimum of the values of critical crack length of the pressure tube under all possible conditions of pressure and temperature during normal operation, hot shutdown, depressurisation and start up.

This philosophy decides the upper bound to pressure - temperature domain. The lower bound is determined by the saturation curve.

Using the afore-mentioned criterion for LBB, a computer code 'CEAL' (Code for the Estimation of Assurance of Leak before break) has been developed. The result of a typical case study for determination of an operating pressure and temperature domain required for LBB of irradiated Zircaloy-2 pressure tube is presented in Fig. 12.

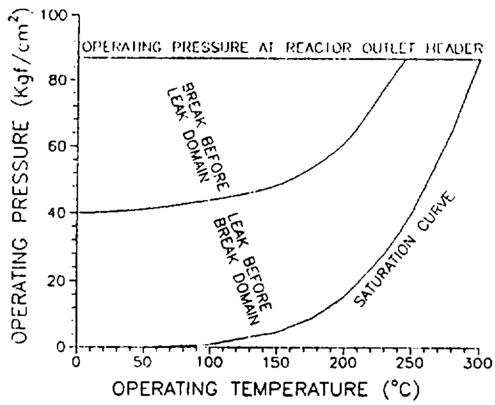

Fig. 12. RESULTS OF ANALYTICAL STUDY TO DETERMINE OPERATING PRESSURE-TEMPERATURE DOMAIN FOR ZIRCALOY-2 PRESSURE TUBE.

7. SUMMARY

Methodologies have been developed in India to model and analyse the major life limiting mechanisms for pressure tubes. Computer codes have been developed for determination of creep & growth related dimensional changes of coolant channels, deuterium pick-up rate in Zircaloy-2 pressure tubes, blister growth following occurence of pressure tube-calandria tube contact, and operating domain to achieve leak-before-break of pressure tubes. These codes have been compared with, and validated with reference to, the results of in-service inspection data and post irradiation examination data. A methodology to assess fitness for service for channels with pressure tube-calandria tube contact has been proposed. Future work will involve fine tuning of the computer codes on the basis of data obtained from additional ISI and PIE performed on a large number of pressure tubes of Indian PHWRs.

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FUEL AND FUEL CYCLE OPTIONS

(Session VII)

Chairman

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A PHWR WITH SLIGHTLY ENRICHED URANIUM ABOUT THE FIRST CORE

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Abstract

Many different studies have been performed in Argentina regarding the use of slightly enriched uranium in the PHWR nuclear plants. These referred mainly to operating plants so that a transition had to be considered from the present natural uranium fuel cycle to the slightly enriched one. In this analysis, technical and economical arguments are presented which favor the use of a natural uranium initial core. The levelized fuel costs are shown to be practically insensitive to the first core and a fast transition is more influential than an initially enriched core.

1. INTRODUCTION

The use of slightly enriched uranium (SEU) in a PHWR has been the subject of a number of analysis, showing the advantages in generating cost and fissile material savings.

The use of this fuel cycle in presently operating reactors is made through a transition from the equilibrium natural uranium (NU) core to an homogeneous SEU core.

Some studies report also different optimizations of the plant in order to adapt the design to the benefits of this new cycle from the start-up. Nevertheless a series of considerations of technical and economical nature favor the solution of starting the PHWR plant with natural uranium and switching afterwards to a SEU cycle by means of an adequate transition.

2. THE START-UP CORE

For a PHWR reactor which operates under on-power refuelling, the initial core is a great singularity from the point of view of the excess reactivity. In fact, the multiplication is of a magnitude never repeated along the plant's life. This means that during the start-up and a long initial period the design must assure the control of this reactivity in all possible situations.

In this sense, the batch refuelling of a LWR makes the first core a less singular situation.

The PHWR plant operates during the initial phase with a high concentration of neutron poison either in the form of boron, depleted uranium, etc. Of course this implies a loss of burnup associated to the first core or even further. Thus, it is conceptually contradictory to enrich the initial core and at the same time increase the poison content to control a higher reactivity.

Moreover, this poisoning in the SEU case will last a longer period because the fuel must reach a higher burnup untill the equilibrium reactivity excess is reached. This excess is similar to the natural uranium case because the tipical operational transients, like xenon recovering require similar reactivity reserves both in the NU and SEU cases.

For the argentine NPPs, the use of 0.85% enrichment in the fuel adds approximately 6000 pcm to the initial core reactivity. To give an idea of what this value means, in the case of Atucha 1 NPP (350Mwe), all the safety rods control a reactivity of around 8400 pcm in the cold core.

From a licencing perspective, the NU initial core solution is preferred, because it is the normal and well established procedure for a PHWR, and the core is much less reactive in the long transition from start-up to the equilibrium situation.

In other words, what is here suggested is that a NU initial core is a solution conciliating technical and economical aspects.

3. GENERATING COSTS

In the following, an economic evaluation is presented for different alternatives of NPP start-up.

The methodology used to calculate generating costs is that of levelized lifetime costs at constant money which is considered adequate for this comparison.

3.1 CANDU reactor

In this case levelized generating costs were calculated for two identical plants, 881 Mwe each, hipotetically started-up with a NU core and a SEU core respectively. The enrichment used was 0.9% U235.

The data for cost evaluation were taken from reference [1]. The prices adopted for uranium and conversion were 40 U$S/kg and 8 U$S/kg as the latest information about fuel cycle costs was not available.

The commissioning date was fixed in year 2000, with a plant lifetime of 30 years. The discount rate adopted was 5% and the effect of adopting 10% was included in the analysis. All costs were discounted at the commissioning date.

As both plants are considered identical, without design changes related to the SEU cycle, capital costs are the same and amount to 23.6 mills/kwh. The corresponding value in [1] is 22.8 mills/kwh, but the load factors used in the calculation are referred ambiguously.

The initial and final cores are included in the fuelling costs, so the comparison is made on this component only.

Tables I and II show the results of levelized fuelling costs for 5% and 10% discount rate respectively.

The different cases correspond to the following description:

(0) : Reference case. The first core and the following are NU with the indicated exit burnups,

-1st. core: 6375 Mwd/tonU
-2nd. core: 7500 Mwd/tonU (equilibrium)

(1) : The first core and the following are SEU with 0.9% enrichment. The exit burnups are,

-1st. core: 10000 Mwd/tonU
-2nd. core 14000 Mwd/tonU (equilibrium)

(1-a) : Same as (1) but with a different burnup sequence ,

-1st. core: 9000 Mwd/tonU
-2nd. core: 12000 Mwd/tonU
-3rd. core: 14000 Mwd/tonU (equilibrium)

(2) : The first core is NU and the following are SEU with 0.9% enrichment and the indicated exit burnups,

-1st. core: 6375 Mwd/tonU
-2nd. core: 12000 Mwd/tonU
-3rd. core: 14000 Mwd/tonU (equilibrium)

(2-a) : Same as (2) but with a different burnup sequence,

-1st. core: 6375 Mwd/tonU
-2nd. core: 9000 Mwd/tonU
-3rd. core: 11000 Mwd/tonU
-4th. core : 14000 Mwd/tonU (equilibrium)

Cases (1) and (2) represent a fast transition to the equilibrium SEU core and cases (1-a) and (2-a) a slower transition.

The burnup values are reasonable for the present calculations but of course they are only indicative and depend on the particular fuel management adopted in each case.

TABLE I : LEVELIZED FUELLING COSTS. 5% DISCOUNT RATE

CASE	FUELLING COST (mills/kwh)	SAVING (%)
(0) NU	2.13	
(1) SEU	1.69	20
(1-a) SEU	1.74	18
(2) NU-SEU	1.71	20
(2-a) NU-SEU	1.80	15

The results show that fuelling costs are reduced in the range 18%-20% when the SEU fuel cycle replaces the NU cycle. When the transition to equilibrium is delayed in time costs increase.

Levelized costs practically do not change if the start-up core is NU or SEU: cases (1), (2), provided that the transition to equilibrium is fast. When the transition is slow the difference increases: cases (1-a), (2-a).

Results are also sensitive to the discount rate. The benefits of the SEU cycle decrease with increasing discount rate.

The SEU cycle is more sensitive to discount rate than NU. This is mainly due to the fact that for NU the back-end costs are relatively more important and so there is a compensation between the penalization of front-end costs and the reduction of back-end cost contribution when a higher rate is considered. The relative weight of the back-end component is 27% for 5% discount rate and it drops to 15% for 10% discount rate.

In the SEU cases the figures corresponding to the back-end component are 18% and 10% respectively. This reduced importance of the back-end inhibites the above mentioned compensation.

Of course the difference in cost between a NU and a SEU initial core (14 million U$S and 20 million U$S respectively), has a decisive influence in the results.

TABLE I I : LEVELIZED FUELLING COSTS. 10% DISCOUNT RATE

CASE	FUELLING COST (mills/kwh)	SAVING (%)
(0) NU	2.10	
(1) SEU	1.78	15
(1-a) SEU	1.84	12
(2) NU-SEU	1.80	14
(2-a) NU-SEU	1.91	9

3.2 Atucha 2 reactor

Table 3 shows results for the argentine Atucha 2 NPP. This is a 750 Mwe PHWR, currently under construction. The enrichment in this case was 0.85% and local fuel cycle costs were considered except for the back-end data which were interpolated among those reported in reference [2] for once-through cycles.

The description of the cases studied is as follows:

(0) : Reference case. The first core and the following are NU with the indicated exit burnups,

-1st. core: 6375 Mwd/tonU
-2nd. core: 7500 Mwd/tonU (equilibrium)

(1) : The first core and the following are SEU with 0.85% enrichment. The exit burnups are,

-1st. core: 9000 Mwd/tonU
-2nd. core 12000 Mwd/tonU (equilibrium)

(1-a) : Same as (1) but with a different burnup sequence ,

-1st. core: 8000 Mwd/tonU
-2nd. core: 10000 Mwd/tonU
-3rd. core: 12000 Mwd/tonU (equilibrium)

(2) : The first core is NU and the following are SEU with 0.85% enrichment and the indicated exit burnups,

-1st. core: 6375 Mwd/tonU
-2nd. core: 10000 Mwd/tonU
-3rd. core: 12000 Mwd/tonU (equilibrium)

(2-a) : Same as (2) but with a different burnup sequence,

-1st. core: 6375 Mwd/tonU
-2nd. core: 8000 Mwd/tonU
-3rd. core: 10000 Mwd/tonU
-4th. core: 12000 Mwd/tonU (equilibrium)

Again, cases (1) and (2) represent a fast transition to the equilibrium SEU core and cases (1-a) and (2-a) a slower transition. The burnup values are, as before, only indicative.

Table III resumes the principal results.

TABLE I I I: LEVELIZED FUELLING COSTS. 5% DISCOUNT RATE

CASE	FUELLING COST (mills/kwh)	SAVING (%)
(0) NU	6.73	
(1) SEU	4.91	27
(1-a) SEU	5.03	25
(2) NU-SEU	5.04	25
(2-a) NU-SEU	5.18	23

The general behaviour of costs is similar to the preceeding case but here the results show less sensitivity to the transition.

It is very clear that in a case with higher fuelling costs, the benefits of the SEU cycle are bigger, specially reminding that in this case the results correspond to a lower enrichment: 0.85% U235.

4. CONCLUSIONS

Many different studies have been performed in Argentina regarding the use of slightly enriched uranium in the PHWR nuclear plants. These referred mainly to operating plants so that a transition had to be considered from the present natural uranium fuel cycle to the slightly enriched one. Little has been said about the start-up core of a new plant.

In this analysis, technical and economical arguments are presented which favor the use of a natural uranium initial core.

From a technical point of view, we can say that for a PHWR, with on power refuelling, the initial core is a singularity in the plant's life. During the long start-up period a very high excess reactivity has to be controlled and this situation is never repeated.

The plant is thus operated during a long period with a high poison content which of course produces a reduced mean burnup of the first core. In the SEU case the poisoned start-up core lasts a longer period because the fuel must achieve aproximately twice the burnup of the NU case (for 0.85% U235 enrichment), before the reactivity excess reaches the standard equilibrium value.

From a licencing point of view, the NU initial core solution is preferred because the reactor is much less reactive during the long transition from the start-up to the equilibrium situation.

Moreover, the economical side of the problem does not favor an initial SEU core.

The levelized fuel costs are shown to be practically insensitive to the first core and a fast transition is more influential than an initially enriched core.

In our case where the sources of investment are scarce, the different cost between a NU and SEU initial core is relevant.

A more flexible fuel design plays an important role in speeding-up this transition and also in reaching high load factors which are of mayor importance in the economics of the plant.

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SLIGHTLY ENRICHED URANIUM FUEL FOR A PHWR

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Abstract

An improved fuel element design for a PHWR using slightly enriched uranium fuel is presented. It mantains the general geometric disposition of the currently used in the argentine NPP's reactors, replacing the outer ring of rods by rods containing annular pellets. Power density reduction is achieved with modest burnup losses and the void volume in the pellets can be used to balance these two opposite effects. The results show that with this new design, the fuel can be operated at higher powers without violating thermohydraulic limits and this means an improvement in fuel management flexibility, particularly in the transition from natural uranium to slightly enriched uranium cycle.

1. INTRODUCTION

It is widely known that the use of slightly enriched uranium (SEU) in PHWR reactors, originally designed to operate with natural uranium (NU), presents economic advantages derived from the fact that less uranium is required for producing the same amount of energy. Several studies related with this alternative fuel cycle in our nuclear plants have been performed and recently a series of SEU fuel elements began its irradiation in Atucha 1 NPP. The enrichment adopted is 0.85% U235. This value is high enough to enable doubling the burnup in the equilibrium core, (roughly from 6000 Mwd/tonU to 11400 Mwd/tonU), and sufficiently low to admit its use with only minor alterations in the fuel management scheme, using the same fuel element and avoiding hardware modifications in the plant.

It can be added that even if the enrichment adopted is well below the theoretical optimum of 1.2% U235, there is little incentive to go further, due mainly to the fact that econonomical savings decrease exponentially with enrichment and for this particular case almost 70% of the maximum attainable saving is reached with 0.85% U235. Furthemore, higher fissile content would imply costly structural changes in the installation.

2. THE FUEL ROLE IN SEU

Extensive studies have been performed in Canada, related with the use of SEU in Candu reactors even if irradiations in power plants have not yet been initiated.

Also KWU identified this fuel cycle as an interesting alternative for PHWRs in the first phases of the Atucha 1 project. Moreover, Germany established a precedent with the conversion of the 57 Mwe MZFR reactor from NU to SEU [1]. This PHWR reactor operated around 10 full power years with enriched fuel: 0.85% U235 first and 1% U235 later. In this period, the fuel exit burnup increased gradually from approximately 7000 Mwd/tonU to 12000 Mwd/tonU in the first step and finally to 16000 Mwd/tonU.

In Argentina, the interest in this fuel cycle began as said, early at the time of the first NPP project. In 1985 twelve SEU fuel assemblies were fabricated for Atucha 1 and recently (January 1995) their introduction in the core was initiated. The fuel design considered up to now is almost identical to the original NU fuel assembly.

It is clear that the fuel plays a principal role in the implementation of the SEU cycle. Once the equilibrium core is reached the fuel exit burnup and the burnups at which shuffling operations are performed are notably higher. Even in the long transition from the natural to the enriched core, the burnup increases steadily, not only for the SEU but also for the NU fuel.

A more flexible fuel, resistant to PCI defects in this extended burnup range is of great significance.

2.1 Use of annular pellets

In [2] a revision was made of the proposed improvements in order to solve the problems related with burnup extension, going from simple solutions as increased gaps, shorter pellets, to more complex ones as graded enrichment or the CANFLEX design.

Here we analyze a modification consisting in the use of annular pellets in the outer ring of the cluster. This design produces several performance benefits. The improvement achieved depends on the void volume in the pellets which at the same time represents a certain burnup decrese. These parameters (power ratios and burnup loss) are quantified for the Atucha I and Embalse NPPs even if our attention is focused mainly on the Atucha case, because of its more inmediate importance.

The particular fuel management scheme used in this plant is broadly as follows:

If the core is imaginarily divided in tree concentric "zones": central, middle and outer, we can say that the fresh fuel is introduced in the middle zone, it is shuffled radially to the center when it has reached a mean burnup of approximately 2500-3000 Mwd/tonU and in the last step it is moved towards the outer zone at burnups in the order of 5000 Mwd/tonU. In this location it reaches the exit burnup and from here it is removed from the reactor.

The general outline is the same for the SEU case, except that the "zones" are different and the shuffling operations are performed at higher (almost doubled) burnups.

The movement from the middle to the central zone, with its abrupt increase in power density is the most delicate related to fuel performance. Also other non-stationary maneuvres like starts-up and power cycles affect the fuel behaviour and favours PCI defects.

In the SEU cycle, these power ramps occur at higher burnups, increasing fuel defects risks.

The use of annular pellets in the outer ring of the fuel assembly, reduces the temperature in the highly rated fuel, reducing the fission gas release, the internal pressure and providing additional space for pellet expansion. The resulting power redistribution in the cluster improves the maximum to average bundle power ratio (peaking factor), meaning that the fuel assembly can be operated at higher power without decreasing the margin to design limits.

3. RESULTS

In Fig. 1, a cross section of the fuel elements used in the two argentine NPPs is shown: (a) the well known Candu 37 rods fuel element in Embalse NPP, and (b) the Atucha 1 cluster with 36 fuel rods and a structural rod in the outer ring.

FIGURE 1

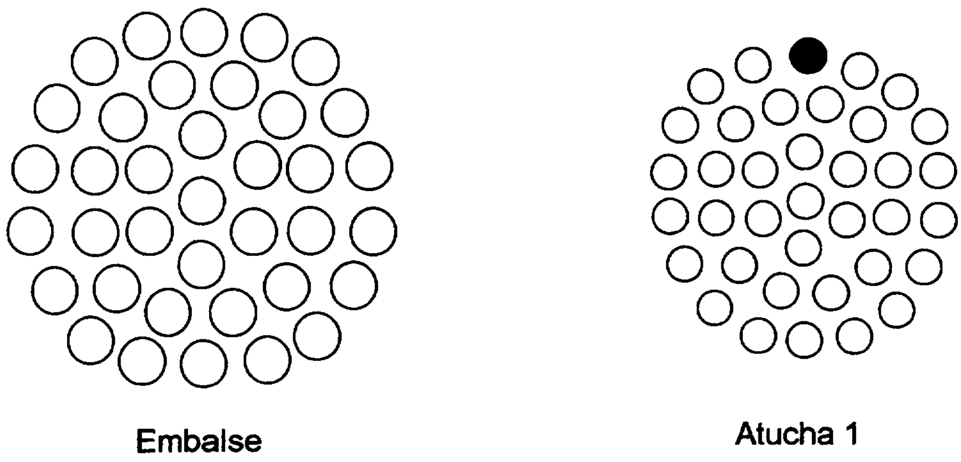

Fig. 2 resumes the relative power in the outer ring of rods in the Embalse cell, when natural uranium and 0.85% U235 are used as fuel in the current bundle. The two additional curves correspond to the use of annular pellets in the outer ring with a central void region corresponding to 6% and 11% of the pellet volume. The calculations were performed with WIMS code [3].

The improvement in the overpower of the outer rods is evident. The maximum value of the curves or peaking factor is reduced when annular pellets are used. This implies that for the same bundle power, the linear power density in the most exposed rod diminishes.

In Table I the behaviour of the peaking factor is given for both NPPs. Increasing the void volume inside the pellet, the peaking factor improves, but the UO2 loss produces a burnup reduction which imposes a compromise between the two parameters.

Using voids in a range from 6% to 11% of pellet volume, reductions in peaking factors of 1.8% to 4.4% for Embalse and 2.7% to 5.5% for Atucha 1 can be achieved. This comparison is made with respect to the actual design with natural uranium fuel. If it is made with respect to slightly enriched fuel the figures improve.

FIGURE 2

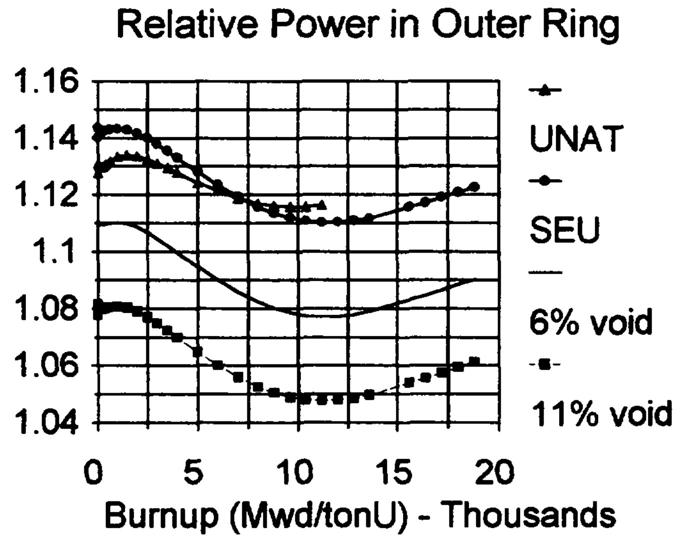

TABLE I: PEAKING FACTORS FOR EMBALSE AND ATUCHA 1

Natural uranium	0.85% U235	0.85% U235 11% void	0.85% U235 6% void
EMBALSE			
1.13	1.14	1.08	1.11
ATUCHA1			
1.10	1.11	1.04	1.07

From this point of view, enriched fuel with the new design performs better than the actual fuel with natural uranium.

The burnup losses have been estimated between 140 and 250 Mwd/tonU for annular pellets with 6% and 11% void volumes in the Embalse case. This is very reasonable keeping in mind that this low enrichment would allow a burnup increase from 7500 to 12500 Mwd/tonU.

For Atucha 1 the estimations are respectively 80 and 160 Mwd/tonU. Again these values are small compared to the burnup increase produced by SEU. In this case reactor calculations confirmed the preliminary estimations on burnup losses [4]. The overall reactor calculations were performed with the PUMA code [5] and some results are given in Table II.

TABLE II : RESULTS FOR ATUCHA 1

	Natural uranium	0.85% U235	0.85% U235 11% void	0.85% U235 6% void
Exit burnup (Mwd/tonU)	6000	11140	10980	11060
Maximum channel power (Mw)	7.34	6.79	6.80	6.79
Maximum linear power (w/cm)	509	451	424	436

The fuel management schemes used for the equilibrium core are different for the natural uranium and enriched fuel. In the enriched case the fuel enters the core more peripherically and of course is shuffled towards the center at higher burnups.

The maximum channel power and maximum linear power which in the natural uranium case are found in the central core zone are shifted considerably towards the periphery due to the flattening of the power distribution produced by SEU.

The results confirm a decrease of 3.3% in the maximum linear power density for 6% volume void and 6% decrease for 11% volume void.

4. CONCLUSIONS

We have analyzed a simple modification of the PHWR fuel element in order to improve its performance in the case of SEU fuel and its related burnup extension. The new design replaces the outer ring pellets by annular ones.

This modification allows to operate the fuel a higher power rate than the actual natural uranium bundle. The allowed increase ranges between 1.8% and 4.4% for pellet holes equivalent to 6% and 11% of pellet volume in the Embalse NPP case. This figures change to 2.7% to 5.5% for the same relative void volumes in the case of the Atucha 1 NPP.

The corresponding burnup losses are small in the context of the SEU cycle values.

There are other solutions which improve even more drastically the peaking factor in the fuel as for example the CANFLEX design or the graded enrichment. Nevertheless, the proposed solution has the advantage of being easy to implement in our case, being good enough for the burnups involved in the SEU cycle.

This design improves the fuel behaviour with respect to the burnup extension derived from the slight enrichment and it is also interesting in case an overall power upgrade is considered.

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SYNERGISTIC FUEL CYCLES OF THE FUTURE

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Abstract

Good neutron economy is the basis of the fuel cycle flexibility in the CANDU reactor. This paper describes the fuel cycle options available to the CANDU owner with special emphasis on resource conservation and waste management.

CANDU fuel cycles with low initial fissile content operate with relatively high conversion ratio. The natural uranium cycle provides over 55 % of energy from the plutonium that is created during fuel life. Resource utilization is over 7 MWd/kg NU. This can be improved by slight enrichment (between 0.9 and 1.2 wt % U235) of the fuel. Resource utilization increases to 11 MWd/kg NU with the Slightly Enriched Uranium cycle. Thorium based cycles in CANDU operate at near-breeder efficiency. They provide attractive options when used with natural uranium or separated (reactor grade and weapons grade) plutonium as driver fuels. In the latter case, the energy from the U233 plus the initial plutonium content amounts to 3.4 GW(th).d/kg Pu-fissile. The same utilization is expected from the use of FBR plutonium in a CANDU thorium cycle. Extension of natural resource is achieved by the use of spent fuels in CANDU. The LWR/CANDU Tandem cycle leads to an additional 77 % of energy through the use of reprocessed LWR fuel (which has a fissile content of 1.6 wt %) in CANDU. Dry reprocessing of LWR fuel with the OREOX process (a more safeguardable alternative to the PUREX process) provides an additional 50 % energy. Uranium recovered (RU) from separation of plutonium contained in spent LWR fuel provides an additional 15 MWd/kg RU.

CANDU's low fissile requirement provides the possibility, through the use of non-fertile targets, of extracting energy from the minor actinides contained in spent fuel.

In addition to the resource utilization advantage described above, there is a corresponding reduction in waste arisings with such cycles. This is especially significant when separated plutonium is available as a fissile resource.

INTRODUCTION

The neutron economy of the CANDU reactor is a result of design features that minimise parasitic neutron absorption. The use of heavy water for the moderator and the coolant limits their parasitic load to 1.5 % (15 milli-k). On-power refuelling avoids the use of burnable poisons for reactivity suppression. Such features limit the fissile inventory in CANDU cores. To appreciate the extent of this advantage, the fissile inventory of 1 GW(e) FBR, LWR, CANDU (Natural Uranium) and the CANDU AB that burns the transuranics in sterile fuel is compared in Table 1. As expected, lower fissile inventory leads to a higher conversion ratio for the production of fissile material from fertile material.

Table 1

Fissile Inventory (te) of 1 GW(e) Cores

FBR	PWR	CANDU	CANDU AB
3 to 4	2 to 3	1	0.05

The ability of CANDU to burn fuel with low fissile content leads to resource extension by the recycling of spent LWR fuel. In addition, it creates a significant fissile resource by raising the conversion ratio. This is especially so for fuel cycles that use thorium as the fertile material. A variety of innovative fuel cycles, some of which illustrate the synergy between CANDU and other systems such as FBR and LWR are described below.

URANIUM BASED CYCLES

Natural Uranium

Currently operating CANDU reactors use a once-through natural uranium cycle. The low fissile content of natural uranium leads to a conversion ratio around 0.8. This provides a plutonium production rate that almost compensates for the depletion of U235. The reason for the reactivity drop with irradiation in this fuel cycle is the buildup of fission product absorbers. The fissile content of the fuel on exit (at about 7 MWd/kgU) is almost the same as in the fresh fuel. This high rate of plutonium production leads to a large (> 55 %) energy contribution by plutonium. At exit the energy contributed by plutonium is > 70 %. Resource consumption in this cycle is 157 te/GW(e).y, (Table 2). Waste arisings are correspondingly the same.

Slight Uranium Enrichment

The large (> 70 %) contribution to energy by plutonium at the end of fuel life points to the benefits of extending the fuel life without reducing the conversion ratio. Slight enrichment of the fuel extends the fuel life but reduces the conversion ratio. However, with enrichment levels below 1.2 wt % U235, there is a net positive gain in resource utilization. With an enrichment level of 1.2 wt % the exit burnup is 22 MWd/kg and the uranium consumption drops to 114 te/GW(e).d. More significantly, the waste arisings decrease to 49.8 te/GW(e).y.

Spent LWR Fuel

Synergy between LWR and CANDU is best illustrated by burning spent LWR fuel in CANDU. There is sufficient fissile content (about 1.6 wt %) in the spent fuel to provide, after removal of the fission products, an energy output of 27 MWd/kg in CANDU. This is in addition to the 35 MWd/kg obtained from the LWR cycle. It should be noted that the fissile utilization (1700 MWd(th)/kg fissile) is poorer compared to the case with an enrichment of 1.2 wt % U235. This is due to a larger reduction in conversion ratio at the higher enrichment level. The uranium consumption, 119 te/GW(e).y is about the same. But the major gain is the reduction in waste arisings, from 49.8 to 18.8 te/GW(e).y.

Table 2

Fuel Cycle Characteristics For LWR's and CANDU

	Fuel Cycle Option	Natural Uranium Requirements (Mg/GWy(e))	Fuel Disposal Requirements (Mg/GWy(e))
1.	Enriched-U in LWR	217	33.2
2.	LWR-Pu recycled in LWR	185	29.2
3.	LWR-Pu and re-enriched LWR-U recycled in LWR	157	24.7
4.	Natural-U in CANDU	157	157.0
5.	Slightly enriched-U in CANDU	114	49.8
6.	LWR-Pu recycled in LWR and recovered LWR-U in CANDU	151	23.8
7.	LWR-Pu and LWR-U recycled in CANDU	119	18.8
8.	Re-clad LWR spent fuel recycled in CANDU	125	19.7
9.	Actinides from LWR spent fuel annihilated in CANDU	0	1.2
10.	Re-clad LWR spent fuel recycled in CANDU/Thorium-U233 converter	98	17.4

There is an incentive to use dry reprocessing of the spent LWR fuel. Dry reprocessing is considered a more easily safeguardable alternative. In the DUPIC, (Direct Use of LWR Fuel in CANDU), the dry process does not remove all the fission product absorbers. The fuel burnup and utilization is less, (18 MWd/kg and 125 te/GW(e).y). This also affects the waste arisings, 19.7 te/GW(e).y.

With the current focus of recycling plutonium in LWRs, there is an opportunity for further resource extension through the use of the uranium recovered from the reprocessing plant. The recovered uranium has a U235 content between 0.8 and 1.0 wt %. It also has a significant U236 content which is a non-fissile absorber. (In a CANDU neutron spectrum which is considerably softer than the LWR neutron spectrum, the parasitic behaviour of U236 is reduced). Here again, the neutron economy of CANDU provides a conversion ratio above 0.7 and an energy contribution from plutonium that is comparable to the natural uranium cycle. Exit fuel burnup is between 15 and 17 MWd/kg of RU. The use of this CANDU fuel cycle together with recycling the plutonium in LWR, reduces uranium consumption from 185 te/GW(e).y to 151 te/GW(e).y.

THORIUM BASED CYCLES

The use of thorium as an alternative fuel to uranium has several attractions. Thorium is more abundant and widespread. Spent thorium fuel is less toxic (has few, if any, of the higher actinides present in spent uranium fuel) and has an ingestion hazard that is an order of magnitude lower. The main disadvantage of thorium is the absence of fissile material in it. This could rule it out as an option if enrichment or reprocessing technology is not available.

CANDU's neutron economy provides a larger advantage when thorium is used as a fertile material instead of uranium 238. The conversion ratio can exceed 0.95. U233 buildup is almost monotonic in CANDU thorium fuel. The fuel life is limited either by fuel performance or by excessive buildup of fission products. Consequently, thorium fuel provides rated power even in a subcritical lattice.

The Self-Sufficient Thorium Cycle

With some re-design of the lattice (and, albeit, with a narrower reactor operating regime) the Th232/U233 cycle in the CANDU reactor can be closed and operated with total independence of external fissile material. The exit fuel burnup is between 10 and 15 MWd/kgHE. This relatively low burnup demands a significant amount of reprocessing in order to recycle the U233 from the spent fuel. Furthermore, some fissile driver is required to startoff the production of U233.

Natural Uranium Driver

CANDU's neutron economy makes it feasible to use a low-grade driver fuel such as natural uranium. Natural uranium CANDU fuel is cheap and the manufacturing technology is well established and has been transferred to several clients. It is expected that thorium CANDU fuel will have the same features. The CANDU reactor provides the use of thorium as a fuel option to clients that do not have access to enrichment or reprocessing technology until the latter become available.

The resource utilization depends on the transition strategy used to implement the thorium cycle into a CANDU reactor that has been operating on natural uranium fuel. Some enhancement of the U233 buildup can be achieved by periodic cooling of the fuel between irradiations. Even in a subcritical lattice, the U233 buildup is sufficient to provide rated power from the thorium channels that are placed in the midst of natural uranium channels. The core thorium inventory is found to be limited by fuel handling capability. With current fuel handling capability, the thorium inventory of the core is limited to about 20 %. Loss of power from fresh thorium channels leads to a reactor power penalty of 12 % during the approach to equilibrium refuelling. Following equilibrium, there is no power penalty predicted. The reason for this (besides the long fuel life of the thorium fuel) is the random selection of channels for refuelling during equilibrium operation which results in a very small number of relatively fresh, i.e. low-powered thorium channels present in the core.

Thorium utilization is comparable to that of the Self-Sufficient Thorium Cycle assuming that a high-burnup thorium fuel is available. This makes the natural uranium/thorium cycle an option to consider by clients when they do not have access to enrichment or reprocessing technology to provide fissile material.

The DUPIC Driver

In the DUPIC/Thorium cycle, the DUPIC fuel replaces natural uranium as the driver fuel. Because of its higher fissile content compared with natural uranium (1.6 vs 0.72 wt %), this cycle permits a larger thorium inventory in the core, as much as 48%. The power penalty during the transition is significantly higher. Also the discharge burnup of the DUPIC fuel remains high, between 15 to 18 MWd/kgHE which implies that the U233 production is sufficient to compensate the neutron load of the thorium.

Reprocessed LWR Fuel

An increase in driver fuel reactivity and in thorium utilization is obtained if fission products are removed from the DUPIC fuel. The fissile content of the driver fuel is the same as DUPIC and the fissile plutonium remains unseparated. With this cycle, the thorium fuel burnup is limited by fuel performance.

The Plutonium-Driven Thorium Cycle

There are two kinds of Thorium/Pu cycles to be considered in CANDU. In the high plutonium utilization thorium cycle, the U233 that is produced during the irradiation of Thorium/Pu fuel is recycled together with a fresh feed of Thorium fuel. This feed may or may not contain fissile Pu as topping to extend fuel burnup. Without Pu topping, the fuel life is limited by lattice reactivity.

In this regard it should be noted that the buildup of U234 with subsequent recycles constitutes, due to the type of neutron spectrum in CANDU, a relatively low fuel burnup penalty. The equilibrium cycle burnup obtained is about 10 MWd/kgHE. Since current CANDU fuel achieves this level of burnup, such a cycle can be implemented without major fuel development.

An increase in initial fissile content is required to reduce reprocessing cost via increased fuel burnup. There are two conflicting requirements that emerge when considering such a cycle; high Pu utilization vs low reprocessing cost.

FBR-CANDU SYNERGISTIC CYCLES

One of the major incentives for the development of breeder reactors is to provide security of fuel supply. This incentive makes sense if the current relatively high cost of producing fuel from breeder reactors can be offset by the strategic advantage of independence, in the future, of uranium supply and cost. A major assumption that is implicit in this approach to ensure the security of fuel supply, is the technical feasibility of fast breeder reactors on a commercial scale. This has not been achieved to-date. Nor is it a certainty in the future. What is certain, however, is that fast breeders are technically feasible but at a high capital cost per installed kilowatt. The latest experience indicates that this cost is an order of magnitude higher than the cost of established thermal reactor power plants. With this cost differential, uranium cost must rise by two orders of magnitude (or its equivalent in commercial availability) to realize the cost benefit of fast breeders.

Basic cost analysis shows that it will be commercially viable to supplement the breeding power of fast reactors with that of established thermal reactor systems to offset the high capital cost of fast reactors in achieving security of fuel supply. In this approach, fissile material from FBRs is used to produce the relatively inexpensive fuel that is currently used in thermal reactors. The range of uranium cost over which commercial viability is achieved in this case will be determined by the conversion ratio of the thermal reactor.

CANDU provides several options to burn fissile material from FBRs. These options are attractive for two reasons. First, the high conversion ratio of a CANDU lattice maximises the energy extracted from the bred fuel. Second, due to the low capital cost of CANDU relative to FBR, an FBR-CANDU system minimises TUEC for a wide range of increased uranium cost. There are several options that are available for the use, in CANDU, of fissile material from FBRs. These options include the direct use of fissile material from FBRs in the MOX, Pu/Th and the Pu/Minor Actinide fuel cycles in CANDU. They also include the indirect options which burn reprocessed spent LWR/MOX fuel in CANDU. The main features of these fuel cycles are given below, starting with the cycle that provides the highest utilization of FBR bred plutonium.

Plutonium driven thorium fuel cycles in CANDU using thorium fuel with low plutonium content (0.9 to 1.1 w% fissile Pu) provide the best utilization of FBR bred plutonium. The low fissile content preserves the near-breeder feature of the Thorium/U233 lattice in CANDU. The plutonium can be used either to initiate a self-sufficient Thorium/U233 cycle in CANDU with subsequent recycling of U233, or it can be used as an integral part of the Thorium/U233 cycle in CANDU to achieve high fuel burnup. In the former option, the fuel life is limited by the lattice reactivity. In the latter, it is limited by fuel performance. In the former case, the fissile plutonium utilization is higher and determined by the buildup of U234 in subsequent reprocessings of spent CANDU fuel. In the latter case it is about 3.5 GW(th).d/kg provided the fuel can achieve 55 MWd/kgHE. Excess plutonium from a 1 GW(e) IFR will support 1.5 GW(e) CANDU installed.

Use of FBR bred plutonium in MOX CANDU fuel provides a fissile utilization of 1.5 GW(th).d/kg in a once-through cycle. This is the result of a lower U238 to Pu239 conversion ratio compared with Thorium. However, it is significantly higher than using the MOX fuel in LWRs. This means that excess fissile plutonium from a 1 GW(e) IFR will support 0.7 GW(e) CANDU.

Use of FBR bred plutonium via the LWR-CANDU tandem fuel cycles improves the overall fissile utilization but it remains less than that provided by the direct MOX CANDU option. The difference is significant if reprocessing of LWR fuel is limited as in the DUPIC (Direct Use of PWR Spent Fuel in CANDU) cycle where the presence of the fission products from the LWR stage restricts the burnup that is achievable in CANDU.

The recycling of fissile material from CANDU spent fuel into FBRs (see Figure 1 attached) is an option that may become available if costs allow the reprocessing of the large volume of CANDU fuel. With one such recycling, supportable CANDU capacity would increase from 0.7 to 1.3 GW(e) with a U/Pu cycle.

Studies of fast reactor cycles show that use of U233 together with Pu239 have higher breeding ratios and lower fissile requirements. Sodium void coefficients are not affected, (1). Using thorium in the blanket but not in the core has the additional advantage of eliminating the reactivity transient associated with Pa233 decay to U233 and reducing the U232 content of U233.

Based on the above, the use of U233 from spent CANDU Th/Pu fuel in FBRs and the subsequently bred Pu in CANDU Th/Pu cycles is a synergistic cycle to be considered.

WASTE ANNIHILATION CYCLES

The Pu/Minor Actinide Cycle

The Pu/Minor Actinide fuel cycle in CANDU is a relatively low utilization cycle that is designed to annihilate minor actinide inventories resulting from fuel reprocessing. To achieve high burnup of actinides, this cycle, unlike the FBR, operates without fertile material in the fuel. This prevents the formation of actinides during the fuel life and provides relatively high net actinide annihilation. Annihilation rates in CANDU are compared in Table 3 with several design concepts. The annual annihilation rate in CANDU is equivalent to the actinide production from an installed capacity of 3.6 GW(e).

Table 3

CANDU-AB PERFORMANCE

	kg Annihilated
ALMR (SPENT LWR FUEL)	273-413
ALMR (Pu FUEL)	825
ALMR (SMALL BURNER)	99
ALMR (LARGE BURNER)	144
LMFBR (ENHANCED SAFETY)	387
PWR (STANDARD)	657
PROTON LINEAR ACCELERATOR	1016
CANDU-AB	1227

Long-Lived Isotope Transmutation

The relatively high neutron flux level in the CANDU reactor, which is a result of its low fissile inventory, is effective in the transmutation of long-lived fission products such as Tc 99 and I 129. The feasibility of this option is enhanced by the space available in the CANDU core due to its large lattic pitch and also the low operating pressure of the moderator system. The annihilation rates shown in Table 4 are based on using Tc 99 as the absorber material in the reactor control elements.

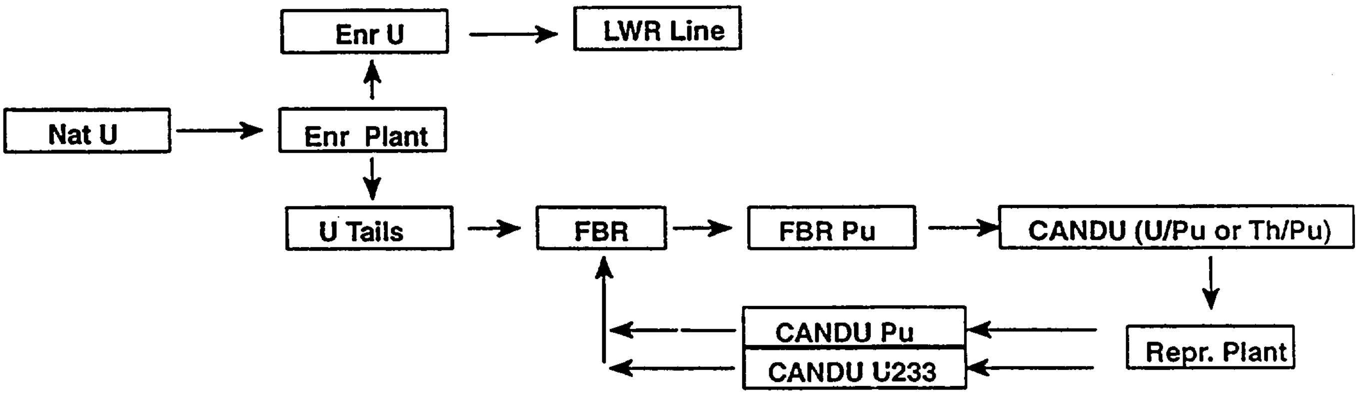
Table 4

TRANSMUTATION RATES FOR Tc99

CASE	FUEL ENRICH-MENT (wt%)	NEUTRON FLUX $(n/cm^2/s) \times 10^{14}$	TRANSMUTATION RATE (kg/a)	TRANSMUTA-TION HALF-LIFE (a)
Central Fuel Pin Metallic Tc99	0.94	1.51	35	44.2
9 Outer Pins	0.95	0.91	40	40.3
Adjuster Rod	1.3	0.91	81	24.5
Moderator Poison	3.2	1.36	207	11.0

Reactor Power 935 MW(e)

Figure 1: FBR/CANDU Synergy



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COMPUTER CODE DEVELOPMENT AND VERIFICATION

(Session 8)

Chairman

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VALIDATION OF COMPUTER CODES USED IN SAFETY ANALYSES OF CANDU POWER PLANTS

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Abstract

Since the 1960s, the CANDU® industry has been developing and using scientific computer codes for designing and analyzing CANDU power plants. In this endeavour, the industry has been following nuclear quality-assurance practices of the day, including verification and validation of design and analysis methodologies. These practices have resulted in a large body of experience and expertise in the development and application of computer codes and their associated documentation. Major computer codes used in safety analyses of operating plants and those under development have been, and continue to be subjected to rigorous processes of development and application. To provide a systematic framework for the validation work done to date and planned for the future, the industry has decided to adopt the methodology of validation matrices for computer-code validation, similar to that developed by the Nuclear Energy Agency of the Organization for Economic Co-operation and Development and focused on thermalhydraulic phenomena in Light Water Reactors (LWR). To manage the development of validation matrices for CANDU power plants and to engage experts who can work in parallel on several topics, the CANDU task has been divided into six scientific disciplines. Teams of specialists in each discipline are developing the matrices. A review of each matrix will show if there are gaps or insufficient data for validation purposes and will thus help to focus future research and development, if needed. Also, the industry is examining its suite of computer codes, and their specific, additional validation needs, if any, will follow from the work on the validation matrices. The team in System Thermalhydraulics is the furthest advanced, since it had the earliest start and the international precedent on LWRs, and has developed its validation matrix. The other teams are at various stages in this multiphase, multi-year program, and their progress to date is presented.

1. INTRODUCTION

Since the 1960s, the CANDU industry has been engaged in the development and validation of safety-related computer codes. The codes have been used in support of safety analyses of CANDU reactors, and in some instances to assist in the planning and understanding of experimental work done at the laboratories. The focus of the industry's validation approach was to gain knowledge through experimental and theoretical studies, implement that knowledge in mathematical models that are validated in separate-effects tests and then installed in computer codes that are tailored to meet current quality assurance practices of reliability and user friendliness.

During the fifteen years leading up to 1990, there was an intense effort on code development and validation to support the CANDU reactors in operation and those under development The task of code validation was supported by an R&D program, presently known as the Safety and Licensing R&D Program of the CANDU Owners Group (COG). The program was jointly funded and reflected the interests that were common to the three Canadian utilities operating CANDU power plants (Ontario Hydro Nuclear (OHN), Hydro Quebec (HQ), and New Brunswick Power (NBP)) and Atomic Energy of Canada Limited (AECL).

Since 1990, the R&D has become more focused on ensuring that code validation is carried out to satisfy both the needs of the industry, for its current design activities and plant operations, and the demands of the regulators. The R&D programs are reviewed both by COG Technical Committees and in-house by AECL. In 1995 June, the industry formed a Code Validation Team, to coordinate code-validation activities in the four partner organizations (OHN, HQ, NBP, and AECL). Building upon work initiated at Ontario Hydro Nuclear the Team's initial focus is the planning of structured validation of the major codes used in safety analyses of CANDU reactors in operation and those under development One of the Team's first outputs was agreement on six main disciplines into which physical phenomena can be grouped conveniently for validation purposes. These disciplines are:

i) System Thermalhydraulics,
ii) Fuel and Fuel Channel Thermal-mechanical Behaviour,
iii) Fission Product Release and Transport,
iv) Containment Behaviour,
v) Physics (comprising reactor physics, shielding, and atmospheric dispersion), and
vi) Moderator Related Thermalhydraulics.

Teams of specialists in each discipline were then formed to carry out the work. Overviews of the current status of validation activities and planning to date in this multi-year validation program are given below.

2. FORMAL APPROACH TO VALIDATION

While the industry's traditional approach to code validation, as outlined in the Introduction, has been in line with international practice, recent developments domestically and internationally have provided the stimulus for a re-examination. Increasingly, the CANDU industry and its regulators expect computer codes to be formally validated within a systematic framework that can be readily audited. Such a framework exists, and its foundations are validation matrices. The Nuclear Energy Agency of the Organization for Economic Co-operation and Development

(OECD/NEA) has recently published [1] validation matrices for LWRs that represent an international consensus in the LWR community on (i) the major, hypothetical accidents, (ii) physical phenomena that might occur during these accidents, (iii) experimental facilities, and (iv) data from separate-effects experiments to be used in the validation of computer codes for safety analyses and licensing submissions. These matrices address thermalhydraulic phenomena in the primary heat-transport circuit, and for pressurized water reactors, also the secondary heat-transport circuit.

The CANDU industry has decided to adopt the validation-matrix methodology for its validation activities, taking into account the state of the art internationally, available expertise, and cost/benefit considerations. Where no international precedents exist, the industry is proceeding with prudence. The steps are typically as follows:

i) identification of accident scenarios to be analyzed,
ii) identification and ranking of physical phenomena relevant to these accidents,
iii) description of the phenomena,
iv) identification of experiments that exhibit the phenomena,
v) description of the source facilities/tests, and
vi) generation of a cross-reference table of phenomena versus relevant experimental data.

The validation matrix comprises the tables in items (ii) and (vi) above.

The industry is examining its suite of safety-analysis codes, with a view of selecting the most appropriate ones for long-term development (if needed), application, and support. The validation matrices will provide the basis upon which to plan further code validation, if needed, to bring code development to closure. The above activities comprise a multi-year validation program, the front end of which is described in the next sections.

3. VALIDATION MATRICES AND THEIR ROLE IN CODE VALIDATION

The validation-matrix methodology has five basic steps, illustrated in Table 1. In the first step, a Technical Basis Document is produced that provides a total overview of all postulated accidents in the design basis of the nuclear plant and the associated main physical phenomena governing the behaviour of plant systems and radionuclides. In the second step, validation matrices are produced for each discipline, relating all relevant physical phenomena to the relevant subset of accidents and to data from experiments, operating plants, mathematical solutions, and benchmark codes. Steps one and two provide the generic knowledge base which is code independent.

Steps three to five are code specific. In step three, a validation plan is produced for each code. The execution of the plan will demonstrate that the code version accurately represents the governing phenomena for each phase of the selected accident scenario. In step four, validation exercises are performed to compare model predictions with selected data sets. Uncertainties in code predictions are estimated. In step five, a validation manual is produced, summarizing code accuracy, sensitivities, and uncertainties for specific applications.

While the validation methodology shows a linear progression through five steps, actual work is being performed in parallel, on steps one and two, and in all six disciplines, to maximize progress on as many fronts as possible and to engage experts in all disciplines. The Validation

Table 1: VALIDATION METHODOLOGY

(1)	Technical Basis Document	Relate safety concerns to main phenomena governing behaviour during each phase of specific accident.
(2)	Validation Matrices (6 in total)	Relate all relevant phenomena to accidents and data sets.
Generic (Code Independent) Knowledge Base		
Code Version Specific		
(3)	Validation Plan	Demonstrate that code version accurately represents governing phenomena for each phase of the selected accident scenario.
(4)	Validation Exercises	Compare model predictions with selected sets (uncertainty).
(5)	Validation Manual	Summarize code accuracy, sensitivities, and uncertainties for specific applications.

Team ensures that the activities are coordinated and that experience gained is shared among participants. The achievements to date and the near-term plans are summarized in the sections below.

3.1 Technical Basis Document

Draft sections of the Technical Basis Document are being produced by experts in the six disciplines, with some sections being in an advanced state of preparation and undergoing peer review. An example is the technical basis for analyses of large loss-of-coolant accidents (LOCA). The logic of that technical basis is illustrated in Figure 1, which relates the safety concerns, behaviours of plant subsystems and radionuclides, and main physical phenomena. Similar descriptions are being produced for other accidents in the design basis.

3.2 System Thermalhydraulics

A validation matrix for system thermalhydraulics has been developed that is based on the physical phenomena that might occur during accidents postulated in the design basis of CANDU power plants. Seven accident categories have been identified and addressed. They are: (i) large LOCA, (ii) LOCA with loss of emergency coolant (EC) injection (LOECI), (iii) small LOCA, (iv) loss of flow, (v) loss of regulation, (vi) loss of feedwater, and (vii) steam-line break. For this ensemble of postulated accidents, 23 phenomena have been identified, assigned an identification number from TH1 to TH23, and their relative importance during the different phases of the accidents has been estimated. That work has been summarized in a 23 x 7 matrix, an extract of which is illustrated in Table 2. For each of the seven accident scenarios, a table has been produced that divides the accident into two to four time periods and identifies primary and secondary phenomena in each period. Table 3 is an extract from the large-LOCA tabulation

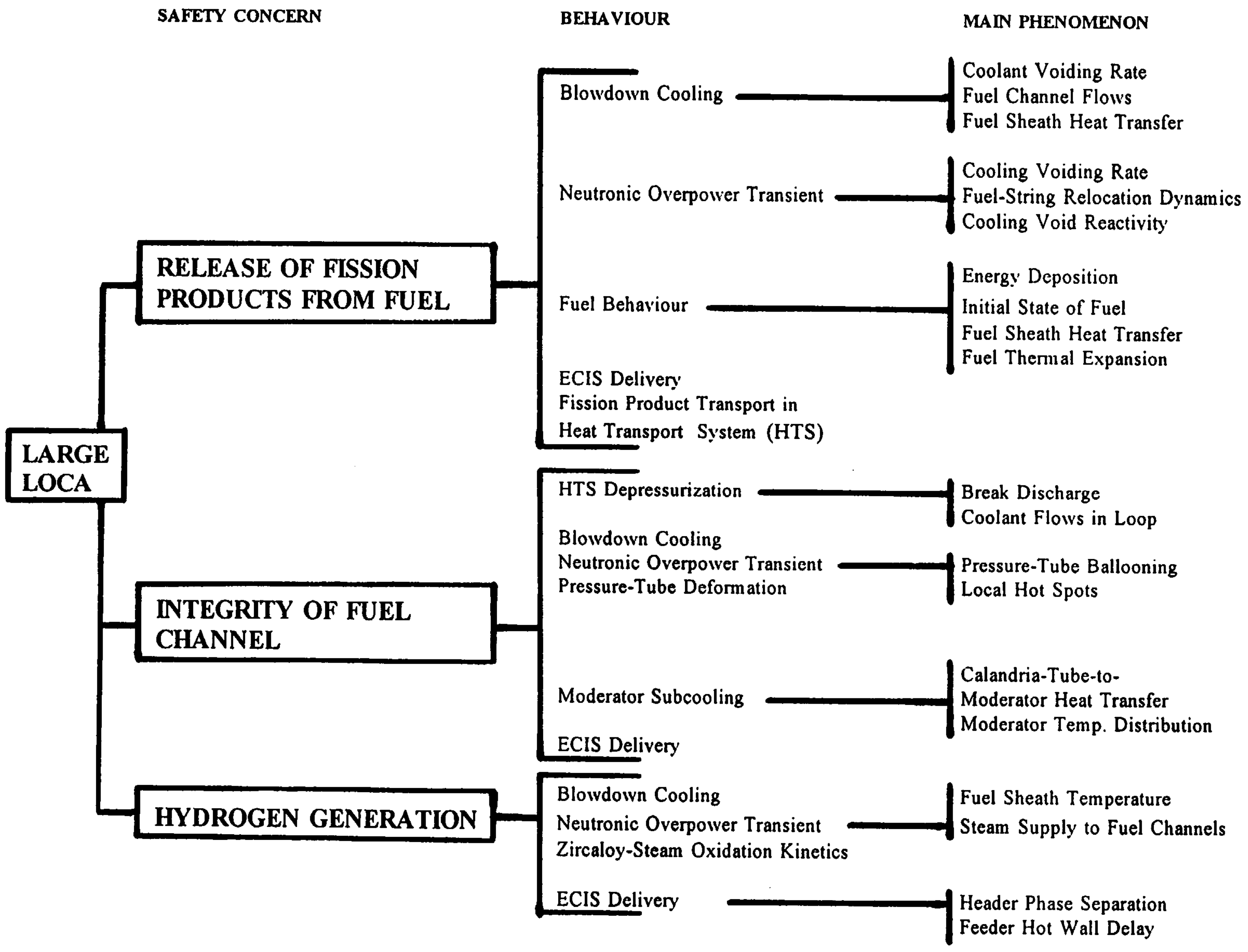

FIGURE 1: TECHNICAL BASIS FOR LARGE LOCA

Table 2: THERMALHYDRAULIC PHENOMENA RELEVANT TO CANDU ACCIDENT ANALYSIS

ID NO	PHENOMENON	ACCIDENT SCENARIO (7)		
		LOCA	LOCA/LOECI →→	STEAM LINE BREAK
TH1 ↓	Break discharge characteristics and critical flow	✓	✓	✓
TH12 ↓	Quench/Rewet characteristics	✓		
TH23	Noncondensible gas effect	✓	✓	

Table 3: RANKING OF PHENOMENA FOR LARGE LOCA

PHASE	POWER PULSE/REACTOR TRIP	EARLY BLOWDOWN COOLING	LATE BLOWDOWN COOLING/EC INJECTION	REFILL
Time(s)	0 - 5	5 - 30	30 - 200	>200
PHENOMENA				
PRIMARY (7) ↓	Break discharge characteristics & critical flow.	Break discharge characteristics & critical flow.	Break discharge characteristics & critical flow.	Counter-current flow.
SECONDARY (3) ↓	Critical heat flux & post-dryout heat transfer	Critical heat flux & post-dryout heat transfer	Phase separation	Waterhammer

Table 4: SEPARATE EFFECTS, COMPONENTS, INTEGRATED EXPERIMENTS, NUMERIC TESTS, AND INTER-CODE COMPARISONS RELEVANT TO THERMALHYDRAULIC CODE VALIDATION

SE1 ↓	Edwards Pipe Blowdown	2 tests
SE25	WL Waterhammer Tests	about 48 tests
CO1 ↓	Stern Labs End Fitting Characterization Tests	about 600 tests
CO5	MR-2 Air-Water Test Loop	about 225 tests
INT1 ↓	Stern Pressure-Tube Burst Tests (IBT Series)	6 tests
INT17	RD-14M Shutdown Cooling Tests	9 tests
NUM1 ↓	JUICE Standard Problems	3
NUM10	Tank Bottom Discharge Test	1
No Inter-Code Comparisons Identified at this Stage		

Table 5: THERMALHYDRAULIC PHENOMENA AND RELEVANT TESTS DATA FOR CODE VALIDATION: Separate Effects Test

ID NO.	PHENOMENA	SE1→	SE4→	SE16→	SE22→	SE25
TH1 ↓	Break discharge characteristics & critical flow	○				
TH12 ↓	Quench/Rewet characteristics		■	■		
TH20 ↓	Waterhammer				■	
TH23	Noncondensible gas effect					

■ Suitable for direct validation
○ Suitable for indirect validation

in which seven primary and three secondary phenomena have been ranked in four significant time periods. Similar rankings have been produced for the other six postulated accidents.

In the next step, relevant available tests, both experimental and numerical, were identified and tabulated. Identification numbers were assigned to separate-effects tests (SE1 to SE25), component tests (CO1 to CO5), integrated tests (INT1 to INT17), and numerical tests (NUM1 to NUM10). An extract from this tabulation is illustrated in Table 4. At this point, no judgement was applied to the suitability of the test data for validation purposes; the data were simply identified. In the next step, the data were reviewed and assessed for suitability for code validation. One of three grades was assigned to each data set as it relates to each of the 23 thermalhydraulic phenomena: (i) not suitable, (ii) suitable for indirect validation, or (iii) suitable for direct validation. An extract from this tabulation is illustrated in Table 5.

To complete the generic part of the validation methodology, descriptions have been produced of the: (i) 23 phenomena, (ii) 37 experimental facilities, (iii) 25 separate effects tests, (iv) 5 component tests, (v) 17 integrated tests, and (vi) 10 numerical tests. The validation matrix comprises the two cross-reference tables: phenomena to postulated accident scenarios (illustrated in Table 2) and phenomena to tests (illustrated in Table 5).

Future work will focus on specific computer codes and the validation matrix. The partner organizations may opt to retain their preferred codes and to identify potential gaps, if any, in the data base and the possible need for additional code development and validation against selected tests from the data base.

The specific tests will be selected to ensure that all phenomena that are likely to be encountered during an accident are addressed. The selection of these tests will be done on the basis of a thorough understanding of the thermalhydraulic phenomena and their rank or relative importance during a postulated accident.

Although the focus of the above work was on CANDU safety analyses, the phenomena have broader applications to other thermalhydraulic systems such as research reactors and experimental loops.

3.3 Thermal-mechanical Behaviour of Fuel and Fuel Channels

The specialist team decided to construct the validation matrix in stages. It was agreed that the initial data sets compiled for inclusion in the matrix would be those potentially suitable for validation of analytical tools used to assess channel-integrity concerns of large LOCAs.

A list of 25 phenomena has been defined that represents all those expected to occur in any of the design-basis accidents. In some cases, mutually dependent phenomena have been grouped and are represented by one observable process. This list has been cross checked for completeness for application to large LOCAs via a detailed review of the relationships between safety concerns, parameters that are used to define margins for each safety concern, and the phenomena that determine the behaviour of each parameter. The latter information will represent the team's contribution to the Technical Basis Document.

Synopses of all phenomena are being prepared. Initial definitions have been compiled and the task of preparing detailed descriptions has been distributed to team members according to their area of expertise. A preliminary ranking of phenomena, as either primary or secondary importance, has been completed for each phase of the large-LOCA scenario. An initial draft list of 117 data sets has been compiled. Drafting of synopses for an initial selection of 30 of these is currently underway, with synopses of 12 of the in-reactor data sets now completed. A draft matrix has been prepared that cross references the 25 phenomena to each of the 117 data sets. This initial correlation is based on preliminary expert judgment and still requires confirmation, following the preparation of data-set synopses.

3.4 Fission-Product Release and Transport

Due to the complexity and clear differences between the phenomena that control the fission-product release and the fission-product transport processes, for simplicity, the discipline was divided into these two sub-disciplines, and specialist teams were formed in each. To avoid superposition, it is necessary to define the region of application for each sub-discipline. The following definitions have been adopted:

i) The Fission-Product Release sub-discipline includes all fission-product phenomena occurring in a fuel element up to the release of radionuclides via sheath failure.
ii) The Fission-Product Transport sub-discipline includes all fission-product phenomena occurring between sheath failure and release of radionuclides into containment.

Lists of 20 fission-produce release phenomena and 23 fission-product transport phenomena have been produced. The lists of phenomena are under review by the team members and other members of the Canadian nuclear industry. Synopses that describe each of these phenomena and the identification of their key parameters are in preparation. As a trial case, the large LOCA combined with LOECI was selected for the phenomena-ranking process. The fission-product release phenomena were ranked as of primary or secondary importance with respect to their perceived impact on the amount of fission-product releases during a particular phase of the accident.

Preliminary identification of available experimental information on fission-product release indicates that the following tests are possible choices for the validation matrix: (i) 45 in-reactor tests, (ii) 200 in-cell tests, and (iii) 15 laboratory tests. Each test will be assessed to determine which phenomena occurred during the course of the test. This experimental data base includes experiments performed around the world. Some of these experiments, primarily in-reactor tests, were CANDU specific. The in-cell and laboratory tests have a wider application area.

In the area of fission-product transport, identification of relevant validation data sets is in progress. The data sets for code validation will include experiments performed in Canada, e.g., laboratory aerosol-transport tests, hot-cell fission-product-transport tests, and in-reactor tests performed in the Blowdown Test Facility at the Chalk River Laboratories. The data sets for the validation of fission-product-transport codes will also include international separate-effects and integral experiments such as those from the PHEBUS-FP program. After appropriate tests have been identified, the data sets will be summarized and the uncertainties in the data will be quantified.

3.5 Containment Behaviour

The discipline was divided into the sub-disciplines of (i) Containment Thermalhydraulics and Hydrogen Behaviour, and (ii) Fission Product Chemistry and Aerosol Behaviour, and specialist teams were formed in each.

The current status of the draft chapter for the Technical Basis Document is as follows. Postulated accident scenarios have been identified, and one is described in detail. Safety concerns for the chosen accident scenarios have been identified, described, and tabulated. Fundamental phenomena have been identified along the sub-discipline lines. Six phenomena have been described, as examples of the detail required for the final document. A table showing the relative importance of the phenomena for the accident scenarios has been produced.

The current status of the draft Validation Matrix Report is as follows. The available data base has been organized into categories, with 25 separate-effects tests, 3 integrated tests, and 7 numerical tests covering the areas of containment thermalhydraulics, hydrogen combustion, fission product chemistry, and aerosol behaviour. An additional category, inter-code comparisons, is included, but no data sets have been identified because the benefit of this category to code validation is not clear at this time. Separate-effects tests, integrated tests, and numerical tests have been described briefly. Validation-base data sets and the number of individual tests in each set have been tabulated. The cross-reference table of the validation matrix that relates data sets to the phenomena identified in the Technical Basis Document has been prepared.

3.6 Physics

A specialist team has been assembled to work on defining a validation matrix for the sub-discipline of reactor physics, seen as the area of high priority. While ad hoc validation work in the sub-disciplines of shielding and atmospheric dispersion of radionuclides is ongoing, it does not yet follow the validation-matrix methodology.

Preparation of the validation matrix for reactor physics will begin shortly, and the steps outlined in Section 2 above will be followed.

In advance of the above work, AECL experts in physics produced preliminary documents on validation of physics codes, in all three sub-disciplines, that are in common use at AECL. These documents collect in one place information that has been generated over many decades and is dispersed in many references. These documents are useful now and are expected to make it easier to develop the validation matrix reports in the physics area.

3.7 Moderator Related Thermalhydraulics

A specialist team has been formed to address moderator related thermalhydraulics, and the team has identified its scope of work.

To date, the following tasks have been completed. A preliminary list of accidents involving the systems has been prepared. A preliminary list of concerns, behaviours, and phenomena for each accident has been developed. A preliminary table relating the major phenomena to accidents has been prepared. Results of the above have been presented at meetings and circulated for comment.

4. FUTURE VALIDATION WORK

The methodology described in the preceding sections defines the course of action needed to achieve the end point, which is computer codes, validated according to a structured methodology, and suitable for future safety analyses of CANDU plants and associated licensing decisions. That end point will bring to closure some of the code-development work and associated R&D, which in some instances has been ongoing for decades. The end products of the work presently under way will be a Technical Basis Document and six Validation Matrix Reports, the first of which has been completed. These documents will provide the basis for planning the next steps in the validation program. In the next steps, the most appropriate computer codes will be selected and, if needed, validation plans for them will be defined. Any further code development will be focused on identified shortcomings. If gaps exist in the validation data base that can be addressed by additional R&D, such R&D will be specified and executed.

5. NUCLEAR SYSTEMS OTHER THAN CANDU PLANTS

The preceding sections address the needs, with respect to validated computer codes, of the operators, designers, builders, and regulators of CANDU plants. AECL also operates other nuclear facilities, notably research reactors, and AECL designs, submits for licensing, and builds small reactors of the MAPLE family. The computer codes used in much of that work are often versions of those used in the CANDU business and hence require similar levels of quality assurance, including validation. The validation program described here provides a solid foundation to which specific validation work can be added to meet AECL's needs in the non-CANDU line of business.

6. SUMMARY AND CONCLUSIONS

The Canadian CANDU industry has 35 years of experience in the development and application of computer codes used in safety analyses and licensing submissions. While these computer codes were validated as a matter of course during their development, that validation was performed according to the practice of the day. No single, systematic validation methodology was used because none existed. Recently, the OECD/NEA developed and published a validation methodology for LWRs. That methodology is centred on a validation matrix, comprising two cross-reference tables: the first identifying physical phenomena that might occur in design-basis accidents, and the second identifying experimental and numerical tests that exhibit the physical phenomena. The validation matrix is generic to the chosen type of nuclear plant and serves as the basis for the validation of specific computer codes.

The Canadian CANDU industry adopted the fundamentals of the OECD/NEA validation methodology and is adapting and extending it to CANDU power plants. Industry-wide teams of specialists have been formed to develop validation matrices in six scientific disciplines,

i) System Thermalhydraulics,
ii) Fuel and Fuel Channel Thermal-mechanical Behaviour,
iii) Fission Product Release and Transport,
iv) Containment Behaviour,
v) Physics (comprising reactor physics, shielding, and atmospheric dispersion), and
vi) Moderator Related Thermalhydraulics.

These disciplines cover a much broader range of phenomena than those addressed by the OECD/NEA.

The team in System Thermalhydraulics has the lead and has produced a validation matrix document. Teams in the other disciplines are at various stages in developing their validation matrices which will be generic in each discipline. The validation program is expected to span several years and to bring to closure the development of computer codes, validated according to a structured methodology, and suitable for safety analyses of, and licensing decisions on CANDU power plants. While this is the primary focus for the work currently under way, the methodology and results will also provide a basis for the validation of computer codes used in safety analyses of nuclear and experimental facilities other than CANDU power plants.

ACKNOWLEDGMENT

The authors acknowledge the work performed by the specialists at OHN, HQ, NBP, and AECL, too numerous to cite individually, and the members of the Validation Team, all of whom devote their energies and time to this joint program. Since the Specialists' Team in System Thermalhydraulics has completed its validation matrix report, portions of which are quoted in this paper, the work of J.M. Pascoe, A. Tahir, J.P. Mallory, and T.V. Tran is hereby gratefully acknowledged.

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COMPUTER CODES FOR SAFETY ANALYSIS OF INDIAN PHWRs

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Abstract

Computer codes for safety analysis of PHWRs have been developed in India over the years. Some of the codes that have been developed in NPC are discussed in this paper.

Computer code THYNAC and ATMIKA have been developed in NPC for the analysis of LOCA scenario. Both the codes are based on UVET model using three equations and slip correlations. Computer code ATMIKA is an improved version of code THYNAC with regard to numerics and flexibility in modelling. Apart from thermal hydraulic model these codes also include point neutron kinetics model. ATMIKA has been extensively used for the analysis of LOCA scenarios. This code is further being developed for dynamic simulation. Estimation of containment pressure/temperature transients is done by code 'PACSR' developed in NPC where discharge of mass and energy obtained from LOCA/Main Steam Line Break analysis is used as input. Hydrogen distribution in containment during accident can also be estimated by using this code. This code includes the modelling for the performance of engineered safety features of containment system. Computer code CONTACT has been developed to analyse thermal-mechanical behaviour of coolant channel under postulated condition of LOCA & coincident impairment of emergency core cooling, so that the predominant heat removal is through the pressure tube - calandria tube to the moderator surrounding the channel. Codes COOLTMP and RCOMP are used to estimate heat-up of primary coolant & core components respectively under off-normal shutdown conditions as may be existing during special maintenance job or postulated failure.

Code validations have been performed either against experiments or the published results of experiments performed elsewhere, or through International benchmark excercises sponsored by IAEA.

The paper discusses these codes, their validations & salient applications.

1. INTRODUCTION

Computer codes for safety analysis of Pressurized Heavy Water Reactors (PHWRs) have been developed in India over the years. Most of the codes in use in NPC have been developed in-house. Their validations have been performed either against experiments, or published results performed elsewhere. In some cases, comparisons against calculated results from other computer codes are also performed. Given below is a brief resume of these codes, their validations and salient applications.

2. LOSS OF COOLANT ACCIDENT ANALYSIS - COMPUTER CODES THYNAC/ ATMIKA

In the Indian PHWR, the design basis includes consideration of a variety of postulated initiating events, including breaks in primary heat transport (PHT) system, resulting in loss of coolant accident (LOCA). A range of break sizes and locations are required to be considered, upto and including a double ended (guillotine) break in the largest diameter piping (headers) in the PHT system. It is necessary to perform LOCA analysis as part of the safety assessment of the plant, and to demonstrate the adequacy of the safety systems, viz. reactor shutdown system, and emergency core cooling system to mitigate the consequences of such accidents.

Computer codes THYNAC [1] and ATMIKA [2] have been developed in NPC for the analysis of LOCA scenario. Both these codes are based on UVET model using three equations and slip correlations. Computer code ATMIKA is an improved and advanced version of code THYNAC with regard to numerics, controls and special modeling like valve, pump, non-homogeneous volume and steam generator. Code ATMIKA offers more flexibility in modelling and stable numerics.

2.1 Code Description

In the code ATMIKA, the various elements of the reactor & coolant systems are represented by a system of nodes connected by flow paths. The governing equations consist of three conservation equations, viz. mass, energy and momentum conservation equation, along with a drift flux correlation.

A staggered mesh arrangement is used where pressure, density and enthalpy are defined at the centre of the cell, and flow is defined at the junction. Mass and energy equations are applied at lumped control volume defined as node. Momentum equation is applied on flow path. The governing equations are used in conjunction with drift flux based on Armand's Correlation which relates void fraction , with volumetric flow ratio B, by the equation = $C_A B$. Where, the coefficient C_A is a function of Pressure, void fraction and inclination. A correlation is derived to calculate C_A to match with wide range of experimental data.

Some of the important models used in the code are point kinetic model for estimation of reactor power; heat conduction model for estimation of radial temperature profile in fuel rod, pipe walls and heat exchanger tubes; a wall heat transfer package for estimation of heat transfer from fuel to coolant and coolant to heat exchanger tube including various transition boundary e.g. CHF, MSFB; Martinelli-Nelson correlation for evaluating two-phase frictional pressure drop; homologous model for evaluating pump behaviour under single phase and two phase conditions; critical flow estimation at break is made using options from several available correlations e.g. Burnell's (for single phase liquid), Moody, HEM, frozen slip model.

In the numerics of the code a modified version of the usual semi-implicit scheme is adopted. In this method, the mass flow is calculated assuming previous value of pressure from momentum equation. The new value of mass flow so obtained is used in energy and mass equation to calculate mixture internal energy and density. The pressure and other properties are found out using the equation of state at new time step. This method is evolved to get maximum numerical stability with higher time step size.

2.2 Application of ATMIKA

Computer code ATMIKA is being used to analyze a spectrum of break sizes at different locations in PHT System of PHWRs. A nodalisation scheme of ATMIKA simulation for PHT System, Emergency Core Cooling System (ECCS) and Secondary System is shown in Figure-1. The analysis covers the plant response to Loss of Coolant Accident with all safety systems functioning as well as response following LOCA with postulated impairment in safety systems viz. ECCS. As large break LOCA imposed most severe requirement on all safety systems, viz. Shutdown System, ECCS and Containment, it has been analyzed for a wide range of break sizes starting from 10% (lower end of large break LOCA) upto 100% double ended guillotine break of large diameter headers. Short-term system response under large break LOCA in Reactor Inlet Header (RIH) and Reactor Outlet Header (ROH) are depicted from Figure-2 to Figure-6. Figures 2a and 2b show the variation of actuation timing of shutdown system and ECCS with break size respectively for the spectrum of break sizes considered for RIH and ROH. It is observed from these figures that reactor shutdown is accomplished within few seconds and ECCS also triggered and latched in desired mode of selective header injection in an effective manner. Figure-3 shows variation of peak sheath temperature with a range of break size. Figure-4 shows variation in time to attain peak sheath temperature and temperature more than 800°C with range of break size. This figure also gives the duration of sheath temperature more than 800°C (above which contribution of metal-water reaction increases significantly) with break size. Range of depressurization and coolant flow transients at the centre of broken pass following RIH breaks is shown in Figures-5 & 6 respectively. These figures also identify the region / break range within the spectrum of large break LOCA which results in early flow stagnation and produces most severe fuel temperature excursions. The identified limiting (critical) breaks which imposes most severe requirement on ECCS alongwith guillotine break of RIH and ROH which imposes most severe requirement on containment are analyzed for long-term system response covering entire transient period starting from early blowdown to long-term quasi-steady state phase. It is seen, in general, period of flow stagnation is brief, by the time sustained flow from low pressure recirculation system is established in the PHT system, adequate water inventory is still left in the accumulator and complete fuel-clad quenching and refilling occurs in less than ten minutes in large break LOCA scenarios. After rewetting of the channels, long term recirculation injection from ECCS maintains the reactor core in reflooded condition to remove decay heat on long term steady state basis. Analysis has also been done with and without the availability of Primary Coolant Pumps (PCPs) during the transient. The analysis indicated the need for tripping PCPs

ATMIKA : LOCA ANALYSIS
KAIGA FEEDER BREAK

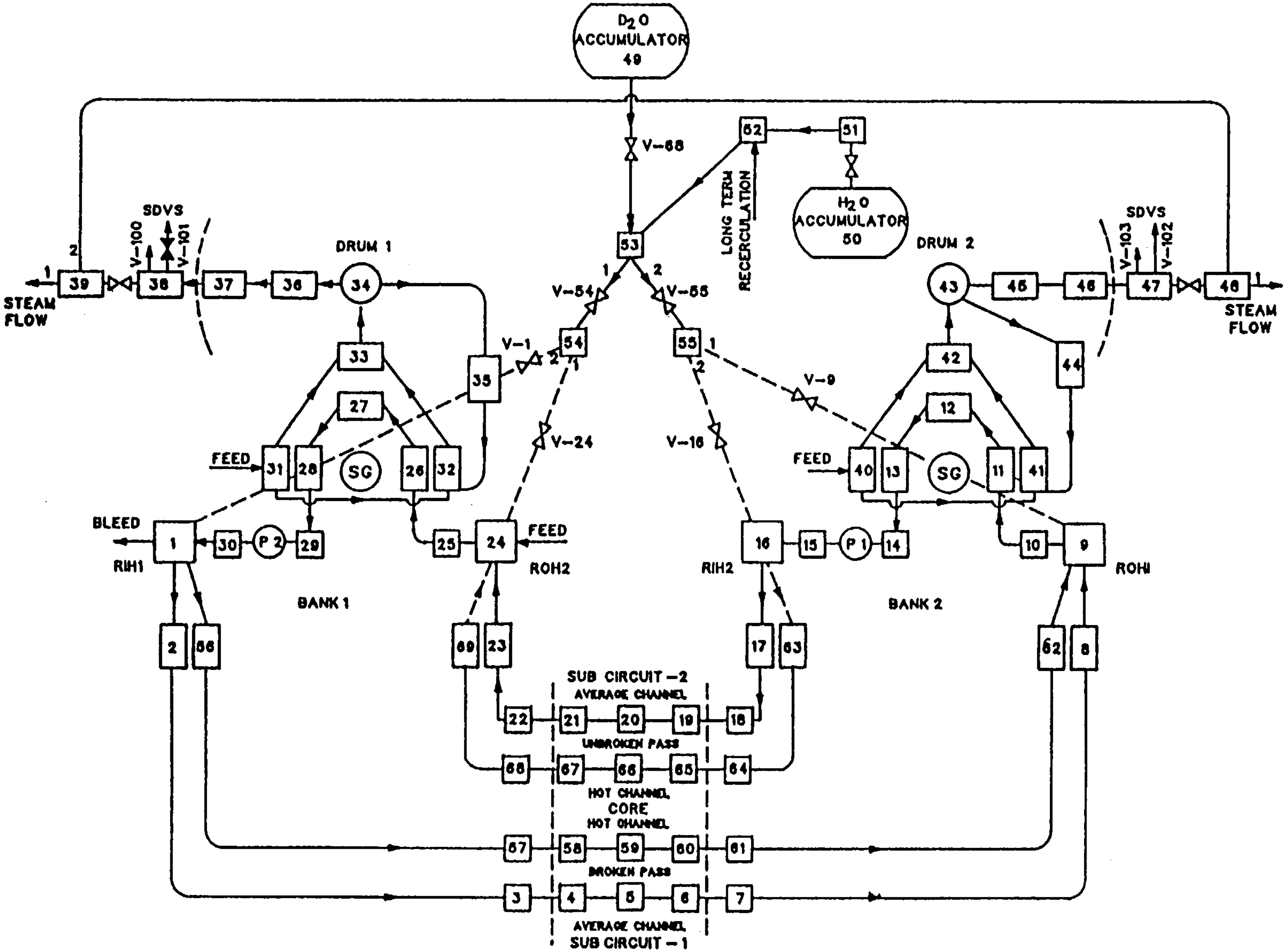

FIG :- 1 NODALISATION SCHEME

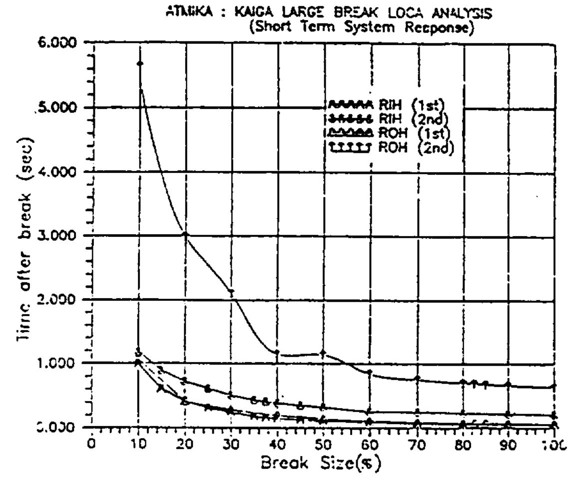

FIG. 2a : VARIATION IN TIME OF TRIP SIGNAL WITH BREAKSIZE

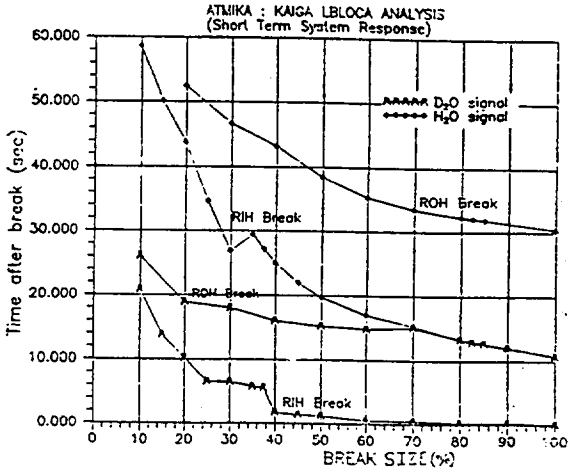

FIG. 2b : VARIATION IN TIME OF ECCS SIGNALS WITH BREAKSIZE

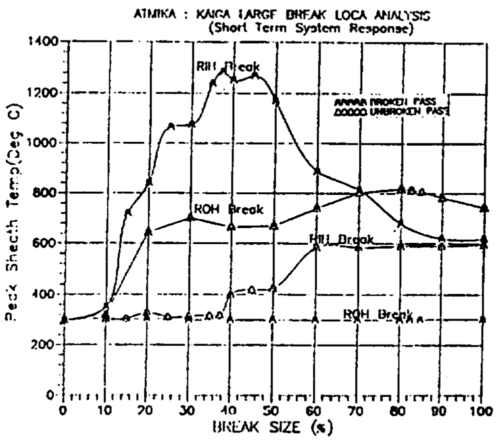

FIG. 3 : VARIATION OF SHEATH TEMPERATURE WITH BREAK SIZE

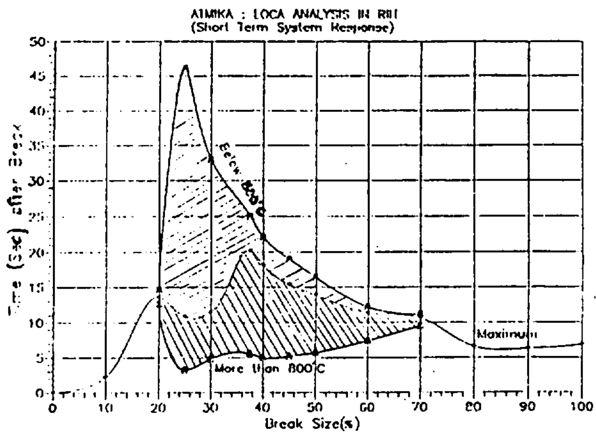

FIG. 4 :TIME TO ATTAIN VARIOUS SHEATH TEMPERATURES WITH BREAK SIZE

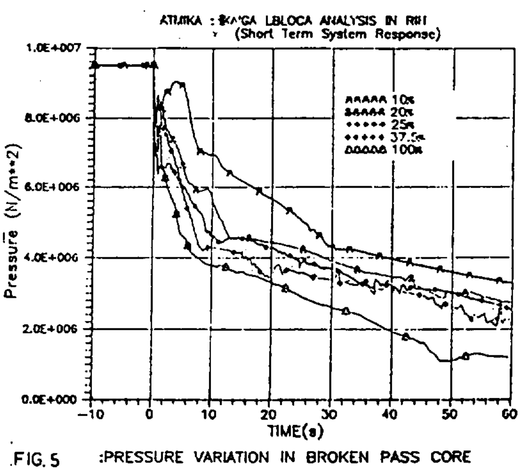

FIG. 5 :PRESSURE VARIATION IN BROKEN PASS CORE

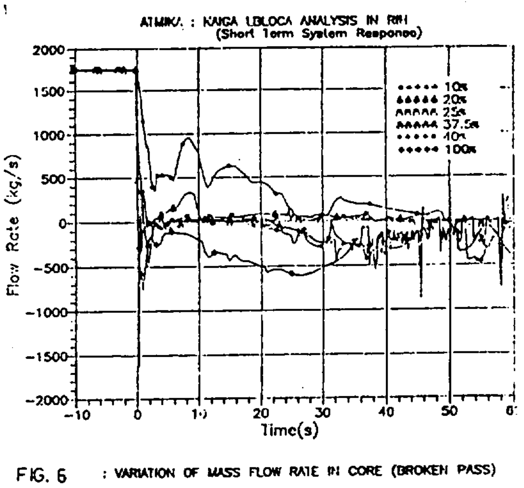

FIG. 6 : VARIATION OF MASS FLOW RATE IN CORE (BROKEN PASS)

under certain LOCA conditions involving ECCS injection in Type-II/Type-III mode to reduce the stagnation period [3].

In applying the results from this analysis with regard to the extent of fuel failures and resulting radiological consequences, due consideration is given to the uncertainties in predicting fuel temperatures in horizontal channels for low flow/high void conditions in the PHT system.

2.3 Validation of ATMIKA

The computer code ATMIKA has been validated [4] against the available experimental results of blowdown from "RD-4 loop", a Canadian test facility designed to simulate LOCA in PHWRs. Validation analysis has been performed with respect to the following two experiments [5]: 7.5% break at inlet feeder with pumps operating and 50% break at inlet feeder with pumps bypassed.

The predicted results of ATMIKA are compared with the experimental results for the two cases. Depressurization transient at heated section outlet, sheath temperature variation in heated sections and differential pressure across heated sections are shown in Figure-7a,b & c for 7.5% break with pump operating and in Figure-8a,b & c for 50% break with pump bypassed respectively. As seen in these figures, in general, the agreement between predicted and experimental results is reasonably good, with consistent trends on pressure transients during LOCA and conservative predictions with respect to fuel sheath peak temperatures.

Further validation exercises for the code are planned, as more data becomes available.

3. CRITERIA FOR FUEL FAILURE DURING ACCIDENT CONDITIONS

While analysing accident sequences viz. LOCA involving impaired cooling of fuel, prediction of the consequences requires estimation of the extent of fuel failures that may occur during the accident. For this purpose, a quantitative criteria was developed to define the conditions for fuel failure [6]. This criterion is built on the consideration that fuel cladding may fail when (a) <u>local</u> stress equals burst stress or (b) when oxidation of the clad exceeds a pre-set limit.

The burst stress depends on temperature and extent of oxidation of zircaloy clad. The local stress is affected by straining as determined by temperature and pressure dependent creep of the clad. Thus the possibility of clad failure will ultimately depend on history of temperature and pressure seen by the clad, and its oxidation. An analytical model, which also considers the fission gas pressure based on fuel burnup was developed. In this model, the uniform strain of clad considered in estimating the time dependent fission pressure is conservatively assumed to be limited to 5%.

Fig.9 shows a typical prediction, derived from above criteria, which shows the time in which fuel would fail, vs. coolant pressure for different limiting clad temperatures.

4. REACTOR CHANNEL HEAT-UP EFFECTS DURING POSTULATED ACCIDENT CONDITIONS OF LOCA & NON-AVAILABILITY OF ECCS

In Indian PHWRs, one of the postulations considered in design safety analysis is a dual failure involving LOCA simultaneous with coincident failure of ECCS. A computer code, "CONTACT" has been developed to simulate the reactor channel behaviour under high temperature transient expected under these conditions. In the code, reactor channels are grouped based on power distribution. The analysis is caried out on a representative channel (Max. rated channel of that group) of each group. The channel is divided into 10 segments corresponding to 10 active fuel bundles of the channel. In the cross section of the channel the fuel elements, pressure tube and calandria tube is modelled by system of annular rings. The analysis is performed with a standard one dimensional heat transfer equation derived from lumped parameter technique of heat generated in fuel and radial heat transfer to the moderator by radiation, natural convection and then boiling heat transfer. Pressure tube sagging under high temperature creep is calculated by bending creep strain as a function of time. Metal-water reaction is assessed by Baker-Just equation. A detailed analysis, using this code has been carried out for this scenario [7] and some of the typical results of the analysis are presented here. These include Fuel, Pressure tube and Calandria tube temperature histories (Fig.10a), clad oxidation histories (Fig.10b) and variation of total heat generation in the reactor core (Fig.10c) under postulated accident conditions.

These analytical studies led to the conclusion that the moderator system is an effective heat sink under these postulated accident conditions provided at least 10% of the normal Process Water cooling flow is available. With moderator serving as the heat sink, the core damage effects would be limited to extensive fuel clad failure (around 77%) but no melt of UO2. About 30% of clad zircaloy inventory is predicted to oxidise, generating 23Kg.- moles of deuterium.

Modification and improvements in the computer code "CONTACT" are planned with respect to simulation of transient 2.D heat conduction in the fuel element and pressure tube. Further, asymmetric sagging model of the coolant channel, required to account for asymmetric heat generation in the coolant channel associated with convective cooling of steam flow in the channel and exothermic heat generation in metal-water reaction of Zircaloy with steam is being simulated to predict more refined fuel temperature transient and hence associated fuel sheath failures in the postulated accident.

5. CONTAINMENT ANALYSIS - COMPUTER CODE PACSR

The design of containment system requires evaluation of pressure, temperature transients in the containment following postulated accidents resulting from break in the pipe of primary heat transport system or secondary steam line system. For this purpose, computer code 'PACSR' (Post Accident Containment System Response) [8] was developed in NPC for pressure suppression type of containment system which has been adopted in Indian PHWR stations since MAPS. In this code the containment can be

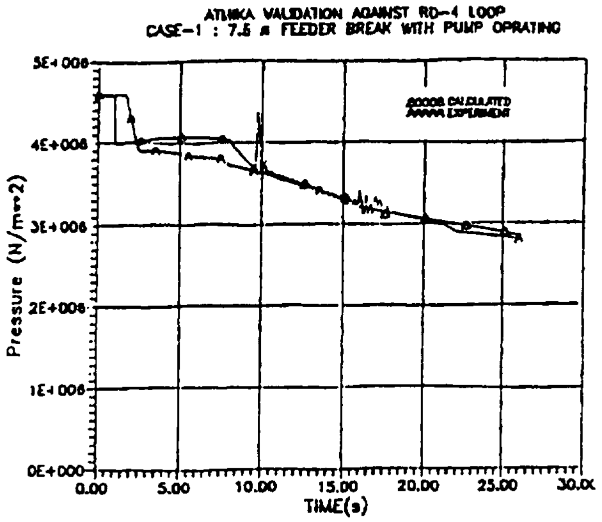

Figure-7a Pressure variation in HS-1 outlet

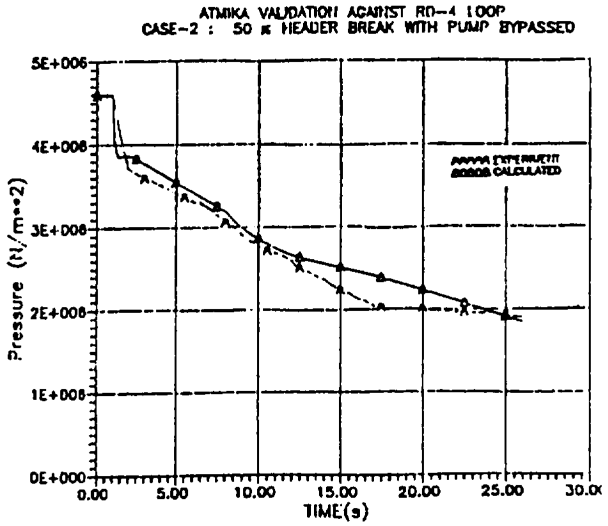

Figure-8a Pressure variation in HS-2 outlet

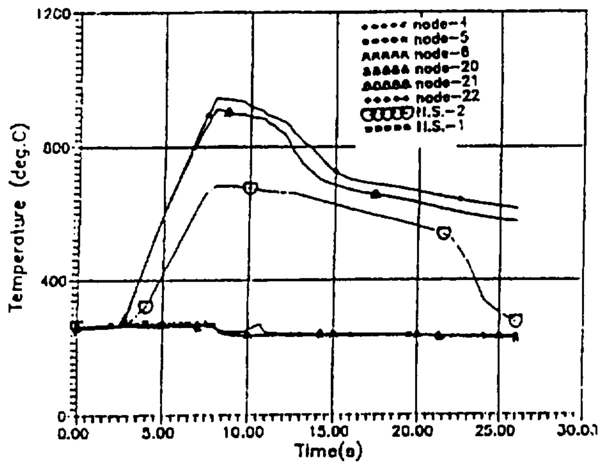

Figure-7b Sheath temperature variation in heated sections

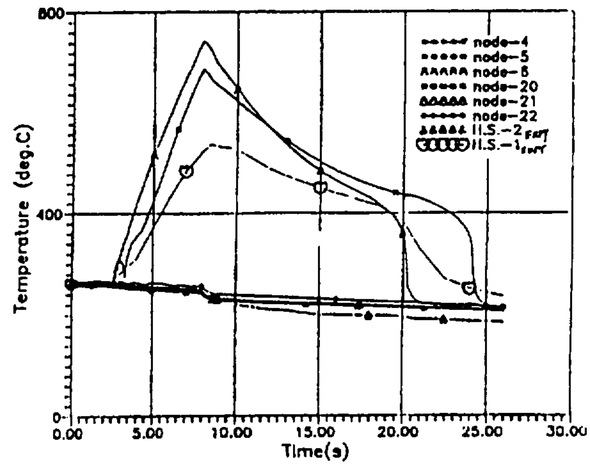

Figure-8b Sheath temperature variation in heated sections

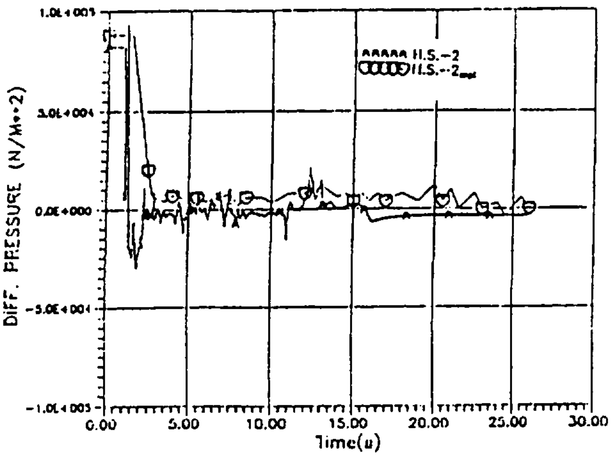

Figure-7c Differential pressure variation across HS-2

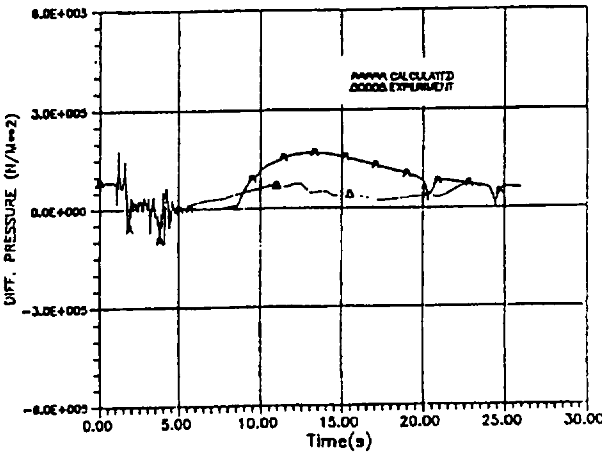

Figure-8c Differential pressure variation across HS-1

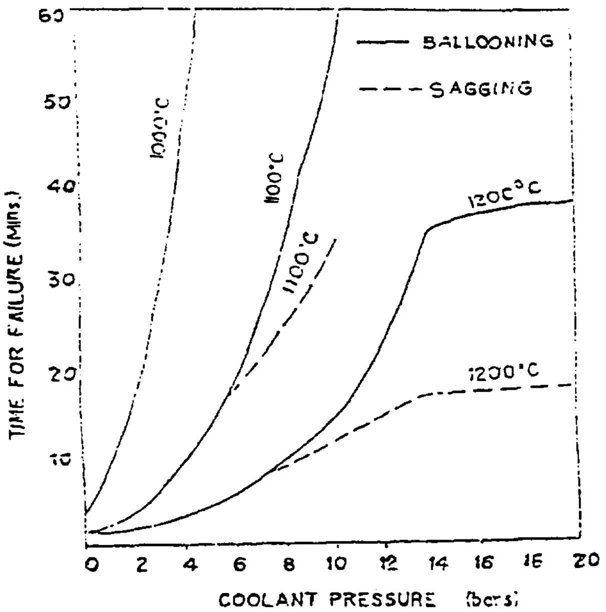

Figure-9 PHWR fuel failure prediction

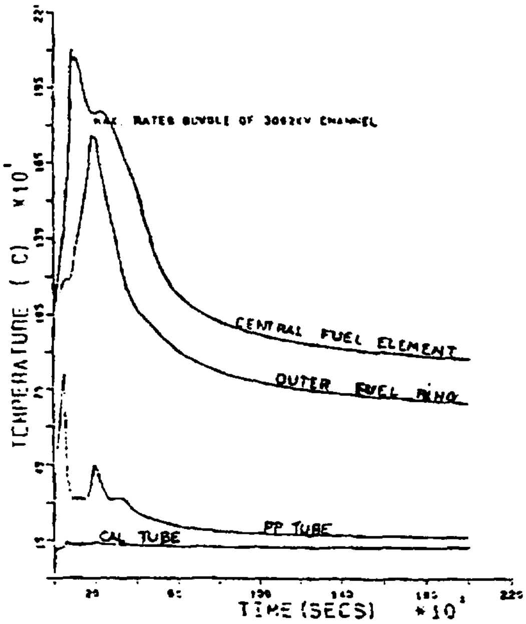

Figure-10a Fuel and temperature following LOCA and ECC failure

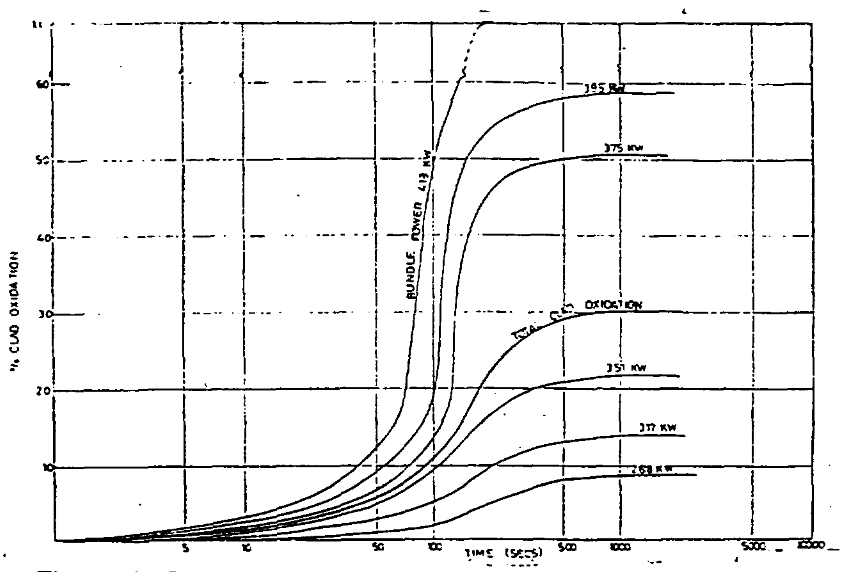

Figure-10b Clad oxidation vs. time following LOCA and ECC failure

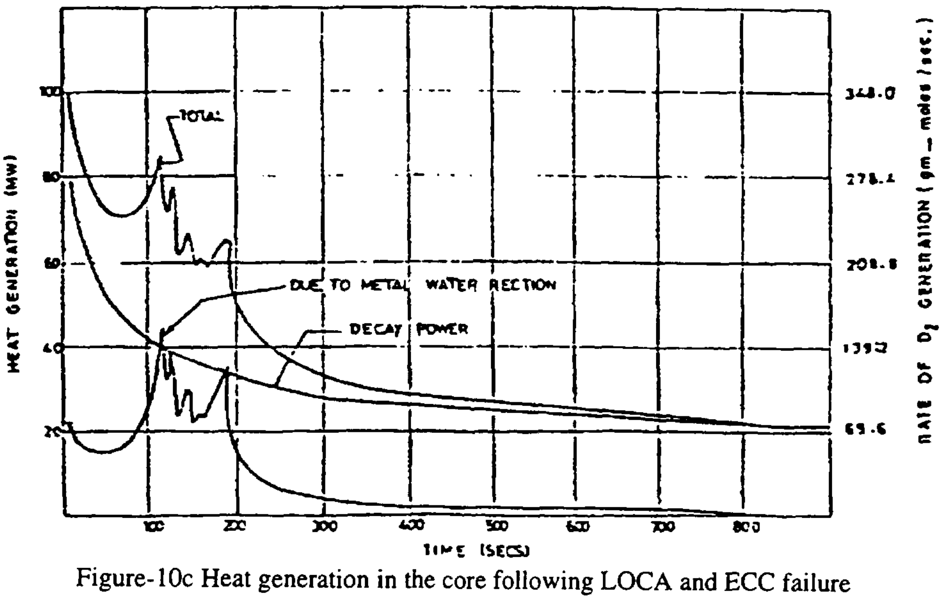

Figure-10c Heat generation in the core following LOCA and ECC failure modelled by a number of compartments separated by appropriate interconnections. Each compartment is treated as having two regions : liquid and vapour. The liquid region consists of water on the floor as either condensed steam or suppression pool water. Above the liquid region in each compartment the vapour region comprises of non-condensible gases (air and hydrogen), steam and liquid droplets.

The time dependent mass and energy blowdown discharge rate into the containment system is an input to the code. A finite difference method is used for the solution of differential equations which are solved explicitly. The analysis for vent clearing and subsequent flow through the suppression pool vents considers inertia of water and air columns, as well as the varying opening area of the horizontal vents. The code can model various features for containment depressurization and cooldown viz. containment air coolers, spray system and structural walls/floors in the containment as heat sink. For modelling the latter, several heat conducting elements, with a provision to subdivide into two regions of different material, have been modelled by one dimensional heat conduction equation. Temperature drop due to interfacial resistance between two regions can be modelled. For condensation heat transfer at element surface several correlations have been provided, namely Tagami, Uchida, Nusselt, Othmer and diffusion controlled heat transfer coefficient. The diffusion controlled heat transfer coefficient for condensation, which depends on partial pressure of steam, and saturation pressure corresponding to temperature of wall surface, gives the closest matching in validation exercises. Accordingly this correlation is currently being used in applications of this code.

The code PACSR has been validated against a range of experimental results, which include a set of experiments performed on a one-tenth scale model of PHWR containment test facility located at MAPS, Madras [9], against published results of experiments on the German HDR containment test facility (ISP-23) [10] and against experiment D15 on Bettelle Frankfurt Containment model.

The one-tenth scale model of PHWR containment simulates the various internal compartments as well as the vapour suppression pool. To simulate the mass and energy discharges from a LOCA break, a PHT system model vessel was provided outside with a discharge pipe having a rupture disc at its end leading into the containment vessel. Fig.11 shows the results for original calculation (curve marked C) as well as the one with the improved modelling (curves C2 and C3), and compares these with test results (Curve-T) [11]. The improved modelling pertains mainly to the correlations for heat transfer from containment atmosphere to structures. The published results of a test on German HDR containment, (Test No.V44, offered as an International Standard Problem exercise ISP-23) were also used for computer code 'PACSR' validation. Fig.12 shows the calculated pressure transient in the dome region as calculated by code PACSR and compares it with experimental results as well as reported values calculated from other computer code viz. PRESCON (Canada), CONTAIN 1.1 (Netherlands) etc. For further validation, data obtained from an experiment (D15) on Battele Frankfurt containment model were used. Fig.13 shows the calculated pressure transient in break compartment by code 'PACSR' and compares it with experimental values. The peak calculated pressure is about 15% higher than the observed one [12].

Fig.14 depicts typical results of analysis performed with PACSR for Kaiga containment showing pressure transients in various compartments following a double ended Main Steam Line Break.

This code has also been used for study to simplify and optimize the existing dousing scheme at RAPS and to see if its flow modulating scheme could be eliminated. For performing this analysis the code PACSR was suitably modified to model the dousing feature. The results of study are shown in Figure-15 which give the contianment peak pressure for various dousing flows (% of existing maximum flow), for various LOCA break sizes [13]. From these results it is concluded that a fixed dousing flow in the region of 30-35% of the existing maximum flow would cater to all LOCA break sizes.

6. HYDROGEN DISTRIBUTION IN CONTAINMENT DURING ACCIDENT CONDITIONS

The studies of hydrogen distribution and transport in containment during certain postulated accident conditions, are particularly important as they determine the potential combustion of hydrogen. A modified version of computer code PACSR (named PACSR-MOD2) has been developed for the analysis of hydrogen and its distribution in the reactor containment with regard to flow,

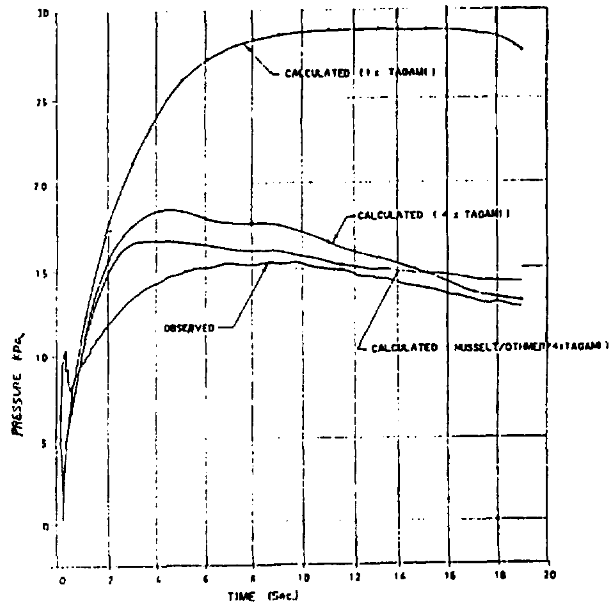

Figure-11 Containment pressure transient for CASE 1; Effect of heat transfer correlation on surfaces

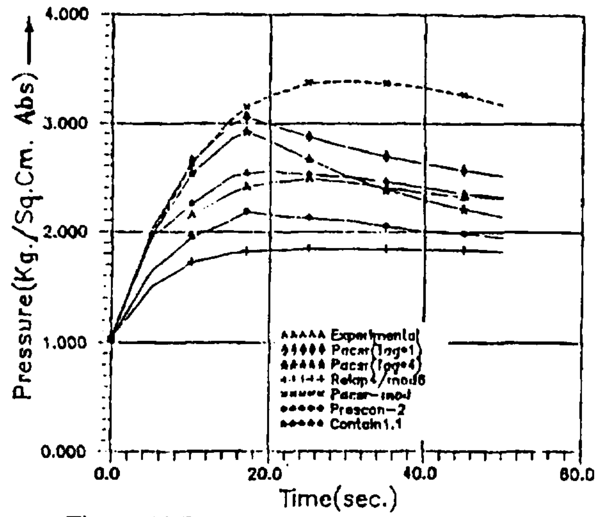

Figure-12 Pressure in compartment No. 13 (HDR containment)

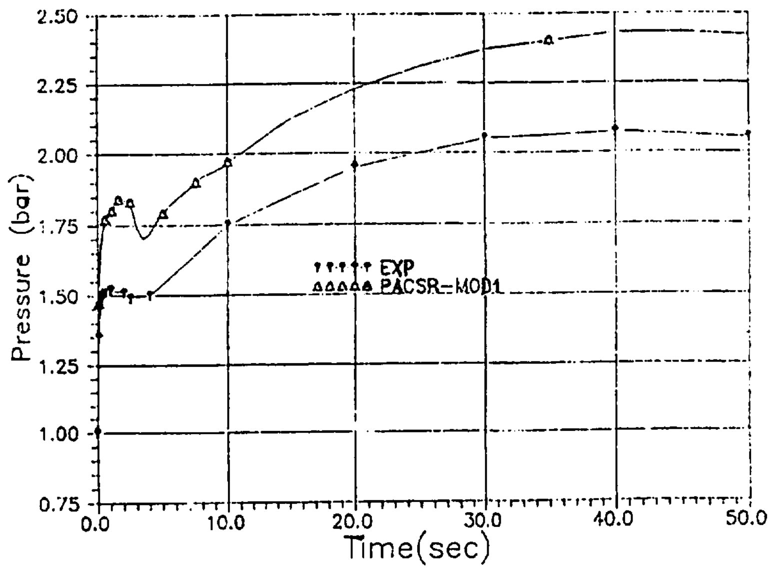

Figure-13 Battelle containment model D15 experiment (Medium term analysis)

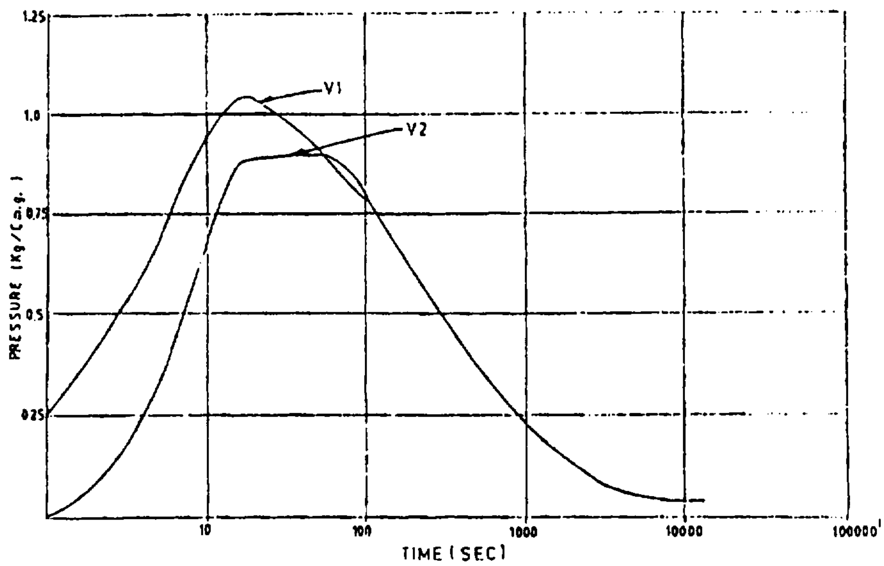

Figure-14 Pressure transient following LOCA

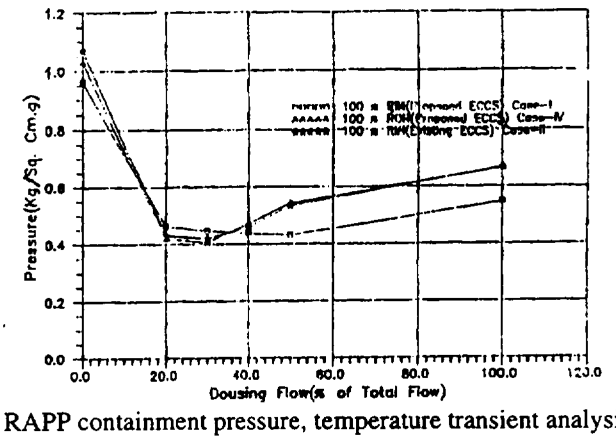

Figure-15 RAPP containment pressure, temperature transient analysis (LOCA) condensation and diffusion. By incorporating momentum equation for the calculation of flow it was made possible to subdivide one compartment by several small compartment connected by virtual juctions. In addition to the provision to model hydrogen, it has been made possible to split air into two components, oxygen and nitrogen, so that modelling for recombiners and hydrogen burn can be done. For validation purposes, this modified code was used to analyse two hydrogen distribution tests [14] carried out in the Battelle Frankfurt containment model test facility in Germany. The results of the analysis are compared with experimental results in Figures 16 & 17. This code has been applied to calculate hydrogen distribution among various compartments of the primary containment of Indian PHWR.

7. ANALYSIS FOR HEAT-UP OF PHT COOLANT/REACTOR COMPONENTS UNDER OFF-NORMAL/SHUTDOWN CONDITIONS

7.1 Computer Code COOLTMP : PHT Coolant Heatup

For certain special maintenance jobs, it may be necessary to interrupt the normal means of decay heat removal from the core. Before permitting such interruption, it is essential to assess whether the decay heat has come down sufficiently to be safely removed by alternate means. For prediction of channel coolant temperature during reactor shutdown with non-availability of normal decay heat removal system, computer code 'COOLTMP' was developed in NPC. The code COOLTMP performs one dimensional energy balance to account for distribution of residual heat in the channel to moderator, end shield cooling water and components, reactor components and heat removal through feeders in the feeder cabinet. It has provision to use constant or variable feeder cabinet and end shield temperatures. In addition, there is a provision to add the heat sink of the boilers. Solution technique used is implicit finite difference scheme. Code COOLTMP has been extensively used to predict temperature rise in PHT coolant under varying operating full power days and cooling period for all the PHWR power stations from RAPS to KAPP. The estimated PHT system temperatures are sensitive to residual power and feeder cabinet cooling conditions.

COOLTMP was validated against available experimental results of RAPS-1 wherein the headers were drained and MAPS-2 where the S/D cooling pumps were stopped and channel coolant temperature measured. Using this computer code channel coolant temperatures are predicted for hottest channel and average core for various residual powers obtained for different operating full power days and subsequent cooling periods. The comparison of calculated results with experimental results is shown in Figs. 18 and 19.

7.2 Computer Codes RCOMP : Reactor Component Heatup

Computer Code 'RCOMP' has been developed for predicting temperature of critical core components and cooling fluids under off normal conditions including Station Blackout. This computer code employs solution of one dimensional energy balance in the residual heat present in various reactor components, moderator and heat removal mechanism through process water/fire water and vault cooling systems. The transient residual heat in various components obtained with respect to time, have been incorporated in the code. Similarly, the heat exchangers for heat removal through process water/fire water has been modelled. The model for heat removal from channel to moderator, end shield, end fitting have been done for a single lattice and a lumped system has been considered for the overall reactor core.

Figs. 20 & 21 show the results of a study carried out with this code to evaluate the temperatures in various reactor components/fluids following Station Blackout in NAPS. While in the first case (Fig.20) no credit is taken for mitigation action, in the second case (Fig.21) effect of fire water injection into the end shield is considered to restrict the temperature rise [15].

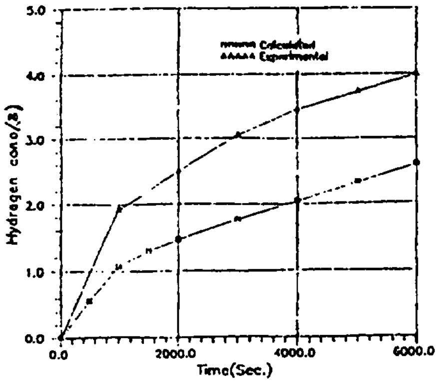

Figure-16 Verification of PACSR against Battelle-Frankfurt hyd. TEST-6 (Compartment No.-CH11)

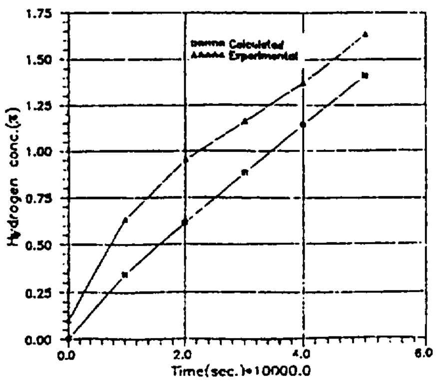

Figure-17 Verification of PACSR against Battelle-Frankfurt hyd. TEST-12 (Compartment No.-CH11)

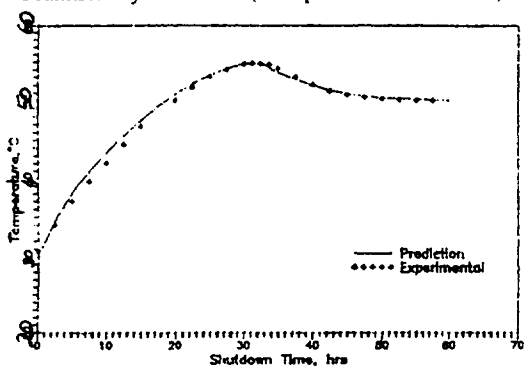

Figure-18 PHTS coolant average temp. estimation for RAPS-1 with no moderator & S/D cooling partial end shield cooling & feeder cooling available during header draining (Expt. conducted on 23.10.82 AT 10:45 hrs at 16FPD)

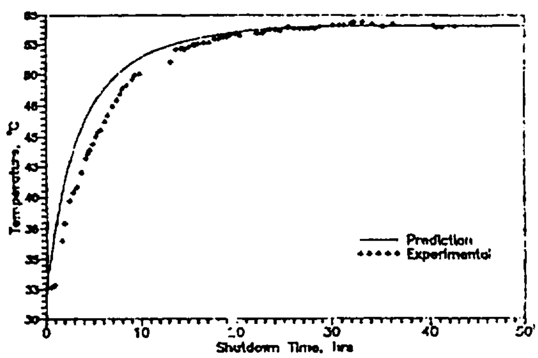

Figure-19 PHTS coolant average temp. estimation for MAPS-2 with moderator end shield and feeder cooling available and shutdown pumps stopped (Unit operated at 75 FP, 120ds cooling)

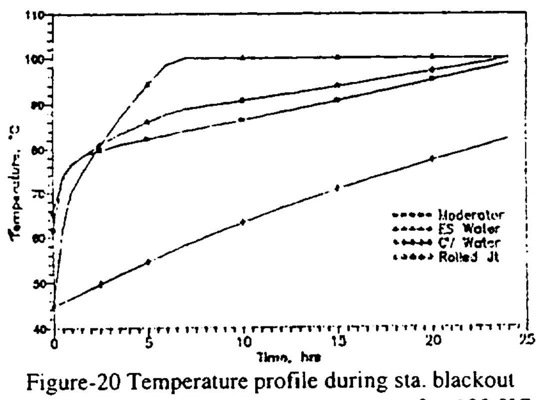

Figure-20 Temperature profile during sta. blackout Case A: End shield water temp. constant after 100.0°C

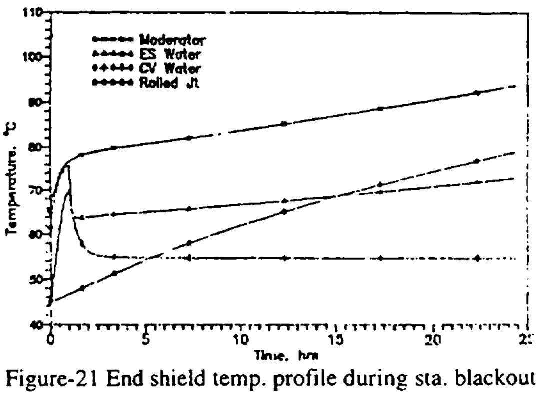

Figure-21 End shield temp. profile during sta. blackout Case B: Firewater injection end shield after 1 hour

8. CONCLUSION

Capability to perform comprehensive safety analysis of NPPs requires the availability of a wide variety of well validated computer codes, as well as experienced experts in using these codes. In India, most of the codes in use for PHWR safety analysis have been developed inhouse, since such codes were not available from outside. Developing a well qualified code which can be used confidently for a variety of situations, is a painstaking and time consuming activity, but has its rewards in that a deep understanding of the code is available within the organisation & modification & extensions as required for specific applications can be done readily. In NPC development of computer codes for safety analysis is a continuing activity aimed at further refinements and additional validations in existing codes, as well as development of newer codes for areas of analysis still not adequately covered by existing codes.

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DEVELOPMENT AND VALIDATION OF COMPUTER CODES FOR ANALYSIS OF PHWR CONTAINMENT BEHAVIOUR

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Abstract

In order to ensure that the design intent of the containment of Indian Pressurised Heavy Water Reactors (IPHWRs) is met, both analytical and experimental studies are being pursued at BARC. As a part of analytical studies, computer codes for predicting the behaviour of containment under various accident scenarios are developed/adapted. These include codes for predicting 1) pressure, temperature transients in the containment following either Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB), 2) hydrogen behaviour in respect of its distribution, combustion and the performance of proposed mitigation systems, and 3) behaviour of fission product aerosols in the piping circuits of the primary heat transport system and in the containment. All these codes have undergone thorough validation using data obtained from in-house test facilities or from international sources. Participation in the International Standard Problem (ISP) exercises has also helped in validation of the codes. The present paper briefly describes some of these codes and the various exercises performed for their validation.

1. INTRODUCTION

The containment structure of Nuclear Power Plant (NPP) acts as the last physical barrier to prevent the release of radioactivity into the environment. Hence, in the event of accidents within the plant, ensuring the safety and integrity of the containment structure assumes considerable importance. In order to ensure and demonstrate the IPHWR containment integrity, both analytical and experimental studies are being pursued at BARC for improving the understanding of the various complex phenomena associated with the containment related accident scenarios. The analytical studies pursued at BARC include development of suitable computer codes and their thorough validation. In-house test data and/or the international databases available in open literature have been used for code validation. In addition, validation through participation, whenever possible, in International Standard Problem (ISP) exercises have also been performed. Besides the codes developed at BARC, a few useful computer codes developed elsewhere are also under study at BARC. This paper presents a brief description of the code development and validation activities being pursued at BARC.

2. THE CONTAINMENT OF INDIAN PHWRS

The PHWRs of standardised design in India use a double containment (Fig. 1), the inner (primary) containment being made of prestressed cement concrete and the outer (secondary) containment being made of reinforced concrete. The annular gap between the two containments is maintained under partial vacuum to prevent leakage from within to the atmosphere. The primary containment can be further subdivided into two accident based volumes called V1 (drywell) and V2 (wetwell). These two volumes are interconnected by the vent system via the vapour suppression pool. The vapour suppression pool, which passively absorbs energy released during LOCA, also helps in the scrubbing of significant fraction of fission products released, if any, in the event of accidents leading to core damage.

3. ACCIDENT SPECIFIC SCENARIOS

The scope of analytical studies pertaining to containment safety encompasses consequences of both, Design Basis Accidents as well as the severe accidents.

3.1. Design Basis Accident (DBA)

The DBA postulated for the containment is the postulated Loss of Coolant Accident (LOCA) involving a double ended rupture in the primary coolant system or in the main steam line, followed by activation of Reactor Protection System (RPS) and the Reactor Safety System (RSS) as intended. The consequences of such an accident are restricted to release of high enthalpy steam-water mixture into the drywell (V1), and then via the suppression pool into wetwell (V2). The effectiveness of containment safety features in limiting the peak pressure and temperature values during such an accident depends upon several parameters as listed below.

- Geometrical configuration of the containment.
- Inter-compartmental gas transport mechanisms through complex shaped flow paths.
- Complex energy absorption processes by conduction, convection and condensation.
- Effectiveness of suppression pool in energy absorption.
- Thermodynamic state of equilibrium/non-equilibrium.

The computer codes to be developed for performing analysis of containments following DBA must ensure that all the above listed parameters and processes are adequately modeled along with the basic sets of the governing conservation equations of mass, momentum and energy.

3.2. Severe accidents

Severe Accidents, also termed as Beyond Design Basis Accidents (BDBA), are defined on the basis of postulation of multiple failures, e.g. LOCA

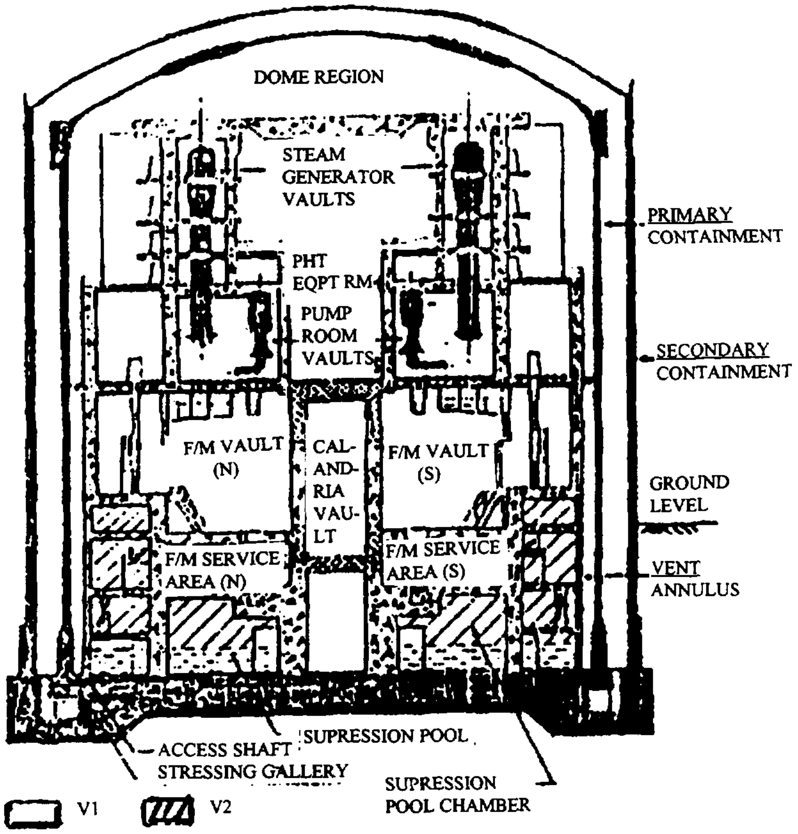

Fig. 1 Typical cross section of 220 MWe reactor building

followed by failure or delay in the activation of RPS and/or RSS. Due to the choice of postulation of initiating events, the progress of severe accidents is generally highly accident specific. Yet, typically the severe accidents might lead to: a) release of large quantities of hydrogen gas into the containment, and b) release of fission products in the form of aerosols into the containment. The associated issues of concern to containment are the following:

- Hydrogen distribution in the containment.
- Deflagration/detonation of hydrogen-air mixtures and the Consequent loads on the containment structure.
- Aerosol behaviour following release from the core into the primary coolant circuit.
- Aerosol transport behaviour in the containment building.

4. CODE DEVELOPMENT AND VALIDATION

The codes developed at BARC include, CONTRAN (CONTainment TRANsients), HYCOMB (HYdrogen COMBustion) and HYRECAT (HYdrogen REmoval by CATalyst). In addition, two other well known codes, viz. TRAPMELT3 and NAUA-MOD5 (obtained from IAEA and NEA data bank), for aerosol behaviour studies in primary coolant circuit and in the containment respectively are also commissioned. All these codes have gone through extensive validation exercises. Brief highlights of some of the validation exercises performed at BARC along with salient features of the codes are presented in the following sections.

4.1. Computer code CONTRAN

The code CONTRAN has been developed for predicting the pressure and temperature transients in the vapour suppression pool type containment following LOCA. The multi-compartment containment geometry is modeled in the code by a network of volumes interconnected by junctions. The detailed formulation of the code accounting for the various aspects/processes listed earlier under 3.1 is described in Ref. [1]. Some of the salient features of CONTRAN are as below:

1. Provision of up to 20 volumes with a maximum of 6 junctions and 3 heat slabs of different materials per volume.
2. 1-D transient heat conduction model using ither uniform or non-uniform grid, for structural heat transfer calculations.
3. Empirical models for convective heat transfer from steam-air mixture to the structures inside the containment and from the outer wall of the containment to the atmosphere.
4. Empirical models for condensation heat transfer to the concrete/steelstructures within the containment.
5. Vent clearing transient model for vapour suppression pool based on 1-D momentum equation [2].
6. Differential pressure driven hydrogen transportation model with Adiabatic, Isochoric, Complete Combustion (AICC) model for hydrogen.

4.1.1. Validation of CONTRAN

The code has been extensively validated using the test data from various sources. These include, a) 1/10th scale model containment test facility at Kalpakkam, India, b) BFC test facility, Germany and c) CSNI Numerical Benchmark exercise.

a) Test data from Containment Test Facility, Kalpakkam

The experimental test facility (Fig. 2) at Kalpakkam, (India) is a 1/10th scale model of the vapour suppression pool type multi-compartment IPHWR containment in which a large number of tests simulating blowdown from a PHT model were carried out to study the influence of various governing parameters on containment behaviour [3,4,5]. The code CONTRAN has been extensively validated using these test data. One such typical result showing comparison between code predictions and the experimental data in respect of pressure transients in V1 and V2 for the test M4L0 [1] is shown in Fig. 3. The designated test was performed to simulate a large break LOCA in MAPS (Madras Atomic Power Station) reactor.

It can be seen from Fig. 3 that the first pressure peak of 121 kPa, corresponding to pressure buildup in the fuelling machine vault till the rupture of blowout panel has been predicted correctly. The subsequent pressure transients and the value of the second pressure peak of 116 kPa (due to pressurisation of V1 until vent clearing) seem to be on the conservative side with respect to the test data. The predicted early initiation of pressure transients in V2 and higher values of pressure than the experimental values are due to

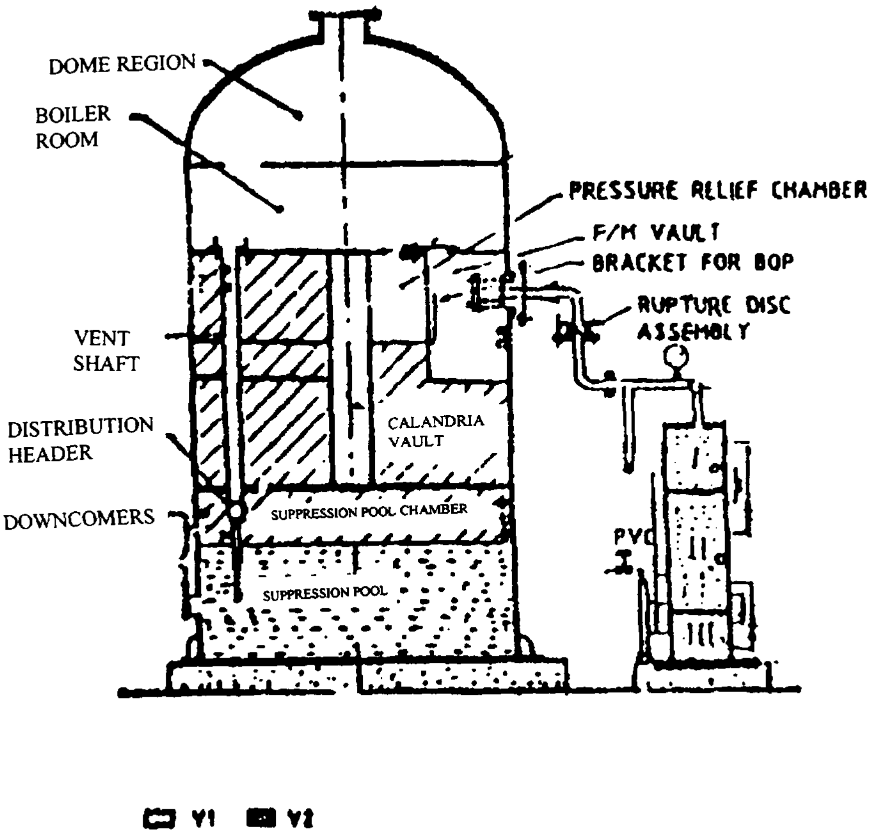

Fig. 2 One tenth scale vapor suppression pool containment experimental facility

higher predicted pressures in V1. The comparison of CONTRAN predictions with data from other tests are also of similar nature and are generally found to be satisfactory.

b) Validation using test data from BFC Test Facility, Germany

The Battelle-Frankfurt Containment (BFC) test facility in Germany (Fig. 4), is a 1/4th scale model concrete containment with a total free volume of about 580 cu. m. [6]. The containment model has several compartments with a provision to select different multi-compartment configurations. Several containment related tests involving response to simulated blowdown from PHT system, hydrogen distribution, aerosol dispersion and performance of catalytic recombiners and igniters are reported to have been carried out in the test facility. Validation of CONTRAN has been successfully carried out [7,1] using the data from the two blowdown tests designated as D1 and D15, reported in literature [8]. The CONTRAN validation results for the test D1 are presented here.

The test D1 involved blowdown of saturated steam with an average enthalpy of 2774.2 kJ/kg in room R6 for 3.0 seconds. The results comparing the CONTRAN predicted pressure transients in the break compartment R6 with the test data up to 2.5 seconds are shown in Fig. 5. The corresponding predictions of COBRA-NC for this test, reported earlier, are also shown in Fig. 5 for comparison. The CONTRAN predicted pressure transients are observed to

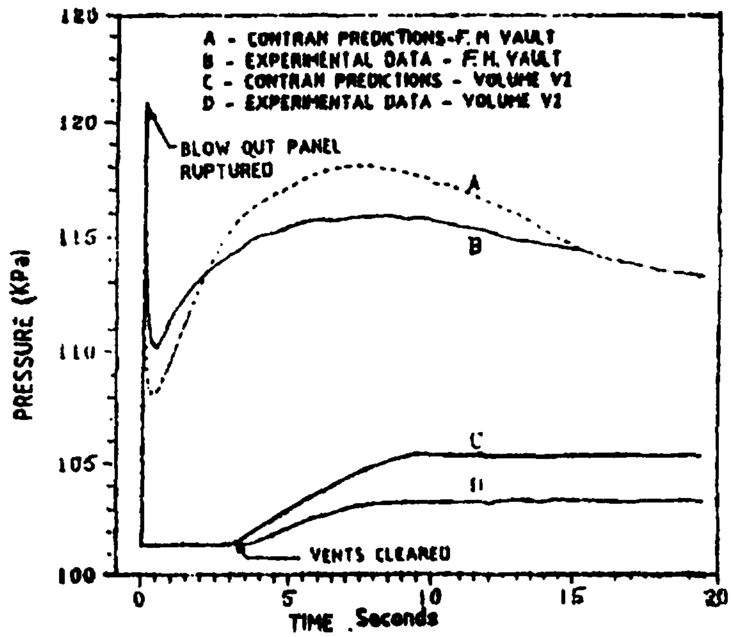

Fig. 3 Pressure transients in V1 and V2 for test M4L0
in one-tenth scale model PHWR containment test facility

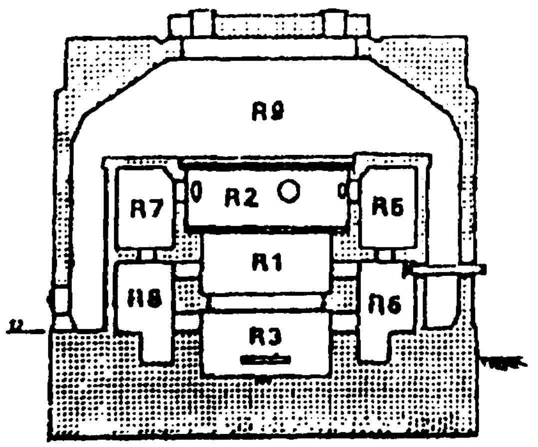

Fig. 4 One-fourth scale BATTELLE Frankfurt containment model

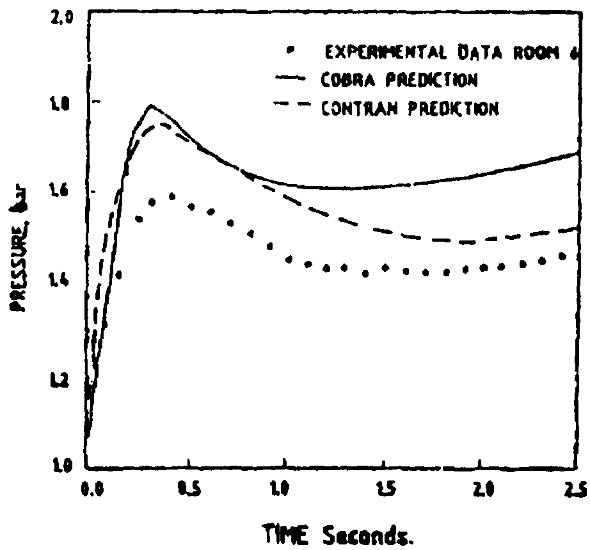

Fig. 5 CONTRAN predictions for ROOM 6 of BFC model

be conservative with respect to the test data. CONTRAN predictions show better agreement with the experiment.

c) CSNI benchmark exercise:

The code CONTRAN has also been successfully validated [9] using the data from the benchmark test [10] devised by the Committee on the Safety of Nuclear Installations (CSNI) of the Organisation for Economic Cooperation and Development (OECD). The benchmark exercise was devised to test the containment analysis computer codes for numerical accuracy and convergence errors in the computation of mass and energy for fluid and in the computation of heat conduction in structural walls.

The benchmark test model, shown schematically in Fig. 6, consists of a single fluid volume into which steam and water are injected and from which heat is transferred to a single concrete wall. Based on the semi-infinite solid heat slab with a specified constant heat transfer coefficient on the inner wall surface, analytical solutions were provided to arrive at the transient wall surface temperature, total pressure and steam partial pressure in the containment. The code CONTRAN successfully predicted all the three transients with remarkably close agreement. Fig. 7 shows the comparison of predicted transients for total containment pressure and structural wall surface temperature with the test data.

d) Hydrogen distribution calculations using CONTRAN

As a first step towards enabling hydrogen distribution calculations, the capability of accounting for differential pressure driven inter-compartmental hydrogen transport has been provided in CONTRAN. It is well acknowledged here that modeling of hydrogen transport needs detailed formulation of many other complex mechanisms. The capability of the code was validated [11] in a limited way using the data of two hydrogen distribution tests performed in BFC test facility [6]. Validation results for one of the tests (Test-6) are described here.

The Test-6 was performed in a 2-room configuration (Fig. 8) of the BFC test facility with total free volume of the compartments as 68 cu.m. A mixture of hydrogen and nitrogen in the ratio of 66:34 (% v/v) was injected at the bottom of lower compartment R1 at a rate of 0.29631 g/s at 292 K for the first 18 minutes and subsequently reduced to 0.12357 g/s for next 108 minutes. The CONTRAN predictions of hydrogen concentration transients in room R-1 and those predicted by COBRA-NC are shown in Fig. 9. along with the test data. The transient trends predicted by CONTRAN can be seen to be identical to the test results. The differences in hydrogen concentration values can be due to the assumption of homogeneous distribution of hydrogen within a compartment and non-consideration of buoyancy force. The validation results for the second test (Test-12) also resulted similar conclusions.

The code CONTRAN, is under extensive use for studying the performance of the containment of IPHWRs for various LOCA/MSLB scenarios [12].

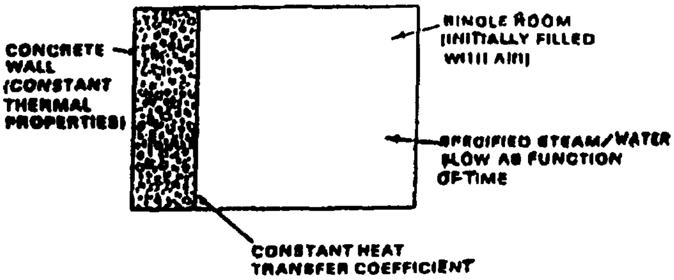

Fig. 6 CSNI-benchmark problem model

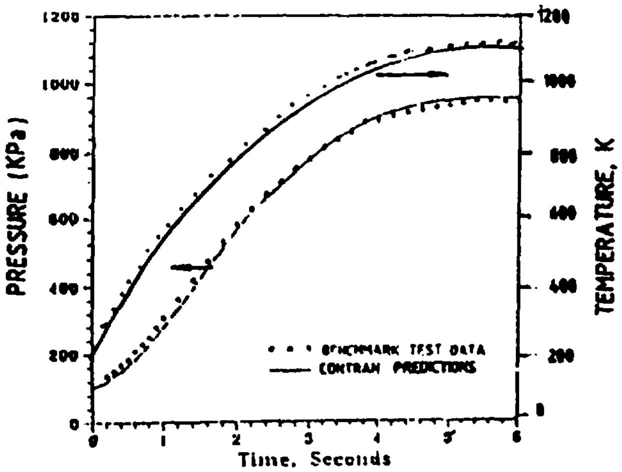

Fig. 7 Total containment pressure and structural wall surface temperature transients

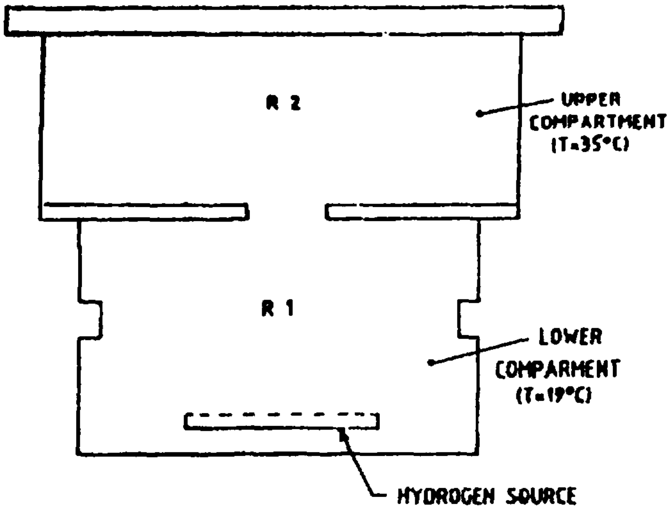

Fig. 8 Compartment arrangement for BFC TEST-4 for hydrogen distribution studies

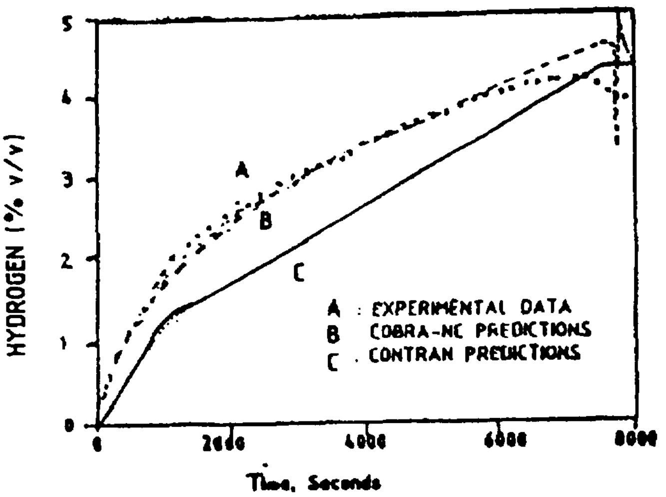

Fig. 9 Hydrogen concentration in ROOM-1 for TEST-6 on BFC model

4.2. Computer code HYCOMB

Due to very large volume of containment of Indian PHWRs, the global average hydrogen concentration assuming 100 % clad oxidation is estimated to be lower than the detonable limit of hydrogen-air mixture. In order to analyse deflagration phenomena in the containment, (although highly unlikely) computer code HYCOMB has been developed. The model in HYCOMB considers deflagration of hydrogen in a single closed compartment for predicting the resulting pressure and temperature transients within it. The peak deflagration pressure and temperature calculations are on the basis of AICC model and the transient deflagration calculations are on the basis of laminar burning velocity model [13] with a provision to extend to the turbulent burning velocity model using empirically available turbulence factors.

4.2.1. Validation of HYCOMB

The code HYCOMB has been validated using experimental data of Kumar et al. [14], wherein several hydrogen combustion tests are reported to have been carried out in a 6 cu.m spherical vessel. Fig. 10 shows the comparison of combustion pressure transients predicted by HYCOMB with the experimental data for the combustion test at 29.5 % (v/v) hydrogen concentration. The code HYCOMB calculates the upper limit of burning velocity on the basis of calculated AICC temperature for the burnt gas mixture (case-a in Fig. 10). The code has also an option of calculating the burning velocity based on mean temperature of the burnt and unburnt species for calculating the pressure transients (case-b in Fig. 10). The close agreement of case-a with the test data shows that combustion of hydrogen-air mixture corresponding to near stoicheometric compositions follows the AICC model. For the lower concentrations of hydrogen, the experimental data is found to fall between the two predictions made by HYCOMB [15] for the above mentioned two approaches.

4.3. Computer code HYRECAT

There are currently four approaches being investigated worldwide for hydrogen mitigation, viz.

- inertisation of the hydrogen-air mixture,
- deliberate ignition of combustible mixtures,
- catalytic recombination of hydrogen with oxygen [16],
- combination of deliberate ignition and catalytic recombination [17] or simultaneous inertisation and catalytic recombination [18].

Among these, the catalytic recombination method seems to be very promising due to its hydrogen removal efficiency even at low hydrogen concentrations and in presence of steam as demonstrated by the German experimental studies. BARC has successfully developed a special type of activated platinum catalyst impregnated on polyester cloth and the laboratory tests have demonstrated its satisfactory performance [19]. Large scale experiments are planned for carrying out engineering feasibility studies and to generate more performance data for the device using this approach [19].

Simultaneously, a computer code HYRECAT has been developed [20] for predicting the performance of the proposed catalytic device. The model is based on the solution of equations for mass and energy conservation within a homogeneously mixed hydrogen-air-steam mixture undergoing oxidation of hydrogen on the catalyst surface. The kinetics of catalytic oxidation of hydrogen is described by Arrhenius type empirical rate equation, (on the basis of unit mass of catalyst or unit area of the catalyst coated surface [21,22]). Appropriate heat transfer mechanisms are modeled for the dissipation of exothermic heat of reaction to the surroundings.

4.3.1. Validation of HYRECAT

The validation of HYRECAT has been carried out [20] using data from two of the several laboratory tests performed at BARC. Results of only the second test are presented here. The test was performed in a 22 litre stainless steel vessel with a catalyst coated curtain of known dimensions and known catalyst loading (i.e. mass of catalyst per unit area) mounted within and initial hydrogen concentration of 5.1 % (v/v). The variation of the total pressure in the vessel was recorded continuously. This is taken as an indication of the consumption of hydrogen and oxygen due to oxidation reaction.

The code HYRECAT was used to simulate the test and predict the data in respect of total pressure, bulk gas temperature and the catalyst surface temperature. Excellent agreement has been observed between the predictions of HYRECAT and the measured total pressure in the vessel as seen from Fig.11. More detailed results of validation are presented elsewhere [20].

Subsequent to the validation of the HYRECAT, the code was used [23] to pre-calculate the transients for the proposed engineering scale tests to be conducted at BARC. These tests would give additional data to validate the code further.

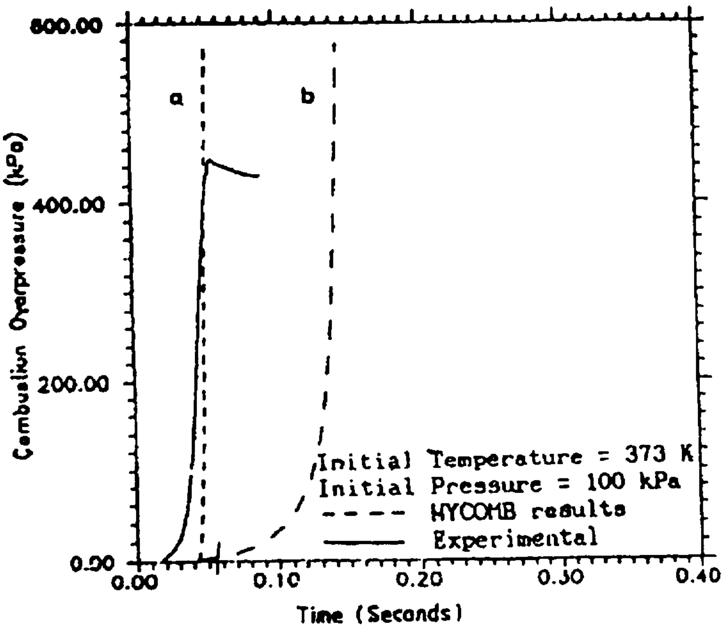

Fig. 10 Combustion pressure transients predictions with experimental data of Kumar et.al.-14 for hydrogen concentration of 29.5 %

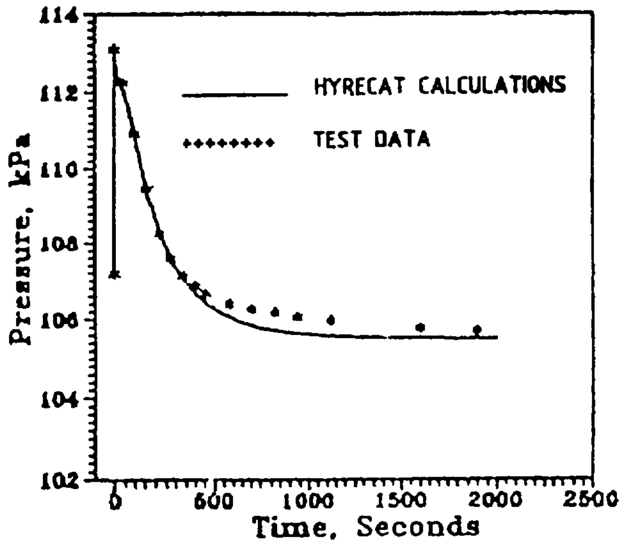

Fig. 11 Pressure transients in EXPT-2 test vessel (22 litre volume)

4.4. Computer codes TRAPMELT3 and NAUA-MOD5

Presently, the computer codes TRAPMELT3 and NAUA-MOD5 are under use for studying the aerosol behaviour in the primary coolant system and in the containment respectively. Both these codes are well documented in open literature [24,25]. Validation of both TRAPMELT3 and NAUA-MOD5 has been done by way of participation in the ISP-34 exercise held under the auspices of OECD and conducted in the FALCON test facility, Winfrith Technology Centre, Dorchester, UK. Results of validation of these two codes are presented briefly in the following section.

The ISP-34 [26] exercise was intended to test and assess models used for fission product transport and deposition behaviour within the primary circuit and the containment. Some of the important parameters for participants to predict were the aerosol deposition profiles and composition, key chemical species and reactions, evolution of suspended material concentrations, and the effects of steam condensation on to the aerosols and particle hygroscopicity for the following two tests:

- FAL-ISP-1 (Open Test): Low relative humidity in the containment, with relatively high particle concentrations of multi-component aerosols involving different chemical species,
- FAL-ISP-2 (Blind Test): High containment relative humidity and low particle concentration.

The tests were performed in the FALCON test facility shown schematically in Fig. 12. The facility comprised of core region, primary circuit piping and the containment. Simulant fuel samples clad in zircaloy-4 were placed in a silica vessel simulating the core region. Heating of the fuel was achieved in a 40 kW induction furnace. Helium gas as a carrier medium is admitted at the bottom of the silica vessel and aqueous boric acid solution can be introduced on the heated sample. The thermal gradient tube and the stainless steel pipe simulated the primary circuit. The containment was a 0.3 cu.m stainless steel chamber connected to the primary circuit.

4.4.1. Validation of TRAPMELT3

While the final comparison report [26] gives all the detailed results, only a few comparative results are presented for the FAL-ISP-1 test. Fig. 13 shows the comparison of TRAPMELT3 predictions of concentration profiles for strontium in the silica tube with the test data. Similar remarkably good agreement was predicted for boron. However, the predictions of concentration profiles of other aerosol species represented the trends satisfactorily but deviated from the magnitudes significantly.

4.4.2. Validation of NAUA-MOD5 [26,27]

Comparison of predictions of NAUA-MOD5 in respect of air-borne mass concentration transients for caesium in the containment with the test data of FAL-ISP-1 is shown in Fig. 14. Similar satisfactory agreement was also observed for boron. However, deposition of the aerosols on the vessel walls was not well predicted by NAUA.

5. CONCLUDING REMARKS

The various codes under use at BARC for analysing containment related scenarios following DBA or severe accidents have been briefly described. The results of the validation exercises performed to test the code capabilities have also been described. The code CONTRAN for predicting thermal hydraulic transients in the containment has been validated extensively. The codes for hydrogen and aerosol behaviour have gone through a reasonable degree of validation.

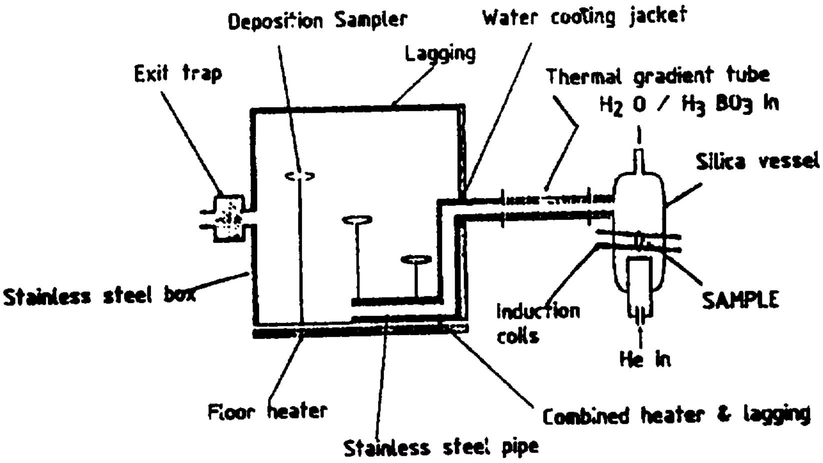

Fig. 12 A schematic of the FALCON test facility

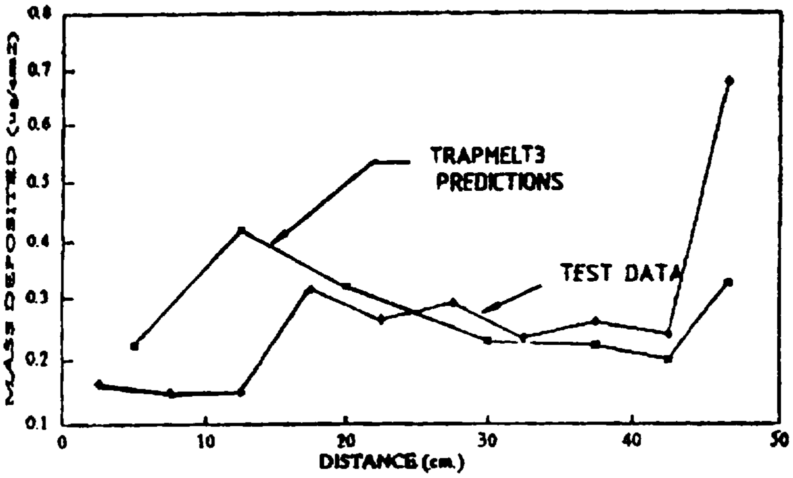

Fig. 13 Comparison of Sr deposition data for FAL-ISP-1 test with TRAPMELT3 predictions

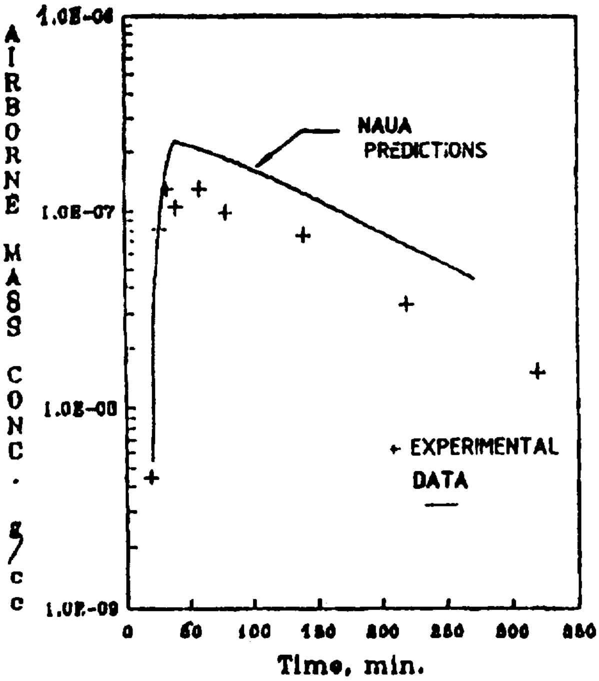

Fig. 14 Comparison of air-borne Cs mass concentration transients with NAUA-MOD-5 predictions for FAL-ISP-1 test

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REACTOR ACCIDENT ANALYSIS

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A STUDY OF THE INDIAN PHWR REACTOR CHANNEL UNDER PROLONGED DETERIORATED FLOW CONDITIONS

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Abstract

The Indian Pressurised Heavy Water Reactors (PHWRs) of 220 MWe capacity have 306 fuel channels in the primary heat transport (PHT) system. Each channel consists of 12 fuel bundles contained in a pressure tube which is surrounded by a calandria tube. The calandria tube is surrounded by cold moderator. The primary coolant flows inside the pressure tube containing the 19 rod fuel bundles. There are several postulated accident scenarios where a quick, large voiding of certain sections of the channel is expected. The fuel temperatures in these sections rise initially because of the stored heat in the fuel and the decay heat of the fuel. Later, at higher temperatures, the metal water reaction between steam and zircaloy clad also produces heat. The modes of heat removal are convective heat transfer to the steam present in the channel and heat transfer to the moderator through the combination of conduction, convection and radiation heat transfer processes. The metal water reaction has a significant effect on the peak cladding temperatures reached by the fuel. The reaction rate varies exponentially with clad temperature. In an accident involving total voiding of the channel, clad temperatures are significantly influenced by the steam flow rate. At higher rate it leads to more reaction and hence more heat generation. On the other hand it leads to better convective heat removal.

Analysis of the above scenario has been carried out using the computer code HT/MOD4 and presented in this paper. The hottest channel of an Indian PHWR has been analysed for its hottest bundle. The stored heat and decay heat generation in different fuel rods have been accounted for. The metal-water/steam reaction is simulated in all zircaloy components. The analysis considers conduction, convection and radiation heat transfer in different components of the channel. Suitable assumptions have been made. The results obtained have been discussed from the point of view of safety

1.0 INTRODUCTION

The Primary Heat Transport System (PHTS) of a Pressurised Heavy Water Reactor (PHWR) of Indian design consists of inlet headers from which coolant is fed to the channels in the reactor core through feeder pipes. As the coolant flows through the core it picks up heat generated in the fuel rods. The coolant then flows through feeder pipes to the outlet header. The hot coolant from the outlet header passes through the steam generator, on the secondary side of which steam is generated. The cold heavy water comes to the other inlet header through the pumps. The coolant path in the PHTS is in the form of a figure of eight, with the coolant flowing in opposite directions in adjacent channels. The largest diameter pipes in the PHTS are the inlet and outlet headers.

Unlike other water cooled reactors, the Indian PHWR has ahorizontal reactor core [1]. The pressure boundary in the core consists of 306 pressure tubes. Each pressure tube houses 12 short fuel bundles. Each fuel bundle consists of 19 fuel pins. The pressure tube is surrounded by a thin calandria tube.

During an accident in a water cooled reactor the heat removal capability may get impaired, thusresulting in fuel damage and release of radioactivity. For e.g. in PHWRs a LOCA coupled with unavailablity of emergency core cooling system may have a potential for large release of radioactivity to the containment [2]. The effects of such accidents are predicted using suitable analytical models. The models chosen depend on the type of analysis required.

The fuel elements of the channel are strongly coupled thermally to the coolant through convective heat transfer under normal operating conditions and are very weakly coupled to each other. But following an accident leading to loss of coolant, convective heat transfer reduces and fuel temperatures rise even when the reactor is shut down. A totally voided channel experiences significant heat exchange through radiation. The thermal coupling among fuel elements becomes strong. The pressure tube temperature also rises. However, the temperature of the calandria tube is not affected significantly since it is convectively cooled by the moderator. A rise in the temperature of pressure tube may lead to deterioration in its mechanical properties. This may lead to deformation (balloning/sagging) of pressure tube. The mode of deformation and the temperature at which deformation occurs would depend on how the coolant tube's axial motion is restricted. The hot pressure tube may deform and touch the calandria tube leading to a temporary temperature spike in the calandria tube.

In a partially voided channel, the vapour and liquid phases may separate out. The temperatures of the fuel elements exposed to steam rise while the elements submerged in coolant remain at lower temperatures. This leads to large circumferential temperature gradients in the pressure tube and fuel elements.

1.1 Total Voiding of The Channel

A channel may void totally for accidents such as LOCA coincident with total failure of the emergency core cooling system. In such a case, the fuel rod temperatures rise and start radiating heat to the pressure tube. The pressure tube temperature rises. The pressure tube transfers part of the heat to the calandria tube by (a) conduction through the gas in the annulus between the two tubes, and (b) radiation to the calandria tube. Finally the calandria tube transfers the heat to the moderator by convection.

A rise in temperature leads to a rapid deterioration in the mechanical properties of the zircaloy pressure tube. Thus, at higher temperatures the pressure tube deforms. There are two possible modes of deformation. One mode of deformation is ballon-ing. This occurs when the pressure tube reaches high temperatures when internal pressure of the tube is still high. On ballooning it touches the calandria tube. The mode of heat transfer changes from radiation to conduction heat transfer, thus enhancing the heat transfer. The fuel temperatures continue to rise. At a certain temperature the fuel clad fails and the fuel bundle slumps on to the pressure tube. Additional contacts are established between fuel rods and pressure tube leading to further enhancement of heat transfer to the ultimate heat sink (moderator).

The other mode of deformation is sagging. This occurs when the channel internal pressure of the tube is already low by the time it gets heated to higher temperatures. Subsequently the fuel bundle also slumps as the fuel clad fails. This again leads to further enhancement of heat transfer to the moderator, the ultimate heat sink.

1.2 MECHANICAL CONSIDERATIONS

The prefered mode of deformation of coolant channel should lead to maximum heat transfer contact area between coolant tube and calandria tube. The two independent modes of deformation of coolant tube are sagging and ballooning. The sagging deformation of a coolant tube is essentially due to bending deflection caused by fuel bundle weight. A contact with the calandria tube due to this mode of deformation can initiate either at a point between the two garter springs or at a point between a garter spring and rolled joint. The ballooning mode of deformation is essentially due to circumferential membrane stretching of the coolant tube caused by remaining PHT pressure. Due to its localised nature of deformation, the entire length of the coolant tube is susceptible to ballooning mode of deformation except the zone near to garter springs. The conusion which can be drawn through this discussion on the deformation pattern, is that the ballooning mode of deformation would generally lead to much higher contact heat transfer area than the sagging mode of deformation. Hence a ballooning mode of deformation is preferable over sagging mode. Such scenario may be achieved by delaying sagging mode during the heatup time of the coolant tube under accidental conditions, provided there is sufficient internal PHT pressure following LOCA. One of the important considerations during ballooning mode of deformation is the possibility of rupture of coolant tube. This possibility further enhances in case there is an initial defect in the coolant tube inner surface. In the following section, we discussed this issue of possibility of pressure tube rupture during ballooning under the presence of a flaw. The subsequent section of the present paper deals with the sagging behaviour of a coolant tube. The effect of coolant tube axial restraint on the sagging temperatrure is also highlighted.

1.2.1 BALLOONING MODE OF DEFORMATION OF A COOLANT TUBE

As discussed above, ballooning mode of deformation of the coolant tube is a prefered mode of deformation. This mode of deformation is likely to provide more heat transfer area than the sagging mode of deformation. However, there exists always a possibility of coolant tube rupture during ballooning. The probability of coolant tube rupture is high in the locations having defects. The defect may be caused due to unfavourable sliding of fuel bundles on the coolant tube inner surface during refuelling operation or following earthquake. A double ended rupture in coolant tube may lead to propagating damage. This may cause axial flying out of coolant tube into fuelling machine vault in case of axially free tube. This may also lead to high enthalpy jet in the calandria after rupturing thin calandria tube. This jet may cause further damage to the surrounding coolant channels.

The rupture of a coolant tube during ballooning essentially depends upon the three parameters, namely, the flaw size, the internal pressure and the circumferential temperature profile. Different combinations of these parameters can make the size of the flaw critical at the time of ballooning and the coolant tube may rupture out. We have carried out a prametric study to obtain the critical combinations of these parameters.

1.2.2 Analysis details

Geometrical modelling- In this analysis, a cross section of the tube is modelled as a two dimensional case. Finite element technique is employed for this purpose. There are 180 isoparametric elements with 560 nodes. The mesh is modified to accomodate a particular size of the flaw to have large number of elements near the crack tip.

Location of flaws- Coolant tubes in a PHWR may have part through flaws. These flaws may be generated during manufacturing or during the movement of the fuel bundles while refuelling. A severe earthquake may lead to sliding of the fuel bundles inside the channel and may cause part through flaws. These flaws act as stress risers during ballooning of the tube and may initiate rupture. The chances of having a flaw at the lower portion of the tube is more due to fuel bundle weights. On the other hand the maximum temperature along the circumference of the pressure tube during severe accident is likely to occur at the top surface of the tube due to channel stratification. However, in the present analysis, the location of the flaw is conservatively assumed at the maximum temperature point.

Temperature profile- The temperature profile following severe accident in a coolant tube can be known through a detailed thermal-hydraulic calculations. These profiles may change case by case depending upon the severity of the accident. In the present parametric analysis, we have assumed a closed form expression for circumferential temperature profile as

$$T(\theta) = Tmin + \Delta T[\,1 - (\theta/\pi)^n]$$

Here Δt is the circumferential temperature drop and n is an exponent. Different computed temperature profile using thermal hydraulic codes can be simulated using this expression by assigning different values to Tmin, DT and n. It may be noted here that the lower value of n localises the high temperature zone near to θ =0. Thus uniform temperature is obtained asymtotically by using a large value of n. In the present parametric study, we have varied DT from 100 Deg.C to 300 Deg.C. The exponent n has been varied to obtain linear, quadratic, cubic, quartic and uniform temperature profiles. The temperature is assumed to increase at a rate 5 Deg.C/sec to determine the creep properties of zircalloy.

Pressure loading- The temperature at which ballooning occurs in a coolant tube, depends on the internal pressure during ballooning. In a typical PHWR, the normal primary operating pressure is 10MPa. With the occurance of loss of coolant accident, the pressure decreases with a rate depending on the size and the location of the break. In the present analysis on ballooning, different coolant channel pressure ranging from 10MPa to 0.5MPa have been considered.

Computer code THESIS- The present analysis has been performed by using inhouse code THESIS (THErmal elastic-plastic AnalySIS), developed by the present authors. This is a 2-D/axisymmetric finite element code with the capability of considering geometric and material nonlinearities. The material nonlinearities include elastoplasticity and thermoplasticity. The capability of the code has further been enhanced to analyse viscoplastic behaviour of materials. Four different general forms of creep laws, such as, power, exponential, strain hardening and time hardening, have been implemented in the code.

High temperature creep model- The progress of ballooning in a coolant tube is essentially due to flow of material at high temperature dominantly under internal pressure. This phenomenon can be modelled by using a high temperature creep model of Zr-2.5 wt% Nb. In the present study, we have used the creep model developed by Shewfelt [1984]. This model has been developed essentiallly through ballooning experiment on Zr-2.5 wt% Nb tubes at high temperature. The creep equations are given as follows.

$$E = \exp(-36600/T)$$

$$F = \exp(-29200/T)$$

$$G = \exp(-19600/T)$$

$$H = (T-1105)^{3.72}$$

Creep rate equation for 450 Deg.C to 850 Deg.C-

$$d\varepsilon/dt = 1.3\times10^{-5}\,\sigma^{9}\,E + 5.7\times10^{7}\,\sigma^{1.9}F\,/\,(1 + 2\times10^{10}\int Fdt)^{0.42}$$

Creep rate equation for 850 Deg. C t0 1200 Deg. C-

$$d\varepsilon/dt = 10.4\sigma^{3.4}G + 3.5\times10^{4}\sigma^{1.4}G\,/\,(1 + 2.74\int GH\,dt)$$

Where t1 is the time when T=700 Deg.C and t2 is the time when
T=850 DegC.

Rupture criteria- With the progress of ballooning, the local strain at the tip of the flaw increases. It is conservatively assumed that the rupture occurs when the local effective strain at the tip of the flaw reaches a value 100%.II

1.2.3 Analysis Results

The ballooning analysis has been carried out for various assumed temperature profiles characterised by different combinations of DT and n. Three values of DT have been assumed. These are 100 Deg.C, 200 Deg.C and 300 Deg.C. Five different orders of profiles have been considered by changing the value of n. These are uniform, quartic, cubic, quadratic and linear. The various sizes of flaws have been considered ranging from 5% to 30% of the tube thickness. The contact between the calandria tube and coolant tube is established, once the average circumferential strain in the coolant tube reaches 18% during ballooning. The computations are done to calculate the local effective strain at the tip of the flaw at 18% average hoop strain for various combinations of the flaw size, the temperature profile and the internal pressure. The results of each combination of these parameters are then used to calculate the critical flaw size, below which, coolant tube can come in contact with the calandria tube without rupture. These results are plotted in Figs A to C for the three different values of DT.

1.2.4 SAGGING MODE OF DEFORMATION OF COOLANT TUBE

The sagging analysis of a coolant tube at high temperature is essentially a thermo-mechanical problem with high temperature creep involving large deformations. Some of the complexities involved with this problem are material properties variation with temperature, multidirectional high temperature creep phenomena, large scale yielding of Zircalloy tube with the progress of yield front in thickness as well as axial directions, increase in reactions at garter spring locations with the progress in sagging, consideration of geometric nonlinearities due to inplane compressive stresses in case of axially restrained tubes, etc. Hence it is necessary to employ a powerful numerical technique, such as finite element, to account for these complexities in the analysis. In the present work a computer code TABS has been developed to suit the present study based on bending theory of finite element formulation. The code uses nine noded heterosis/ eight noded degenerate shell bending elements for modelling. The nonlinear variation of stresses along the thickness as a result of progress in plastic front is considered using layered approach. The stress is assumed to be constant over a layer and number of layers is an user's choice depending upon the available computational power. Plastic flow equation is solved by using Prandtl-Reuss flow rule, von-mises yield criteria and isotropic strain hardening. Minimisation of residuel unbalance load is done by using modified Newton-Raphson iterative procedure. The code is also modified to consider thermal stresses due to in plane as well as across the thickness temperature variation. The temperature variation across the thickness is considered by specifying temperature for each layer, which is assumed to be constant over the layer. The pseudo thermal load vector considering the temperature rise and the extent of yielding for each layer is calculated and assembled with the other load vectors of the different layers of a element, to obtain elemental load vector. Beside elastic, plastic and thermal strains, the code is capable of considering strain due to change in material properties with temperature and creep strain. The code is tested against large number of case studies to evaluate its different analysis capabilities.

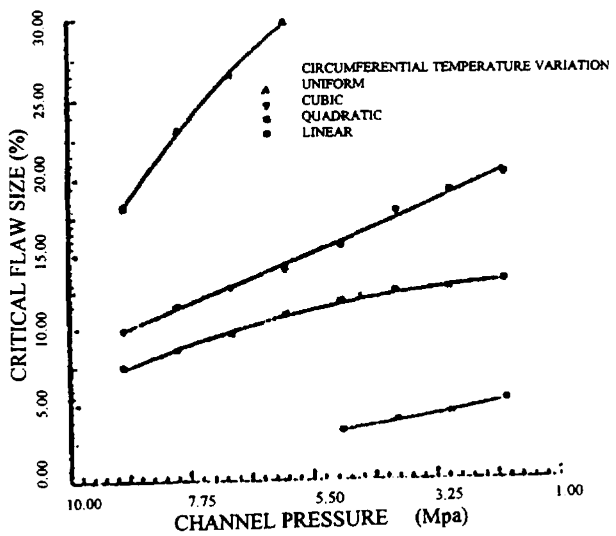

Fig. A Analysis of coolant tube ballooning; Circumferential temperature difference = 100 °C

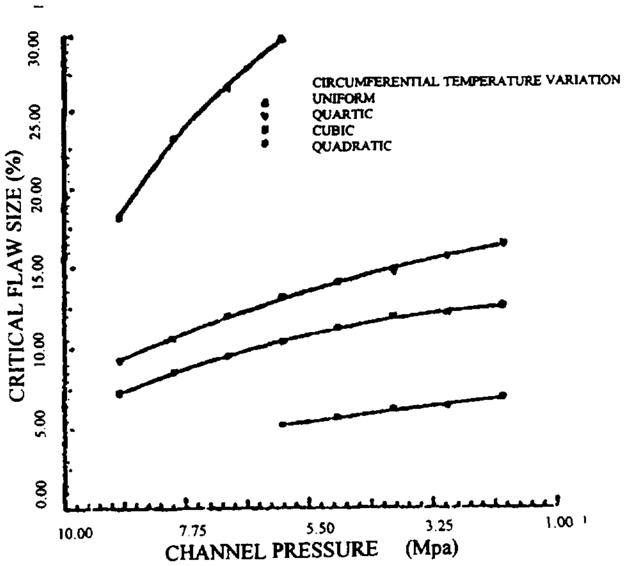

Fig. B Analysis of coolant tube ballooning; Circumferential temperature difference = 200 °C

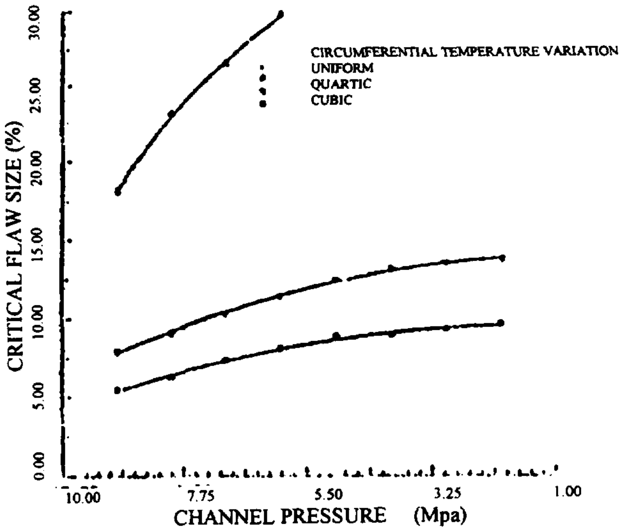

Fig. C Analysis of coolant tube ballooning; Circumferential temperature difference = 300 °C

1.2.5 High Temperature Creep Equation of Zircalloy Tube Dominantly under Bending Moment

One of the important characteristics to be modelled during sagging of the coolant tube is the material creep at high temperature. It is well known that the creep properties for Zircalloy is directional dependent. In our earlier study on ballooning of coolant tube, we used the high temperature creep model by Schefelt,1984. It may be noted here that this model was developed assuming a Zircalloy tube dominantly under ballooning mode of deformation. However, in the present analysis on sagging behaviour of Zircalloy tube, we have used the creep model NIRVANA (AECL-6412).

1.2.6 Analysis Procedure

One-fourth of the coolant tube has been modelled assuming symmetricity with respect to two vertical planes. The effect of garter springs have been modelled by considering equivalent stiffness at the appropriate locations. The Zircalloy property data base, ranging from operating temperature to sagging temperature, are used from MATPRO-11. These properties are modulus of elasticity, poisson's ratio, coefficient of thermal expansion and stress-strain relations. The model is first loaded with fuel bundle weights at the operating temperature. Then the temperature gradient varying with time is imposed on the model. This temperature data base is obtained by using corresponding thermal-hydraulics codes in a seperate analysis. One of the typicalities of this temperature data base is that the temperature value is constant over a significant length of the coolant tube at the centre region. This value drops down to operating temperature near to both the rolled joints. The gap between the coolant tube and calandria tube is monitored by calculating corresponding deformation of calandria tube analytically based on reaction at garter spring location. The computation is terminated once the gap is closed.

1.2.7 Evaluation of Sagging Behaviour of Axially Restrained and Axially Unrestrained Pressure Tube

Axially restrained and unrestrained coolant tubes form two seperate school of thoughts about their relative merits. The axially unrestrained tubes are free from any axial stresses, which may be induced due to thermal expansion and creep with the time of reactor operation. However, in case of double ended rupture of an axially unrestrained coolant tube, there is a possibility of ejection of a part of the coolant tube into the fuelling machine vault. The possibility of this event is eliminated in case of axially restrained tube. However, axially restrained tube experiences much more axial stresses during inservice operation. We have tried to evaluate relative sagging behaviour of axially restrained and unrestrained tubes. The finite element model described above has been analysed for both these conditions seperately. The axially restrained tube has been assumed to be free from axial stress at the begining of the severe accident. Due to thermal expansion and creep, axially restrained tube experienced heavy yielding. The sagging behaviour of both these tubes are projected in figures D and E, using the plots correspond to progress in sagging with the increase in central region temperature.

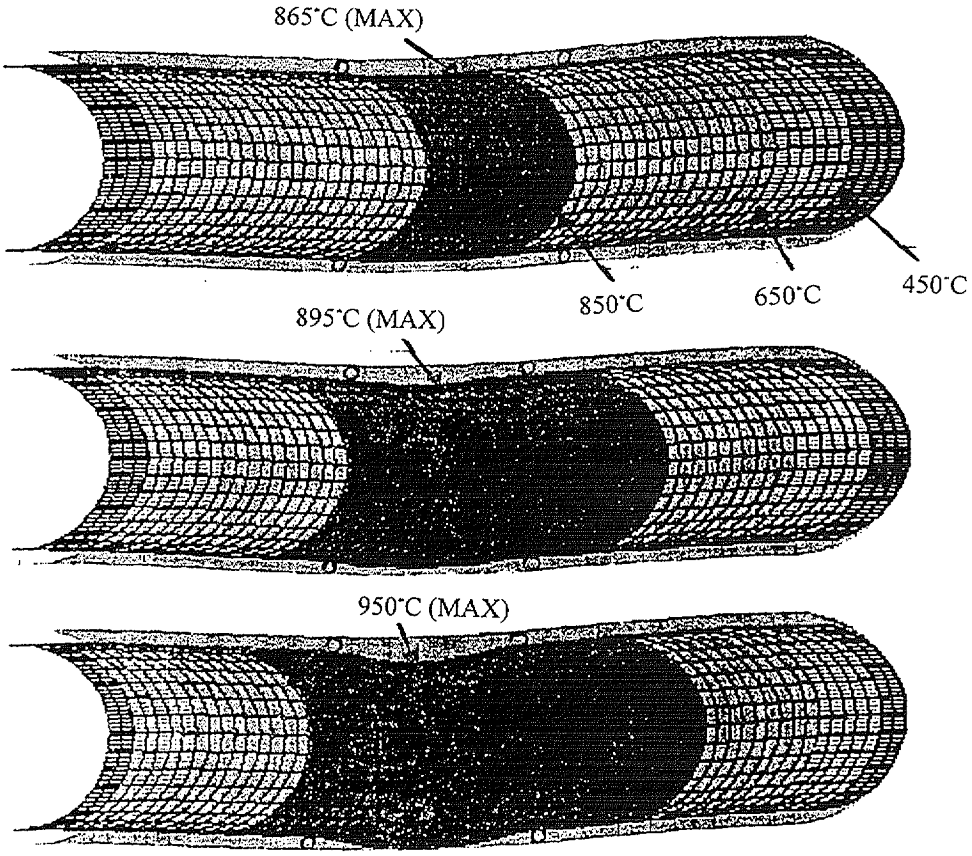

Fig. D Progress of coolant tube sagging with the channel heat up (Case of axially free tube)

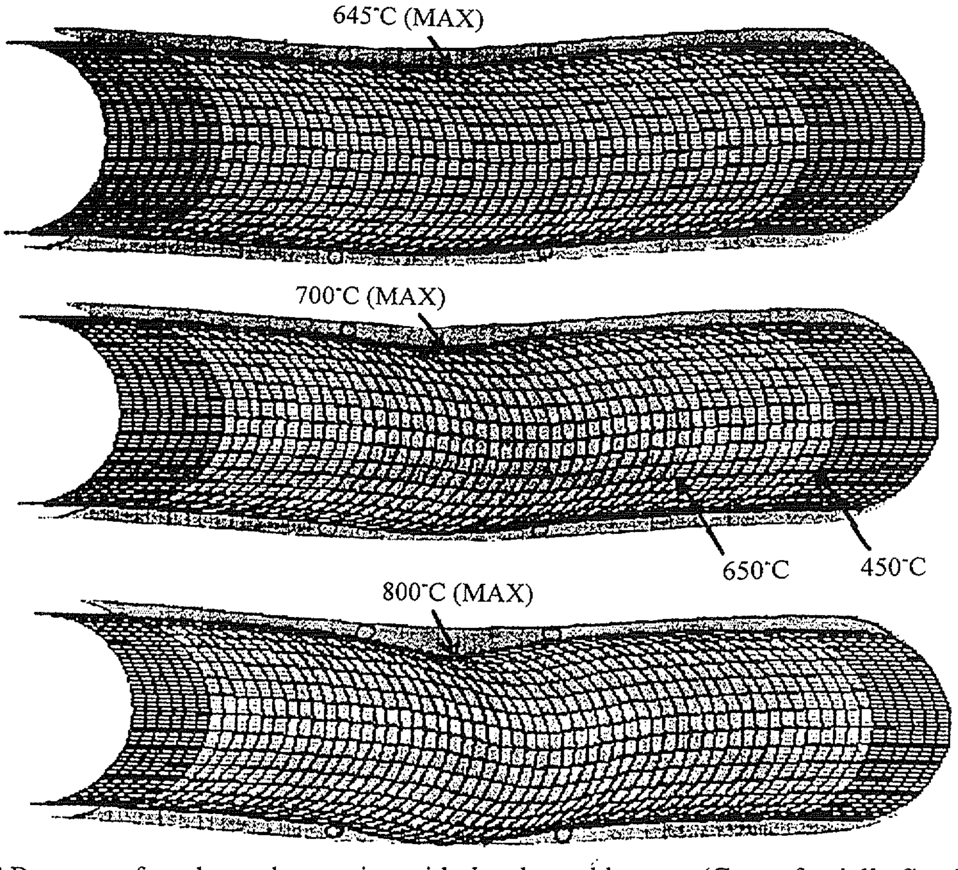

Fig. E Progress of coolant tube sagging with the channel heat up (Case of axially fixed tube)

1.2.8 DISCUSSION AND CONCLUSIONS

Deformation analysis of the coolant tube under channel heat up conditions reveals the following points.

1. The sagging process of the coolant tube starts much earlier for axially restrained tube than axially unrestrained tube.

2. The sagging temperature of axially restrained tube is around 800 Deg. Cel. and that of axially unrestrained tube is around 950 Deg. Cel. Hence, there is approximately 150 Deg. temperature difference exists between the two sagging temperatures.

3. It may be noted here that the ballooning temperature of a typical coolant tube, is between 750 Deg. Cel. to 850 Deg. Cel., depending upon the remaining internal coolant pressure after LOCA. In case the internal pressure is close to 10 MPa (i.e. drop in PHT pressure is insignificant following LOCA), the ballooning can occur at around average coolant tube temperature of 750 Deg.Cel.. However, in case of very low internal pressure (Approximately 0.5 MPa), the ballooning temperature is close to 850 Deg. Cel.

4. Comparing point no. 2 and 3 of the present discussion, it may be said that for an axially restrained tube, the ballooning can preceed sagging only if drop in PHT pressure following LOCA is very low. However, for axially unrestrained tube, the ballooning can preceed sagging for the entire range of internal pressure ranging fron 10 MPa to 0.5 MPa.

5. Hence, it is desirable to have axially unrestrained coolant tube if one aims for ballooning event to preceed sagging, this may lead to more heat transfer area.

Criterion for Channel Cooling and Integrity

Yaremey [2] described subsidiary criteria for fuel channel to remain intact under loss of coolant accident. Th
e criteria are (1) The pressure tube does not fail prior to contacting the calandria tube and (2) Following contact the calandria tube does not experience sustained dryout on the outer surface.

2.0 THERMAL RESPONSE OF REACTOR CHANNEL - FORMULATION OF THE PROBLEM

2.1 The Problem Considered

Consider a PHWR operating at a steady state, upto time t= 0. At t=0 a Loss Of Coolant Accident is initiated.

The reactor trips and the reactor channel experiences decay power for $t > 0$

At the channel section producing highest power
(a) total voiding occurs at $t = 0^+$ if the emergency core cooling system fails completely

or

(b) a sustained stratified flow level occurs at t = 0+ if the emergency core cooling system fails partially.

The steam flow rate and water level (if any) remain steady for t > 0+.

The pressure tube deforms at higher temperatures. The mode of deformation is

(a) ballooning, if the internal pressure of the pressure tube is high, or
(b) sagging under the weight of fuel, if the internal pressure of the pressure tube is low.

The moderator surrounding the calandria tube continues to be available. It is continuously cooled so as to maintain a constant moderator temperature.

2.2 Assumptions:

The following assumptions are made for the development of the model.

For modelling heat conduction :

1. Thermophysical properties of all channel components are constant, except the thermal conductivity of bare fuel (UO_2), which is temperature dependent.

2. The gases in the gap between fuel clad and bare fuel have no thermal inertia and provide a constant thermal conductance. This is conservatively chosen to be 5.68 kW/m2 C.

3. The contact resistance between pressure tube and calandria tube subsequent to pressure tube deformation is constant. The value is based on the experimental results of Gillespi [3] for ballooning and of Thompson and Kohn [4] for sagging.

4. The contact resistance between slumped fuel and pressure tube is constant and conservatively chosen based on the experimental observations of Muzumdar et al. [5]

5. The geometry of the reactor channel remains unchanged throughout the transient. The heat transfer consequent to deformation of the pressure tube and fuel rods is accounted for through different contact conductances.

6. Conduction through the annular gap between the pressure tube and the calandria tube is neglected. This is justified since other modes of heat transfer (radiation and/or direct contact heat transfer) will be dominant.

For modelling convective heat transfer:

1. The convective heat transfer coefficient on the outside of the calandria tube is uniform, but temperature-dependent.

2. Convective heat transfer coefficients chosen in various flow regimes are either conservative or are such that a small error in them does not affect the results significantly.

For modelling radiative heat transfer :

1. All surfaces involved in radiative transfer are diffuse, gray, and opaque.

2. Emissivities of all surfaces are constant.
3. The medium does not absorb, scatter or emit radiation.
4. Net radiation leaving any enclosure defined in Figure 1 is assumed to be zero.
5. All surfaces are in local thermal equilibrium during each time step.

For heat generation :

1. The depression in thermal neutron flux at the centre of fuel is neglected. This assumption leads to prediction of higher fuel temperatures at the centre of fuel and hence is conservative.

2. Heat generated at a node due to metal water reaction, is modeled as an equivalent volumetric heat generation.

2.3 The Computer Programs

A computer program called HT/MOD4 was developed based on the models described. The program is complex and has been developed in different stages [6,7,8,9,10].

The computer program 'SH' was developed for estimating shape factors between any two segments. The computer program 'BKD' was developed to estimate the temperature at which pressure tube will balloon/sag to touch the calandria tube.

2.4 Input To The Computer Programs

The computer program HT/MOD4 is used for analysing the fuel channel behaviour. The input used for the purpose is given below :

1. Bare fuel diameter	1.43 cm
2. Zircaloy clad thickness	0.04 cm
3. Zircaloy clad outside diameter	1.53 cm
4. Internal diameter of pressure tube	8.25 cm
5. Wall thickness of pressure tube	0.40 cm
6. Calandria tube diameter	10.78 cm
7. Calandria tube wall thickness	0.124 cm
8. Density of fuel (UO_2)	10.6 gm/cm^3
9. Density of clad (zircaloy)	6.5 gm/cm^3
10. Specific heat of fuel (UO_2)	328.9 J/kg °C
11. Specific heat of clad (zircaloy)	322.0 J/kg °C
12. Middle circle diameter (fuel bundle)	3.3 cm
13. Outer circle diameter (fuel bundle)	6.36 cm
14. Power output from maximum rated channel	3.08 MW
15. Maximum power per unit length	502 W/cm
16. Coolant average temperature	217 °C
17. Moderator average temperature	66 °C
18. conductance between pressure tube and calandria tube after pressure tube ballooning	11 kW/m^2 °C [3]
19. Contact conductance between	6.5 kW/m^2 °C [4]

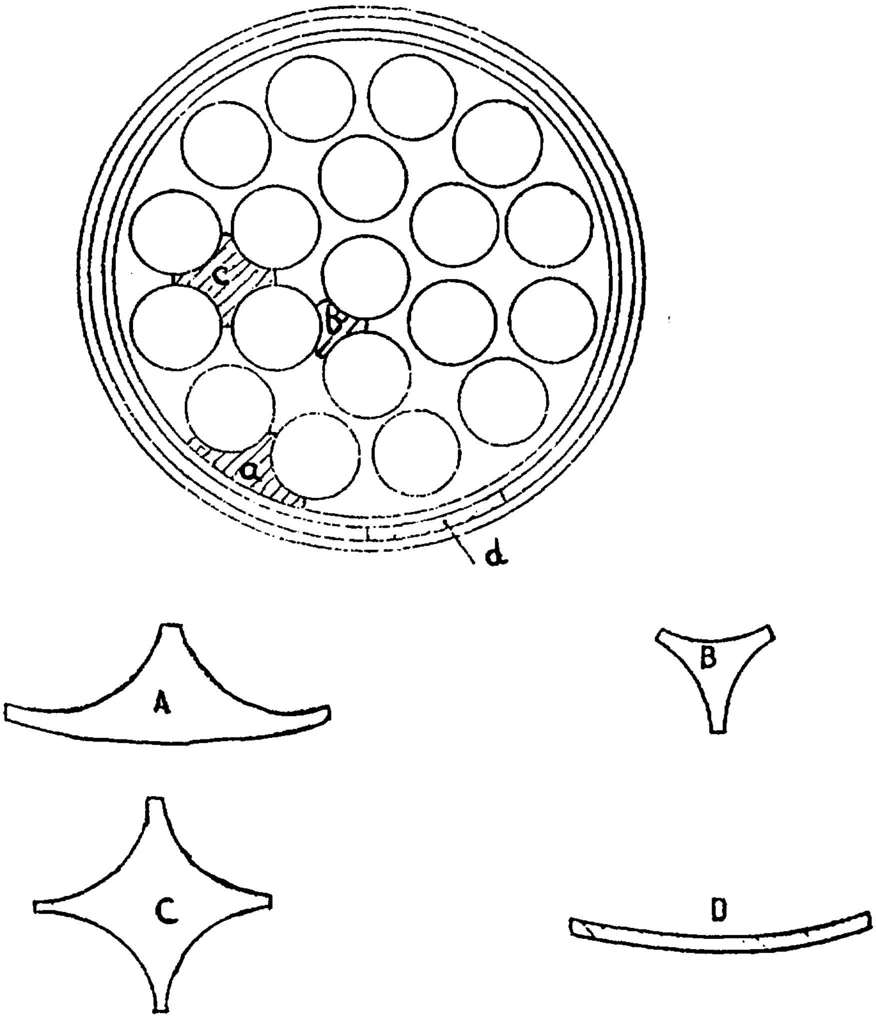

Fig. 1 Types of enclosures for radiative heat transfer

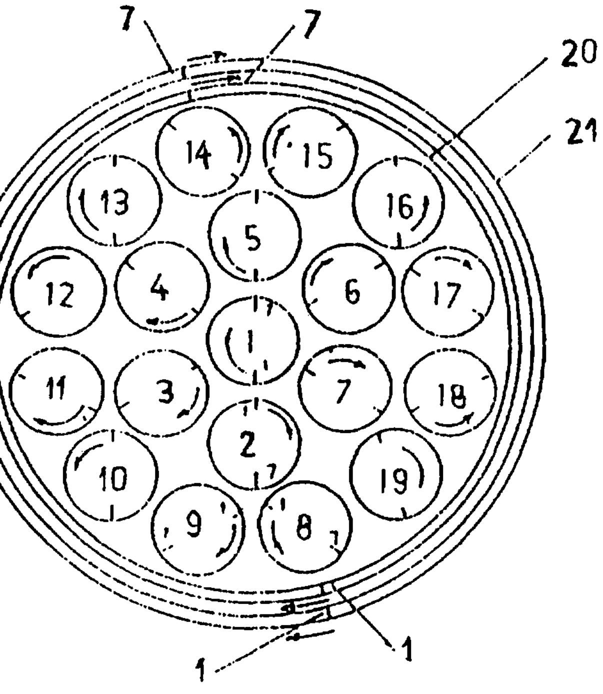

Fig. 2 Rod and node numbers

pressure tube and calandria tube after pressure tube sagging

20. Contact conductance between fuel clad and pressure tube on fuel slumping — 2.0 kW/m^2 °C [5]

3.0 RESULTS AND DISCUSSIONS

The computer program HT/MOD4 was used for analysing the fuel behaviour. Figure 2 shows the cross-section of the reactor channel with fuel rod identification numbers. For the present analysis the fuel rods, the pressure tube and the calandria tube are all divided circumferentially into 12 nodes and axially 10 nodes. Each bare fuel rod is divided into 6 radial nodes, while the fuel clad is divided into 4 radial nodes. The pressure tube and the calandria tube are divided into 3 radial nodes each.

The results obtained are presented in three-dimensional plots. Three-dimensional plots present a qualitative picture of the reactor channel temperatures.

3.1 Channel Temperature Transients For Totally Voided Channel

In this case, the reactor is assumed to be operating at full power at time t=0, when the transient starts. The channel is assumed to be totally voided immediately, i.e. at $t=0^+$. This assumption of an immediate and complete voidage is conservative. The reactor trips at $t=0^+$ and produces only decay power. The following two cases are presented :

Case 1 : The internal pressure in the pressure tube is low and the tube sags under the weight of fuel and makes a contact with the calandria tube as determined by the code TABS

Case 2 : The Internal pressure in the pressure tube is high and the tube balloons and makes contact with the calandria tube as determined by the code TESIS

In either case, it is assumed that all the fuel rods slump on to the pressure tube when the temperature anywhere in the clad exceeds 1800 deg. C. This assumption and the assumption that the pressure tube deforms at 1000 deg. C, is conservative [3,4].

Figures 3 and 4 show the initial, steady state temperature distribution in the reactor channel. The following characteristics are noted.

1. All fuel pellets have an almost parabolic radial temperature distribution, while the clad, the pressure tube, and the calandria tube have an almost linear radial temperature distribution.

2. The fuel rods in the outer pitch circle have the highest peak fuel temperatures while the rod in the center has the lowest peak fuel temperature. This is because of the difference in the relative power generation in the rods.

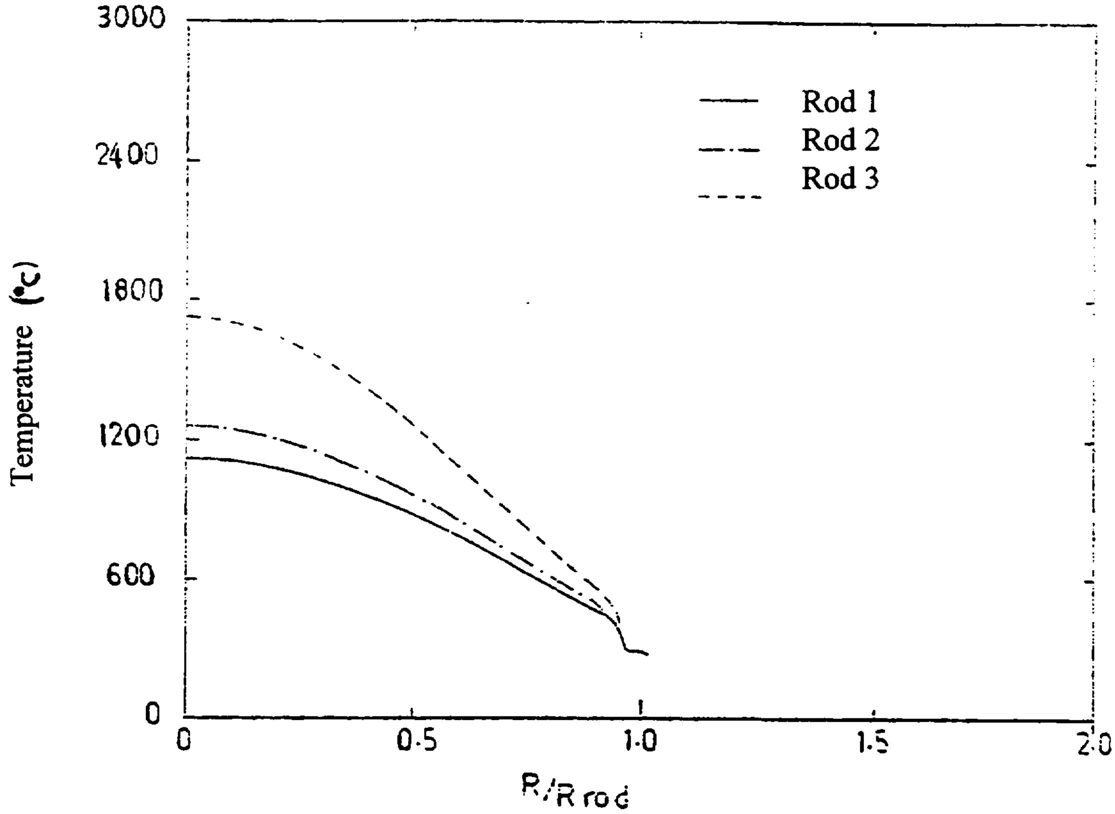

Fig. 3 Initial radial temperature distribution in fuel rods

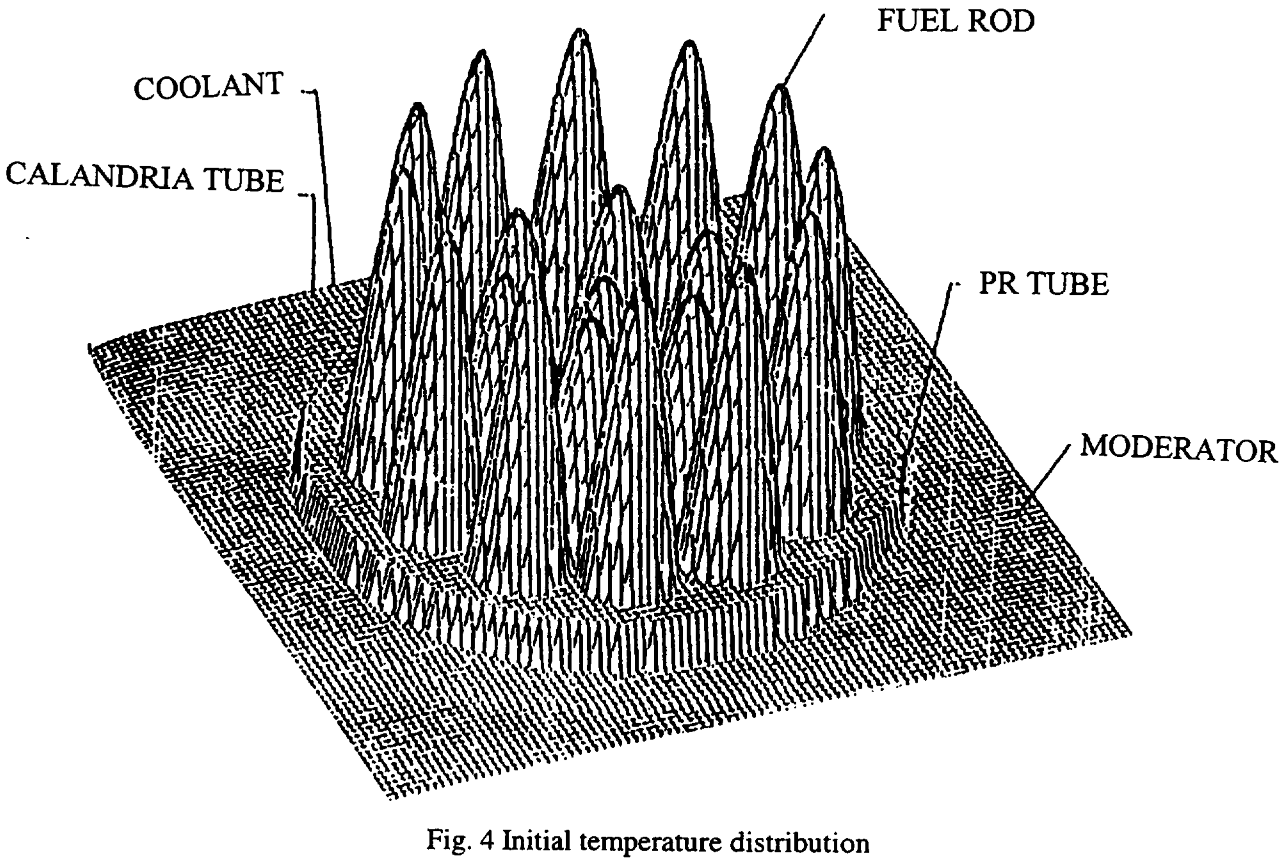

Fig. 4 Initial temperature distribution

3. There is a significant temperature drop in the gap between fuel and clad. This results in higher heat storage in the fuel.

4. The pressure tube temperature is almost equal to that of the coolant and the calandria tube temperature is almost equal to that of the moderator. This is because of good convective heat transfer.

Figure 5 shows the temperature distribution 10 s after the initiation of the accident. The following observations are made :

1. Rods with higher heat generation prior to the accident are at a higher temperature.

2. All fuel rods show a slight peak in the temperatures at the outer surface of the clad (Figure 5). This indicates that the temperature of the fuel clad has crossed the metal water reaction threshold. This reaction has been initiated, leading to heat generation at the surface of the clad and a rise in the local temperature.

3. The pressure tube and the calandria tube temperatures have not increased significantly. This is because the heat received by the pressure tube from the fuel rods on the outer pitch circle and by the calandria tube from the pressure tube by radiation heat transfer is not yet significant.

The temperature distribution in the fuel pellets at 150 s is shown in Figure 6. The nature of the distribution is similar to that in Figure 5 except that the curves are flatter and the values are higher. This is because of a further decrease in the decay heat generation.

Figure 7 shows the diametral temperature distribution at 200 s. The following observations are made :

1. The central rod temperature decreases towards its center. This indicates that heat generation at the surface (due to metal water reaction), dominates over decay heat generation.

2. The temperatures of the rods on the inner pitch circle are higher than those of rods on the outer pitch circle, even though rods on the inner pitch circle produce less decay heat. This again is a consequence of the fact that radiation heat transfer, with the pressure tube as the coolest surface, is the dominant mode. The inner rods are hotter because they are shielded from the pressure tube by the outer rods. By the same argument, the central rod could soon turn out to be the hottest one.

3. It is seen that the temperature distribution in the rods (except the central rod) is not radially symmetric about their axes. For the outer pitch circle rods, locations near the centre of the channel have higher temperatures while locations away from the centre of the channel have lower temperatures. For the rods on the inner pitch circle the temperature gradient is in the opposite direction. This

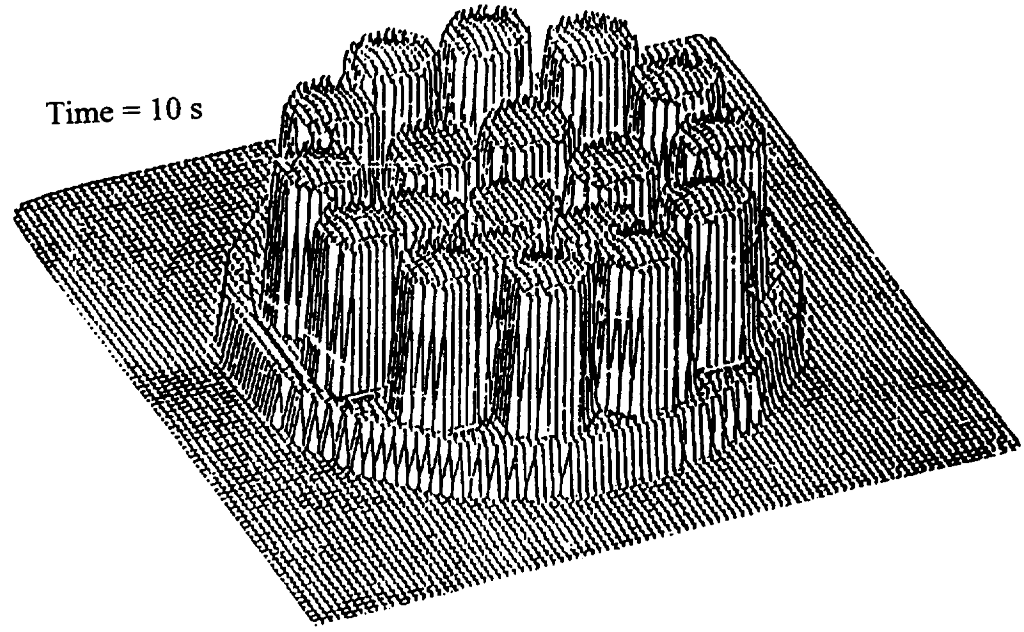

Fig. 5 Temperature distribution in reactor channel at 10 s

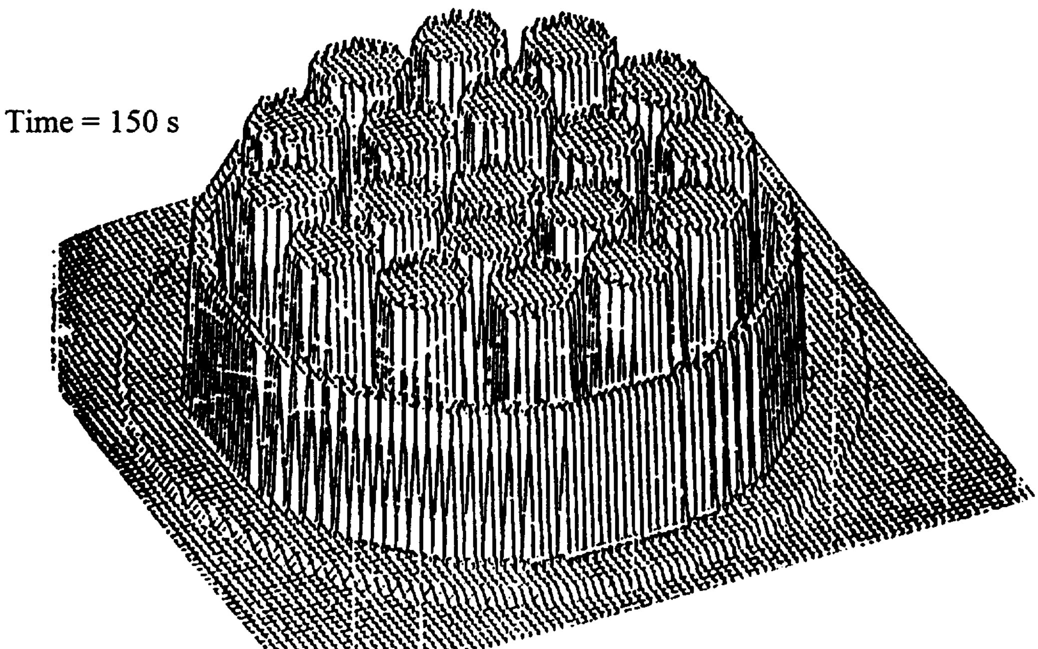

Fig. 6 Temperature distribution in reactor channel at 150 s

confirms the domination of radiative heat transfer, because this mode causes the boundary conditions on each rod to be unsymmetric.

Case 1 : Contact assumed to be due to balloning.

At about 200 s the pressure tube temperature becomesso high that it deforms. The mode of deformation affects the thermal behaviour hence forth. The results for Case 1 (ballooning) are presented first. It is assumed that the pressure tube ballons at a temperature of 1000 deg. C and makes a uniform and strong contact with the calandria tube. The contact conductance is assumed to be 11 kW/m^2 °C. [3]

Figure 8 shows the temperature distribution in the channel at 250 s. It is seen that the pressure tube temperatures have come down significantly subsequent to its contact with the calandria tube. However the rise in the calandria tube temperatures is not significant. The temperature peaks on the fuel surface have also increased in magnitude showing a higher rate of metal water reaction. The temperatures in the channel keep rising and pressure tube ballooning and its contact with the calandria tube is not able to arrest the temperature rise. In the scenario chosen for analysis the clad is assumed to fail at 1800 deg. C leading to slumping of the fuel inside the pressure tube.

Figures 9 show the peak surface temperature transients in the fuel rods, pressure tube and calandria tube following pressure tube ballooning and fuel rod slumping. The temperature transient in the central rod (Rod 1, Figure 2, surface locations 1 and 7) are shown in Figure 9. The initial sharp rise in the fuel surface temperature is because of the redistribution of stored heat in the fuel. This is followed by a gradual rise caused by the decay heat. The rise in temperature becomes steeper at about 1000 deg. C when the metal water reaction in the clad starts producing considerable heat. The rise in the slope of the curve is controlled because the steam supply is limited, and this restricts the rate of metal water reaction. The rate of increase of temperature decreases at a later time because of improved heat removal due to pressure tube ballooning and fuel slumping. After about 240 s the temperatures at locations 1 and 7 drift apart because of dissimilar boundary conditions, caused by fuel slumping.

Figure 10 shows the temperature transient for the pressure tube. Initially the temperature of the tube rises as it receives heat from the fuel rods. After its ballooning at 1000 deg. C the temperature suddenly falls due to the contact with calandria tube. The calandria tube temperatures do not change much because of good convective cooling by the moderator. The circumferential temperature distribution in the pressure tube at different times is shown in Figure 11. It is seen that the pressure tube temperature is initially uniform. After the fuel slumps and makes a contact with the pressure tube, there is a local rise in temperature at the area of contact. The temperature in the zone of contact rises initially, but reduces later on when the fuel temperatures start falling. It is seen that there is a sharp decrease in temperature after about 480 s. This is because, by then at most places the cladding has been completely oxidised and so the rate of heat production due to metal water reaction reduces almost to zero.

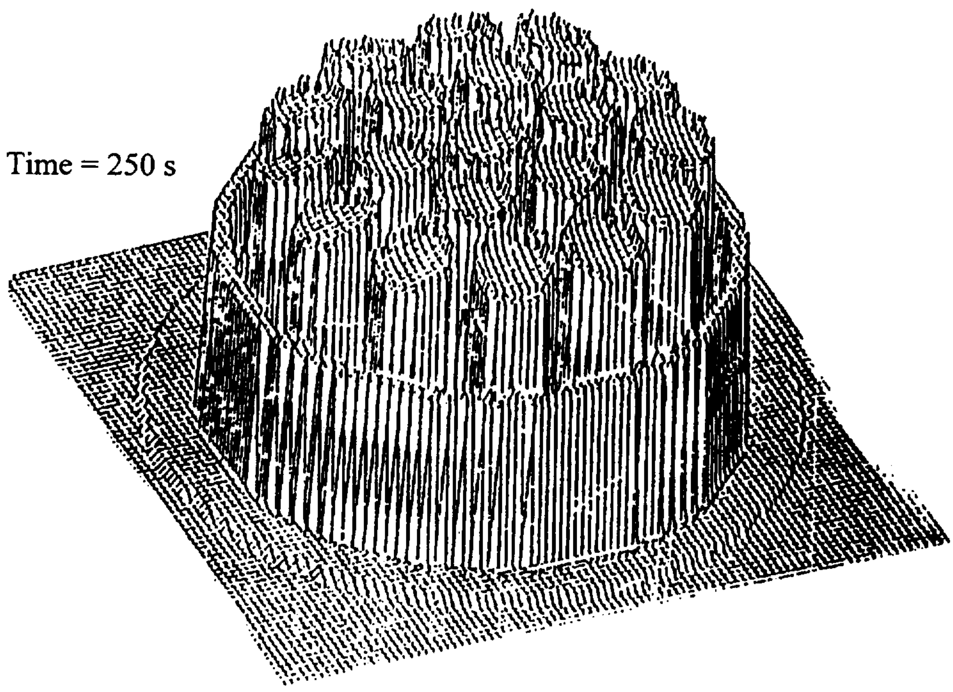

Fig. 7 Temperature distribution in reactor channel at 200 s

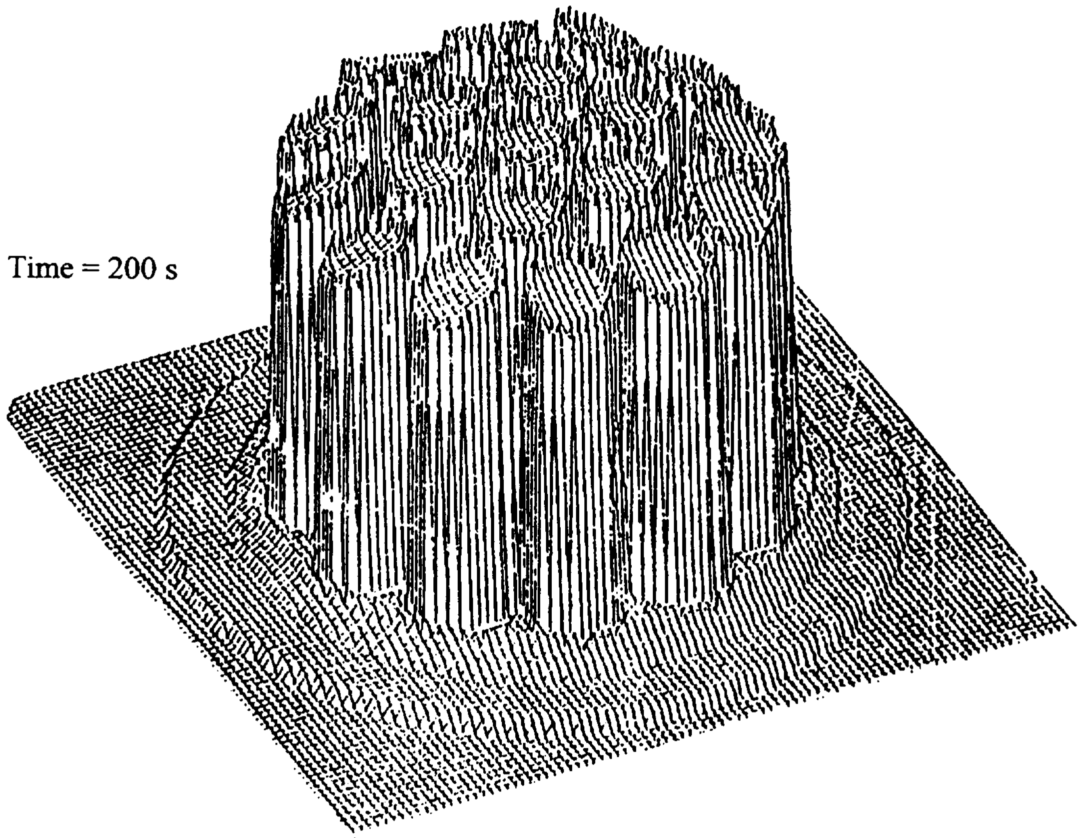

Fig. 8 Temperature distribution in reactor channel at 250 s

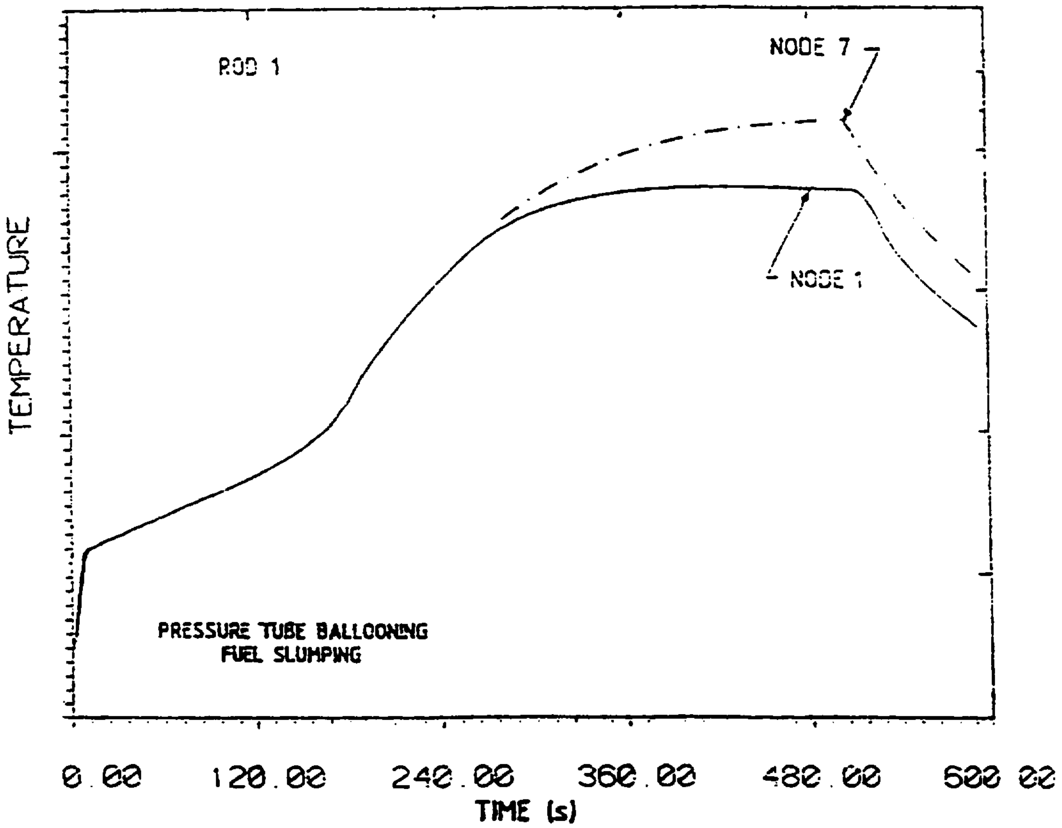

Fig. 9 Surface temperature transient for ROD 1

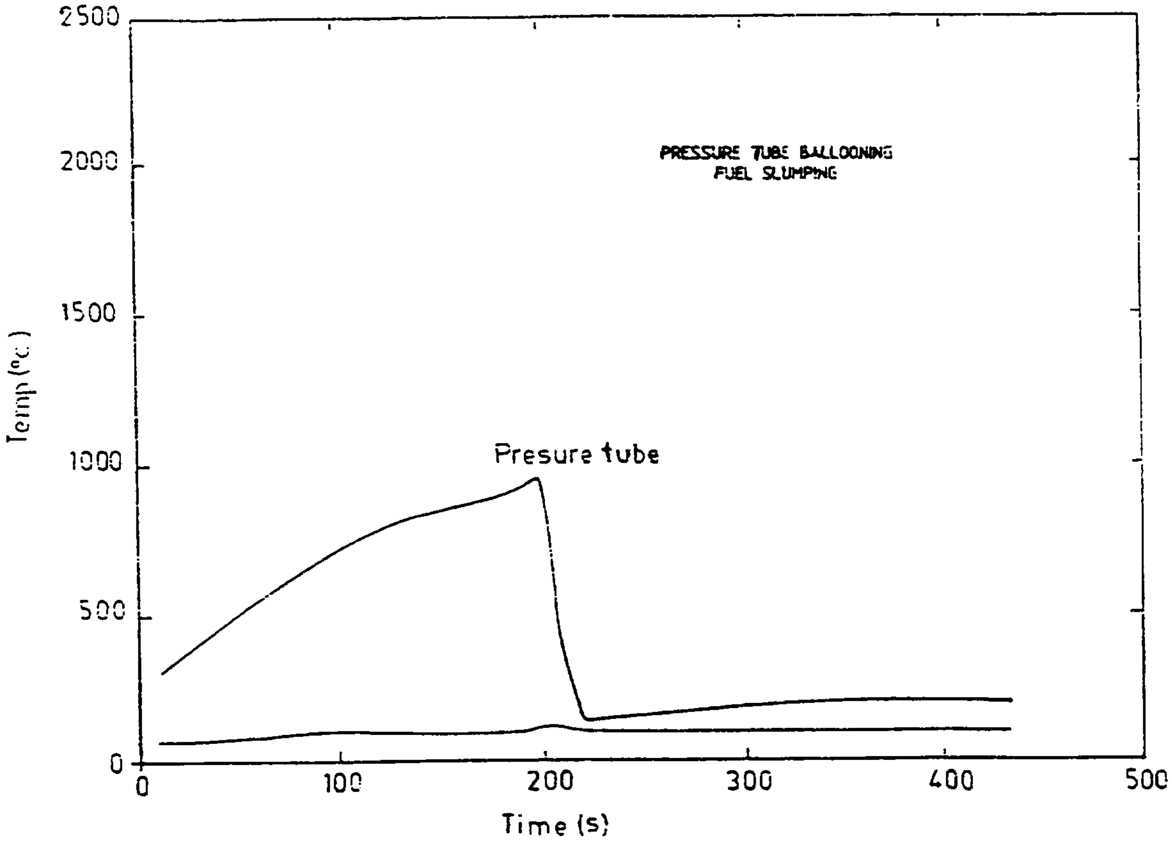

Fig. 10 Peak surface temperature transient

Case 2 :Contact assumed to be due to sagging.

A different behaviour is seen in Case 2. At 1000 deg. C, the pressure tube is considered to sag under the weight of fuel due to deterioration in its mechanical properties. On sagging, it is assumed that the pressure tube touches the calandria tube over 1/3rd of its circumference [4]. The contact conductance is 6.5 kW/m2 °C [4].

The temperature transient for rod No. 1 is shown in Figure 12. It may be noticed that the temperature - time behaviour is identical to the first case for the first 200 s. After that, although the nature of the transient is similar to that of the previous case, in this case the temperatures reached are higher because, (a) only a part of the pressure tube is in contact with the calandria tube, and (b) the contact conductance is smaller.

In the case of sagging, the pressure tube experiences large circumferential temperature gradients. Figure 13 shows the temperature distribution in the pressure tube at various times. Initially, at 150 s, the temperature distribution is uniform and it rises with time as is clear from the plot at 200 s. At about 200 s the pressure tube sags. The temperature at the contact suddenly falls due to transfer of heat by conduction to the calandria tube and then to the moderator by convection. The temperature of the rest of the pressure tube also falls, as is seen from the temperature plot at 250 s. Meanwhile the fuel temperatures keep rising and hence the rate of transfer of heat to the pressure tube increases. The pressure tube circumference not in contact with the calandria tube is not able to transfer heat directly to the calandria tube by conduction. Consequently the temperatures rise and the circumferential temperature gradients increase. This is seen from the plots at 300 s and 350 s. The circumferential temperature distribution at 560 s shows a decrease in temperatures, consequent to a decrease in the temperature of the fuel rods.

Figure 14 compares the peak fuel temperature transients for the two cases involving (i) ballooning and (ii) sagging. As is expected the fuel temperatures are higher in the case of pressure tube sagging. It is also seen that the fuel temperatures do not reach the fuel melting point (2800 deg. C) in either case.

3.2 Effect Of Convective Heat Transfer Coefficient To Steam In The Voided Channel

In the analysis described in section 3.1 the fuel temperatures rise slowly initially. Later at higher temperatures the heat produced by metal water reaction is substantial and causes a rapid rise in fuel temperatures. The results presented in 3.1 indicate that the rise of temperature is largely due to the metal water reaction, whereas the heat removal depends on, to some extent, the convective heat transfer to steam. Since the value of convective heat transfer coefficient used in the previous section is due to Gillespie [3] and is based on limited experimental data, it is considered necessary to study the sensitivity of the predictions to the heat transfer coefficient to steam. The effect of increasing the heat transfer coefficient to steam by 25 percent has been studied for the case of ballooning of the pressure tube (Case 1). Figure 15 shows the peak surface temperature transients in the reactor channel. Study of Figure 15 indicates that

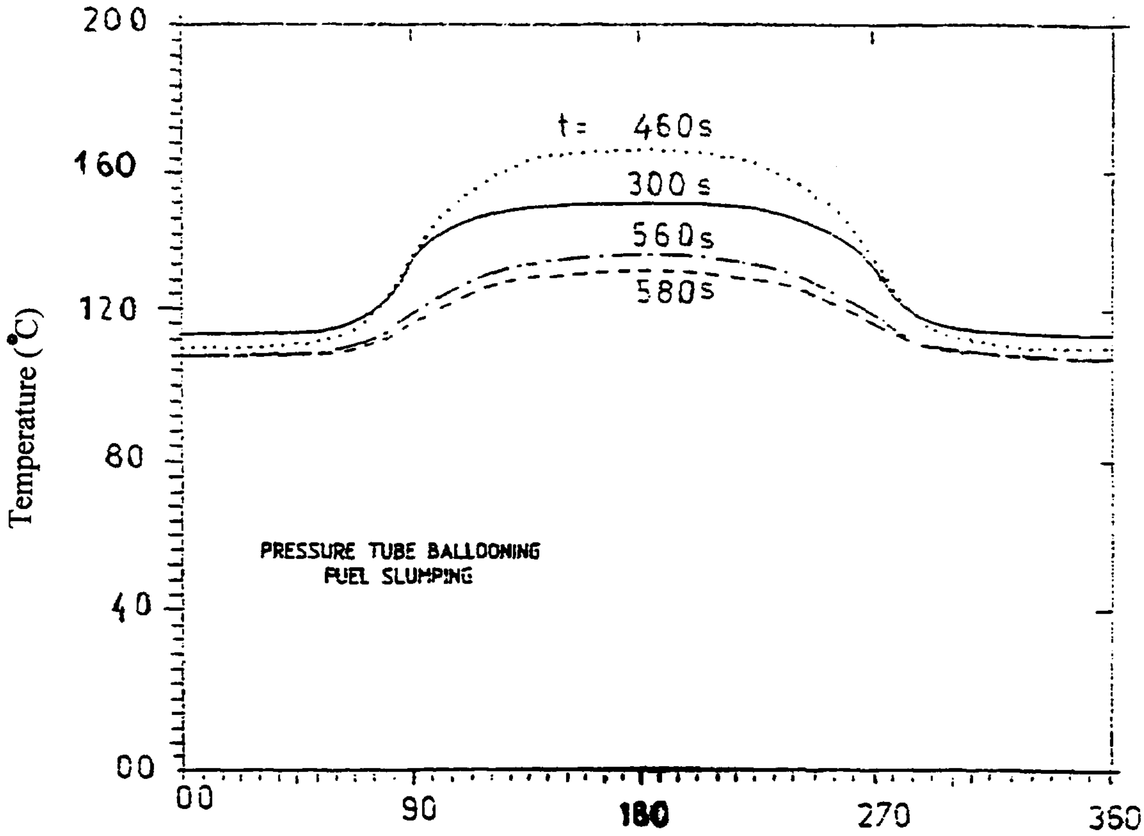

Fig. 11 Circumferential temperature distribution in pressure tube

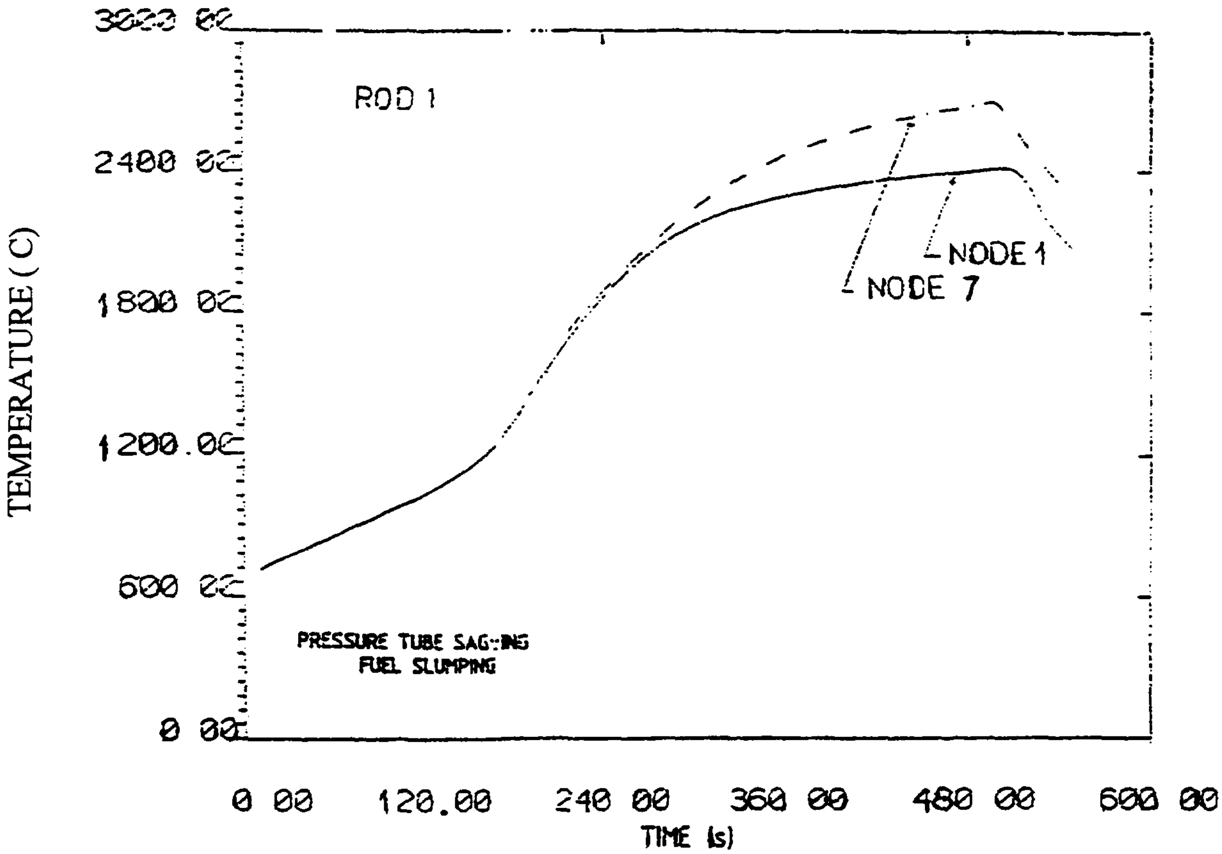

Fig. 12 Surface temperature transients for ROD 1

1. The behaviour in the first 40 s is independent of the heat transfer coefficient to steam. The temperatures in this period are essentially determined by the redistribution of stored heat.

2. With a higher heat transfer coefficient to steam, all temperatures tend to reach a shallow peak and then fall gradually. This indicates that the temperatures do not reach the levels where metal water reaction starts producing a significant amount of heat.

3. All fuel rod surface temperatures tend to come close to each other. This shows that radiative heat transfer among rods is significant.

4. With the higher heat transfer coefficient, the pressure tube temperature rises gradually and then falls. It does not reach a value where the pressure tube will deform.

The results presented in Figure 15 show that the predicted channel temperature transients are very sensitive to the coefficient of convective heat transfer to steam.

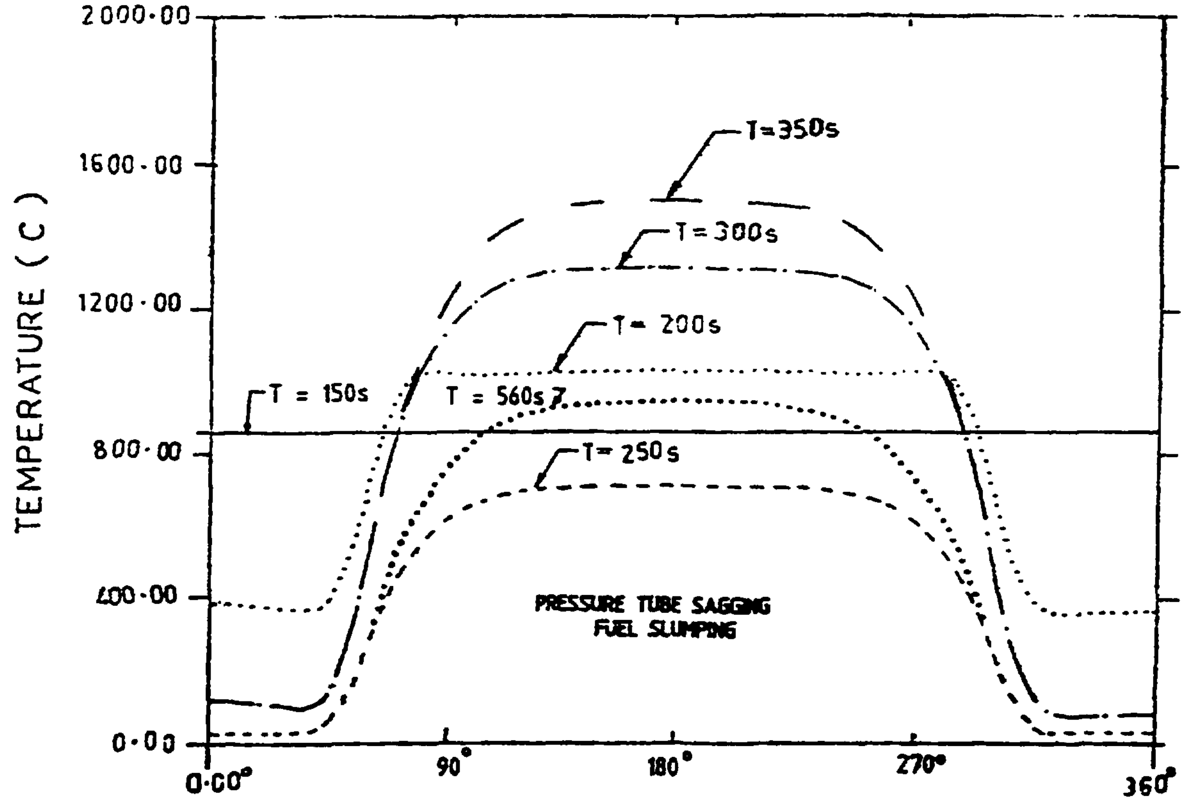

Fig. 13 Circumferential temperature distribution of pressure tube

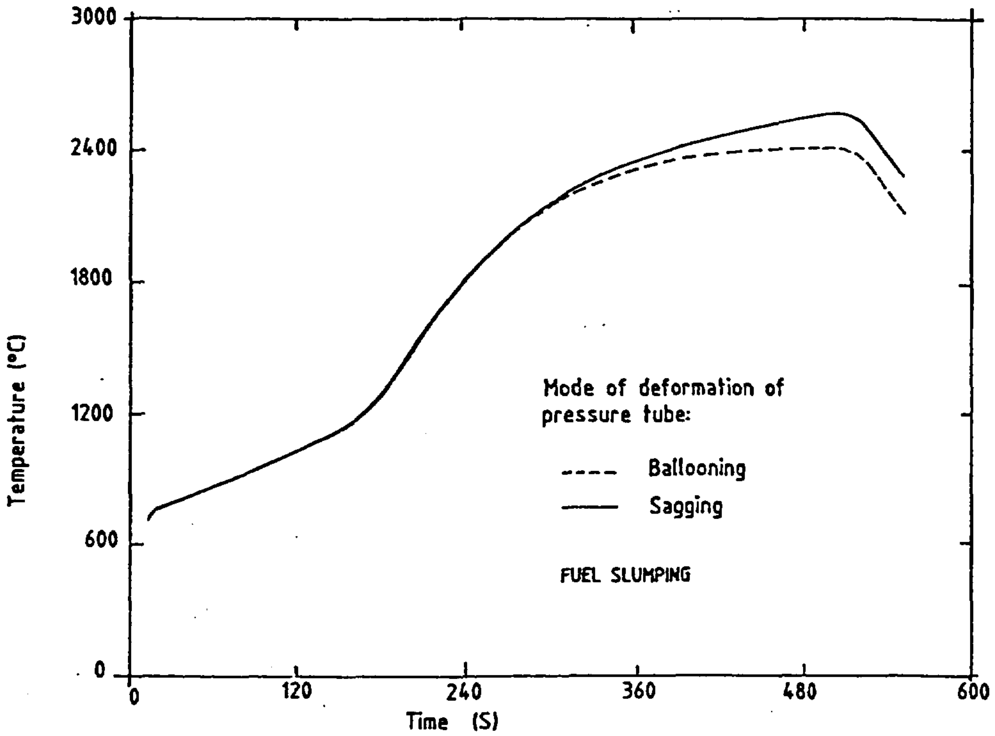

Fig. 14 : PEAK FUEL TEMPERATURE TRANSIENT OF THE CHANNEL UNDER THE TWO MODES OF DEFORMATION

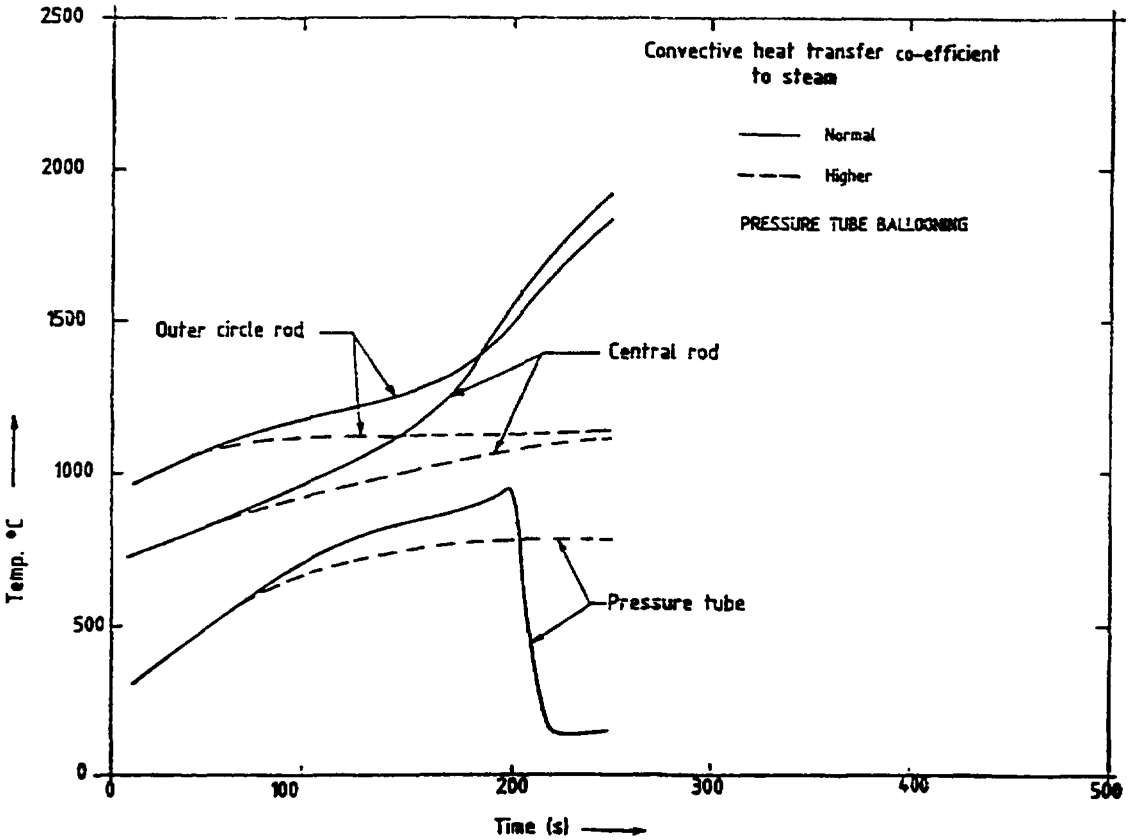

Fig. 15 Peak surface temperature transient for different convective heat transfer coefficients

3.3 Channel Response At Sustained Stratified Flow Levels

A large break loss of coolant accident with partial failure of the emergency coolant injection system may lead to a sustained stratified flow level in a channel. To study the effect of level, the following situation is analysed. The reactor is operating at full power upto time t = 0. The accident leads to reactor trip and it is assumed that a sustained stratified flow level is established in the channel at time $t = 0^+$. Various cases of stratified flow level are considered and these are shown in Figure 16. It is seen from the figure that in all the cases, some of the rods are partially submerged, while some others are fully submerged.

The results for the case of level at half the channel diameter are presented in Figures 17 and 18. Figure 17 shows the circumferential temperature distribution at 10 s after the initiation of the accident. In this case the central rod is half submerged in the coolant and hence it has the largest circumferential temperature gradient. The temperature of the exposed portion rises due to initial redistribution of heat while that of the submerged portion stays low due to strong convective cooling by heavy water.

Figure 18 shows the three dimensional plot of the temperatures in the channel at 10 s. It is seen that the temperatures of the submerged rods have come down significantly. However, the temperature distribution in these rods shows that the stored heat has not yet been totally removed. Also the small temperature peaks at the surface of the exposed rods indicate the start of the metal water reaction.

Figure 19 shows the circumferential temperature distribution in the different rods at 200 s. The gradients in the central rod which were very high at 10 s are now reduced. The distribution in rod 4 now shows a temperature gradient towards the water level. This is because radiation heat transfer has become significant, and the rod is losing heat to water which also acts as a radiation heat sink. Rod 5, although in the same pitch circle has a higher temperature, because it is away from the coolant. A similar behaviour occurs for the outer pitch circle rods 12 and 14 (not shown in the figure).

Circumferential temperature distribution in the pressure tube at 200 s is presented in Figure 20. The pressure tube temperature rises in the exposed area (non wetted). It receives heat from the outer pitch circle rods and is not able to conduct it to the coolant through its submerged portion.

4.0 CONCLUSIONS

The present analysis predicts significant temperature differences along the circumference of the fuel rods. These circumferential temperature variations affect not only the heat transfer but also the time which fuel rods fail and slump on to the pressure tube. Also at higher temperatures, the fuel clad may balloon under the internal pressure of fission gases and cause a flow blockage for any emergency coolant injected or may rupture leading to release of fission products.

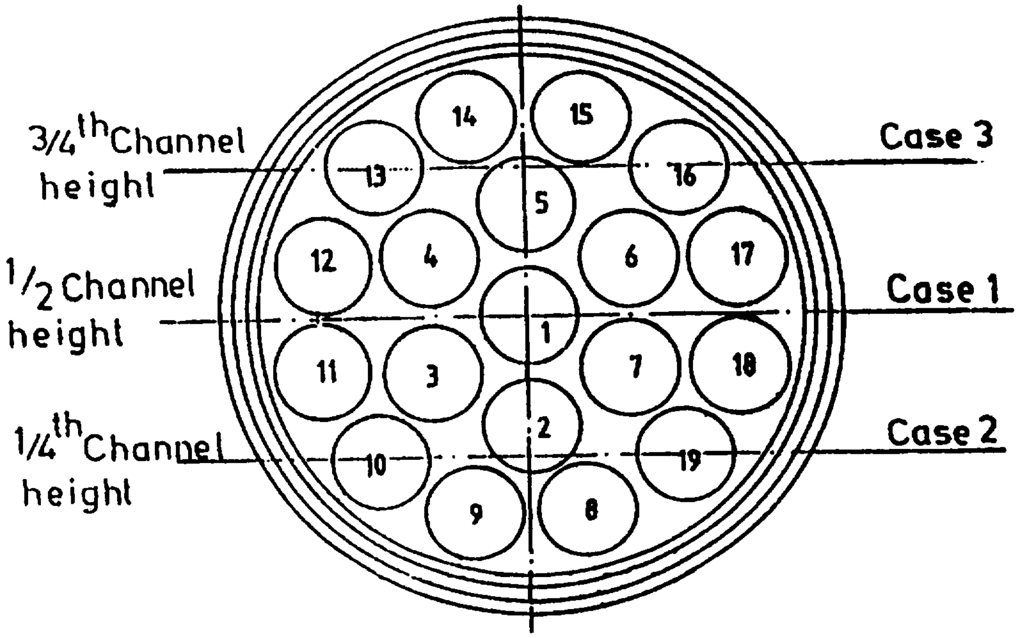

Fig. 16 Water levels studied

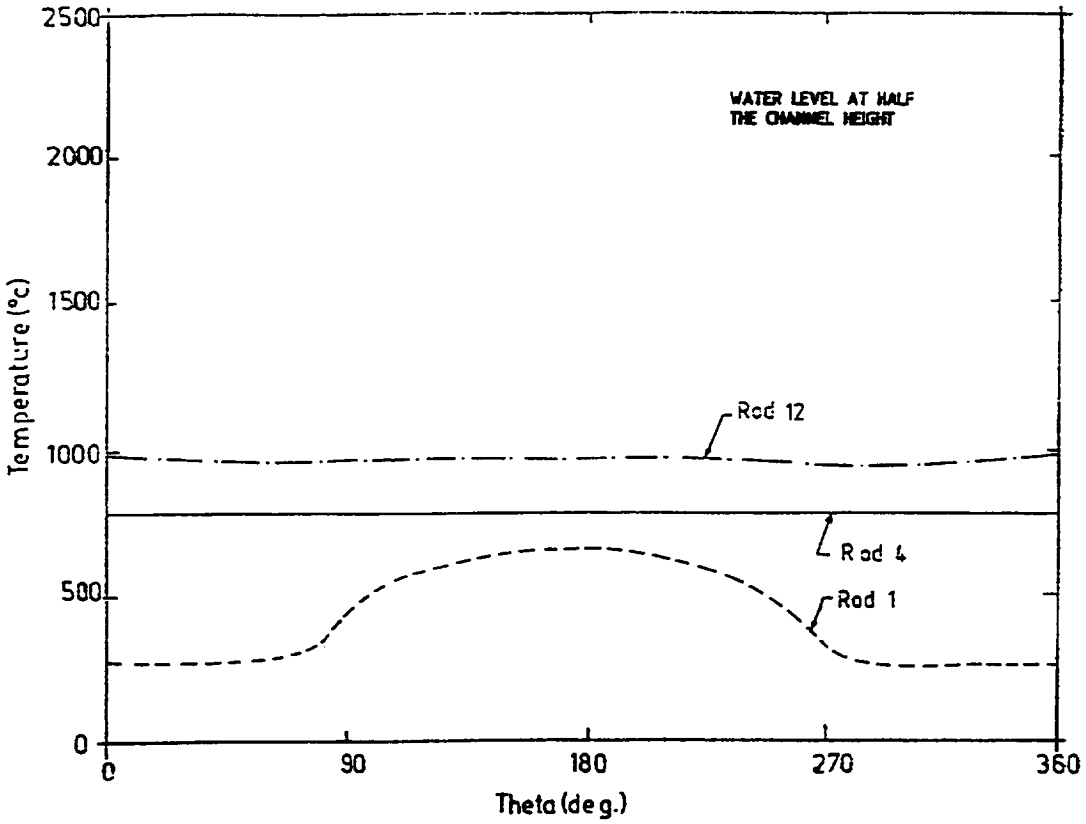

Fig. 17 Circumferential temperature distribution for RODs 1,4 and 12 at 10 s

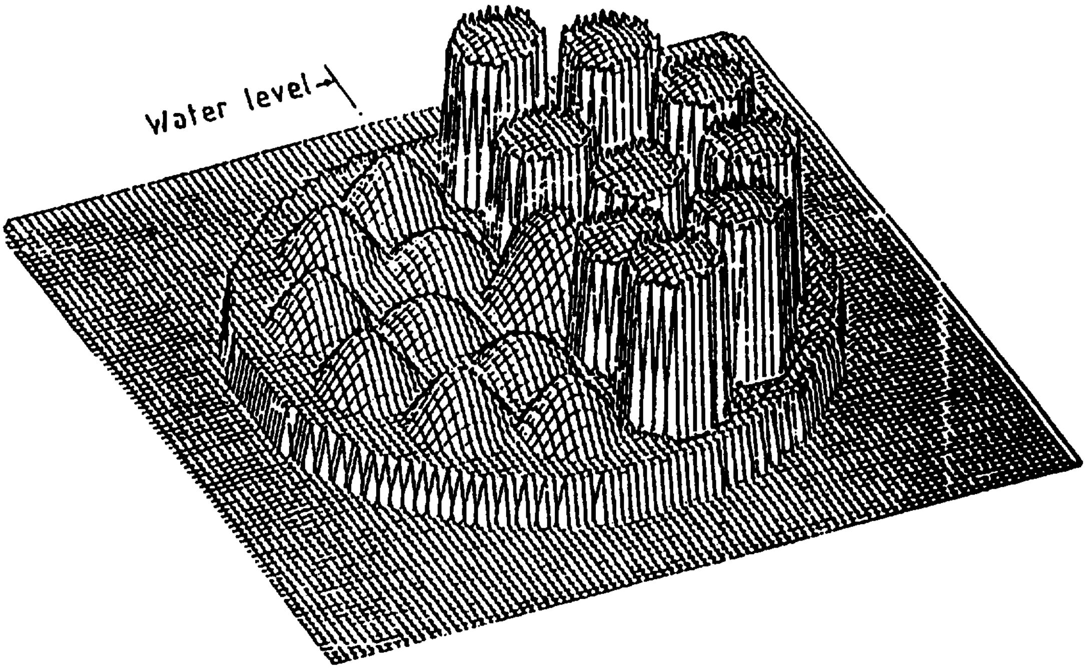

Fig. 18 Temperature distribution in reactor channel with water level at half the channel height at 10 s

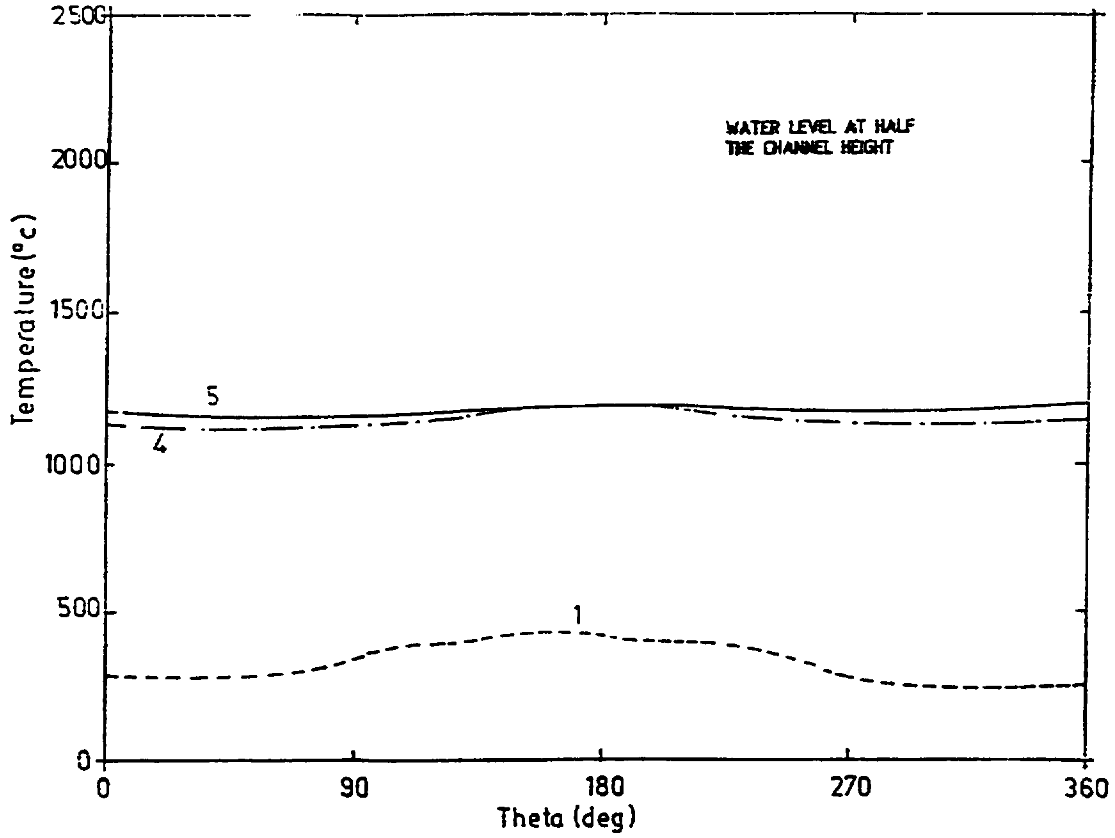

Fig. 19 Circumferential temperature distribution for RODs 1, 4 & 5 at 200 s

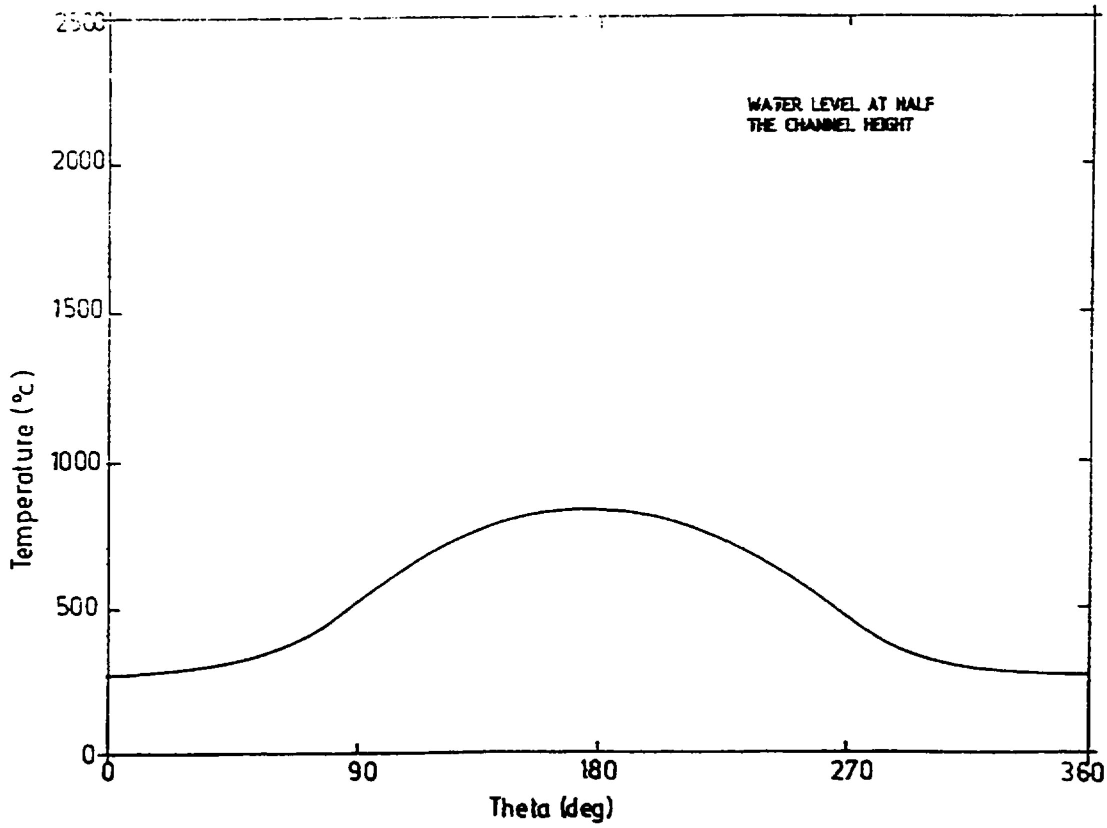

Fig. 20 Circumferential temperature distribution for pressure tube at 200 s

For the case where the channel is completely voided, the analysis reported is conservative but even then fuel temperatures remain below the melting point of uranium dioxide for a case of loss of coolant accident coincident with total failure of the emergency core cooling system. The required condition for this is the availability of moderator cooling.

It is observed that heat produced due to metal water reaction determines the temperature rise in different channel components. The pressure tube deformation and its contact with the calandria tube is not able to prevent fuel failure because of temperature rise. The fuel cluster slumping onto the pressure tube reduces the temperature rise in different channel components significantly. However, the temperatures start falling only when the cladding of some fuel rods oxidises completely, thus reducing the energy released during the metal water reaction.

This analysis shows that circumferential temperature gradients appear in the pressure tube subsequent to the pressure tube making a contact with the calandria tube. These temperature gradients are smaller for the case of pressure tube ballooning (Figure 11) but can be significant for the case of pressure tube sagging (Figure 13). In case of pressure tube sagging the temperatures and their gradients continue to rise for a considerable period. In view of this the integrity of the pressure tube needs to be further examined even after its contact with the calandria tube. However, the calandria tube integrity is not threatened. Its temperature and the temperature gradients do not increase significantly.

For the cases where a sustained stratified flow level exists in the channel, The results obtained indicate that if the level of the stratified flow is more than half the channel height, the temperatures in the channel components are controlled and may not endanger channel integrity.

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COOLANT VOID REACTIVITY BEHAVIOUR IN PRESSURE TUBE TYPE PLUTONIUM FUEL LATTICES OF ATR

XA9846616

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Abstract

It is essential to the reactor physical field in respect of the thermal neutron reactor specially utilizing plutonium to consistently clarify the effect of the resonance in the around 0.3eV for plutonium239 and the effect of the various fuel compositions and fuel material concentrations on the void reactivity in the ATR lattice. For this purpose, using the improved WIMS code to be able to calculate the detailed thermal neutron spectrum containing the resonance in 0.3eV for plutonium239 and various components due to the reactivity relevant to the void reactivity, the void reactivity in the HWR lattice has been analyzed in detail. For the purpose of improving the accuracy of the calculated void reactivity, the nuclear data for the WIMS code has been replaced with the JENDL-3.1 library. As the results of the analysis, for the identical macro thermal neutron absorption cross section for fuel, the void reactivity is reduced to the negative side more for the MOX with high content of plutonium239 in plutonium than in uranium.

The reduction of the void reactivity to the negative side by plutonium239 is due to the presence of the resonance in 0.3eV for plutonium239. That is because the higher the content of plutonium239 is, the less the recovery effect for neutron density within the resonance energy in 0.3eV is with increase in coolant void fraction, so that the decreased resonance nuclear fission rate for plutonium239 contributes to the negative side for the void reactivity. This effect is more remarkable for the larger pitch lattice with softer thermal neutron spectrum.

1. INTRODUCTION

ATR is the heavy-water-moderated, boiling-light-water-cooled, pressure-tube -type reactor which is now under developing in Japan. Fugen is a 165 MW(e) prototype of ATR. Site construction began in December 1970 and the plant commenced commercial operation on March 20, 1979. Since then,Fugen has been operating safely and smoothly more than for 15 years. ATR has outstanding flexibility of utilizing both depleted uranium fuel and plutonium fuel with excellent performance. The whole core can be filled with MOX fuel. The ATR core with the MOX fuel shifts the coolant void reactivity to more negative than with the UOX fuel. This fact is verified by both critical experiments using the DCA[(1),(2),(3),(4)] and operation experiences of Fugen[(5)].

From now,plutonium increases because of the high burn up of light water reactor fuel and the plutonium isotopic composition becomes complexed. It is planning that the multi-rod fuel cluster with the higher enrichment MOX fuel will be loaded in the Fugen core in order to attain the higher burnup. It is, therefore, very important for advances in ATR to clarify the mechanism why MOX fuel shifts the coolant void reactivity to more negative side. This paper presents the analytical results for clarification why MOX fuel shifts the coolant void reactivity to more negative side by using the ATR nuclear characteristics estimation code WIMS-ATR[(6)] whose accuracies are verified by the heavy water critical experimental analyses of DCA.

2. ANALYTICAL METHODS

2.1 Approaching Method

It has been found from the experiment that the different effects between uranium and plutonium nuclides in the fuel and the different effects between the plutonium isotopic composition ratios in the MOX fuel on the coolant void reactivity result in the presence of Pu-239 resonance at around 0.3 eV[1),(2)]. However, in order to support sufficiently the conclusion obtained from the experiment, the fuel enrichment and the plutonium isotopic composition dependency of the coolant void reactivity need to be broadly analyzed by a wide variation of indicators for fuel enrichment (for example, the macroscopic absorption cross section of fuel corresponding to the 2200m/s neutron, Σ_{a0}).

For this purpose, the various components relevant to the coolant void reactivity in the various MOX fuel lattices are determined using the ATR lattice calculation code WIMS-ATR with calculation accuracy confirmed by comparing between calculations and experiments obtained by the heavy water critical experimental facility(DCA). And the range of the validity for the conclusion obtained from the experiment and the effect of the Pu-239 resonance at around 0.3eV are quantatively evaluated by the present analyses.

Here, from necessity for consideration of abstracting the effect of the resonance in the MOX fuel lattice on the coolant void reactivity, the calculation is carried out by considering the following items.

1) We are to exactly assess the effect of the resonance cross section of plutonium using the Japanese Evaluated Nuclear Data Library (JENDL-3.1)[(7)] which is compiled recently.

2) Mesh-dividing the cluster-type fuel lattice in detail in the heterogeneous system, the components of the fuel rod and components of the coolant for each layer within the cluster against the coolant void reactivity are divided for assessment .

3) The basic library of the WIMS-ATR code has been compiled with 69 energy groups and the group at around 0.3 eV of the Pu-239 resonance absorption is divided in detail,and then the components in this energy region for grouping can be sharpened.

Then, total six groups of the region for the fast nuclear fission threshold value of U-238 and over (10 MeV~0.82MeV), region for slowing down (0.82MeV~5.53keV), region for resonance absorption (5.53keV~0.625eV), region for the epithermal neutron (0.625eV~0.4eV of cadmium cutoff energy), region for thermal neutron resonance of Pu-239 and Pu-241(0.4eV~0.22eV) and region for 1/v (0.22eV~0.0eV) were applied.

The MOX fuel supplied to the DCA critical experiment was the one which has two kinds of plutonium isotopic composition ratio(S grade ≒ 92 w/o plutonium fissile and R grade ≒ 74 w/o plutonium fissile), and so the MOX fuel lattice in the present analyses conformed to this. Further, the macroscopic thermal neutron absorption cross section Σ_{a0} for the fuel with the velocity(2200 m/s) corresponding to the most probable value of the energy for the Maxwellian distribution of the neutron energy spectrum at room temperature is adopted as the common indicator showing the enrichment of fuel or the enrichment of plutonium. The value of the Σ_{a0} for enrichment ranges from 0.2 for uranium fuel(approximately natural uranium) to 0.5(corresponding to the enrichment of about 30,0000 MWd/t for the attainable burnup). This range of Σ_{a0} corresponding to plutonium enrichment was also used in the MOX fuel lattice. Integrated variety of components for nuclides in each fuel corresponding to Σ_{a0} is shown in Table I.

2.2 Method for Deriving Component due to Coolant Void Reactivity

The component can be derived from the definition equation for the coolant void reactivity in the

TABLE I FUEL COMPOSITION FOR ANALYSES

	Σ_{a0} (cm-1)	^{235}U+Puf (wt%)	Fuel composition (wt%)						
			^{16}O	^{235}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
UO_2 (UOX)	0.20	0.88	12.15	0.7690	87.08	–	–	–	–
	0.25	1.19	12.15	1.047	86.80	–	–	–	–
	0.30	1.51	12.15	1.325	86.53	–	–	–	–
	0.35	1.82	12.15	1.603	86.25	–	–	–	–
	0.40	2.12	12.15	1.881	86.97	–	–	–	–
	0.45	2.46	12.15	2.159	85.69	–	–	–	–
	0.50	2.77	12.15	2.437	85.41	–	–	–	–
PuO_2-UO_2 (MOX(R))	0.20	0.82	12.12	0.6239	87.13	8.205E-02	2.751E-02	1.196E-02	3.804E-03
	0.25	1.01	12.12	0.6223	86.90	2.300E-01	7.712E-02	3.351E-02	1.066E-02
	0.30	1.20	12.12	0.6207	86.67	3.779E-01	1.267E-01	5.507E-02	1.752E-02
	0.35	1.39	12.12	0.6190	86.45	5.258E-01	1.763E-01	7.662E-02	2.438E-02
	0.40	1.58	12.12	0.6174	86.22	6.737E-01	2.259E-01	9.818E-02	3.124E-02
	0.45	1.77	12.12	0.6158	85.99	8.217E-01	2.755E-01	1.197E-01	3.810E-02
	0.50	1.96	12.12	0.6141	85.76	9.696E-01	3.251E-01	1.413E-01	4.496E-02
PuO_2-UO_2 (MOX(S))	0.20	0.83	12.12	0.6240	87.14	1.081E-01	1.034E-02	1.095E-03	7.700E-05
	0.25	1.05	12.12	0.6225	86.93	2.940E-01	2.811E-02	2.980E-03	2.080E-04
	0.30	1.26	12.12	0.6211	86.73	4.798E-01	4.588E-02	4.860E-03	3.400E-04
	0.35	1.47	12.12	0.6196	86.52	6.657E-01	6.365E-02	6.740E-03	4.700E-04
	0.40	1.68	12.12	0.6181	86.32	8.516E-01	8.142E-02	8.620E-03	6.000E-04
	0.45	1.89	12.12	0.6167	86.12	1.037E+00	9.920E-02	1.051E-02	7.000E-04
	0.50	2.11	12.12	0.6152	85.91	1.223E+00	1.170E-01	1.239E-02	8.700E-04

infinite lattice. The reactivity ρ can be defined as follows.

$$\rho = \frac{K_\infty' - K_\infty}{K_\infty K_\infty'} \qquad (2.1)$$

Here, k_∞, k_∞': the infinite multiplication factors respectively before and after the change of the coolant void fraction.

The following are committed for simplified equations.

(1) The coolant void reactivity is represented by the change of the eigenvalues for the variation of 0% void(H_2O) to 100% void(air) (The condition of 100% void is accompanied by dash marks).

(2) The region for fuel material is represented by f of an under-described mark and the regions for coolant, structural material, moderator, etc. except fuel material are each represented by s of an under-described mark.

(3) The mark of <> is applied to the integral value for neutron flux in energy and space.

(4) The variation of the reaction rate between the unvoided lattice and the voided lattice is as follows.

$$<\delta\Sigma> = <\Sigma>' - <\Sigma> \qquad (2.2)$$

In the WIMS-ATR code, the following normalization condition is adopted.

$$\left.\begin{array}{c} <\Sigma_{a,f}>' + <\Sigma_{a,s}>' = <\Sigma_{a,f}> + <\Sigma_{a,s}> = 1.0 \\ <\nu\Sigma_f>' = k_\infty' \ , \ <\nu\Sigma_f> = k_\infty \end{array}\right| \qquad (2.3)$$

Accordingly, the infinite multiplication factor is as follows. Thus,the definition of the coolant void reactivity ρ_v by coolant void fraction change from 0% to 100% in the equation of (2.1) can be reformed as follows.

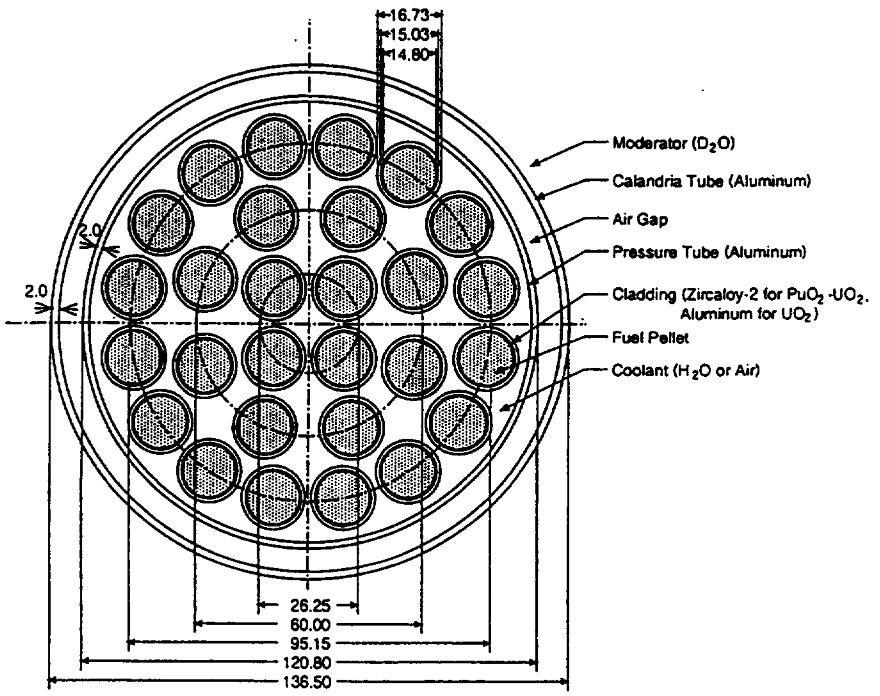


Fig. 1 Cross-sectional view of 28-rod fuel assembly. Dimensions are given in millimetres.

$$\rho_v = \frac{<\delta\nu\Sigma_f> - <\delta\Sigma_{a,f}> k_\infty - <\delta\Sigma_{a,s}> k_\infty}{k_\infty k_\infty'} \qquad (2.5)$$

In the equation of (2.5), the first term represents the component due to the nuclear fission for the coolant void reactivity, the second term the component due to the neutron absorption in the fuel for the same, and the third term the component due to the neutron absorption for the material except the fuel nuclides. In the ATR, the third term can be considered to represent almost the components of coolant because of the negligible neutron absorption for the material except coolant.

2.3 Verification of Calculation Code

It needs to be exactly evaluated by the JENDL-3.1 library incorporated WIMS-ATR code how the resonance absorption of Pu-239 and Pu-241 at around 0.3 eV contributes to the coolant void reactivity.

The behaviors of q^{49}_{25} of the neutron spectrum index of the plutonium resonance fission at around 0.3 eV and the thermal neutron flux distribution in the pressure tube, which are especially necessary for the present evaluation, are obtained for the lattice pitches of 22.5cm and 25.0cm for the parameters of the enrichment of plutonium fuel, plutonium isotopic composition ratio and coolant void fraction in the critical experiment using DCA. The cross sectional view of the clustered fuel which is served to the critical experiments is shown in Fig.1. Here, the q^{49}_{25} obtained from the DCA experiment is defined as follows.

$$q^{49}_{25} = \frac{\int_0^{\infty} \sigma_f^{49}(E)\,\phi(E)\,dE}{\int_0^{\infty} \sigma_f^{25}(E)\,\phi(E)\,dE} \qquad (2.6)$$

where, $\sigma_f^{49}(E)$, $\sigma_f^{25}(E)$:microscopic fission cross section for Pu-239 or U-235, respectively. In the range of the present analyzed fuel lattice, it is confirmed by experiments and analyses that the cadmium ratio of Pu-239 fission ranges from 10 to 20(the cutoff energy of cadmium ranges from 0.4eV to 0.5eV)[1]. Therefore, the integral range for the energy shown in Eq.(2.6) is almost up to the cutoff energy of the cadmium and the Eq.(2.6) represents the indicator showing remarkedly the behavior of the neutron at around 0.3 eV. Further, the thermal neutron flux distribution in the pressure tube is derived from the cadmium ratio of dysprosium reaction rate ratio[8].

The comparison of the experimental value of q^{49}_{25} in the 0.87 w/o MOX(R) fuel and calculated value with the WIMS-ATR code, for the parameters of the coolant void fraction and lattice pitch, is shown in Fig.2. From the results in Fig.2, the calculational precision of the cell averaged q^{49}_{25} value is estimated about ±2.2% and this is within experimental error ±4.0%. And it is also understood that the value of q^{49}_{25} indicating the behavior of the neutron spectrum of Pu-239 resonance at around 0.3 eV increases toward the outer fuel pin from the inner fuel pin in the voided lattice, while in contrast with this it decreases in the unvoided lattice. This is because the depression of 0.3 eV neutron flux recovers due to the thermal neutron scattering effect of the light water in the unvoided lattice, while this depression is promoted toward the inner fuel pin due to absence of the coolant in the voided lattice.

The comparison of the experimental value and calculated value of the thermal neutron flux distribution within the unit lattice is shown in the Fig. 3. In the experiment, the azimuth angle for square lattice is obtained in the directions of right angle and diagonal, but the calculated value of the calculated equivalent cylindrical lattice becomes identical in the direction of the circumference. The thermal neutron flux distribution is normalized at the value of the outer surface of the pressure tube. The calculation accuracy of the thermal neutron flux distribution is estimated about ±2.5% as a cluster fuel average from Fig.3, that almost approximates the experimental error ±2.0%. The thermal neutron flux falls greatly toward the fuel cluster from the heavy water moderator and the falling increases with increase in the fuel enrichment. For the absence of the thermal neutron scattering effect of the coolant in the voided lattice the thermal neutron flux distribution within the pressure tube becomes flattered compared to that of the unvoided lattice. In the excessive moderation lattice with 25.0cm pitch, due to the increased thermal neutron absorption effect of the coolant by the softened thermal neutron spectrum, the distribution is inclined to be falled more than that in the lattice with 22.5 cm pitch.

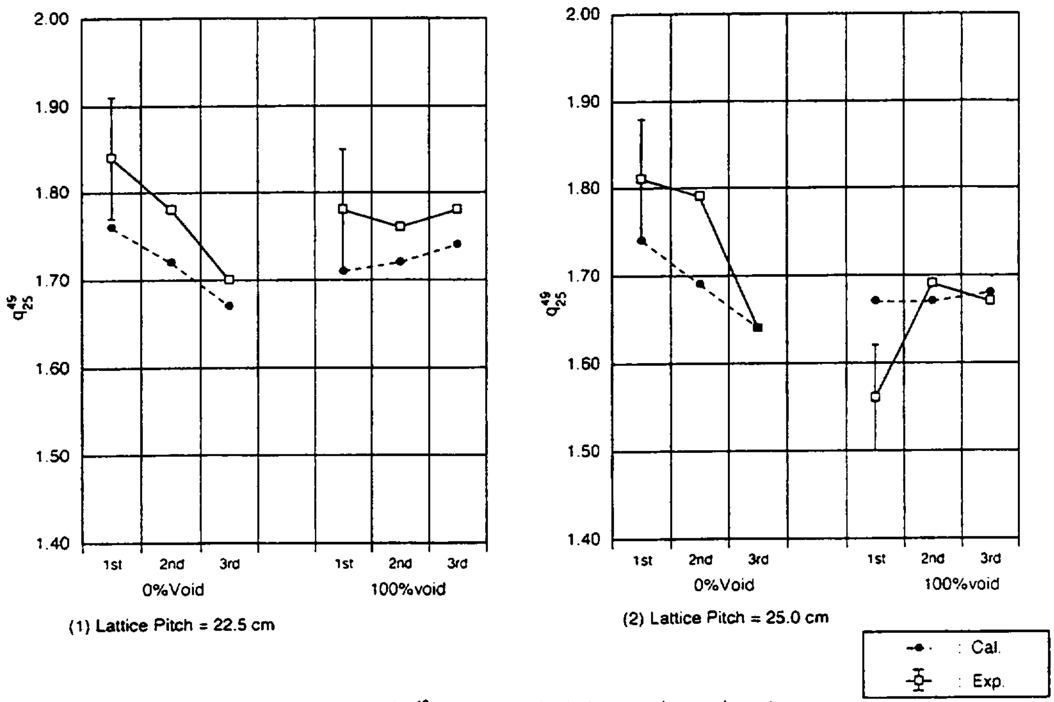
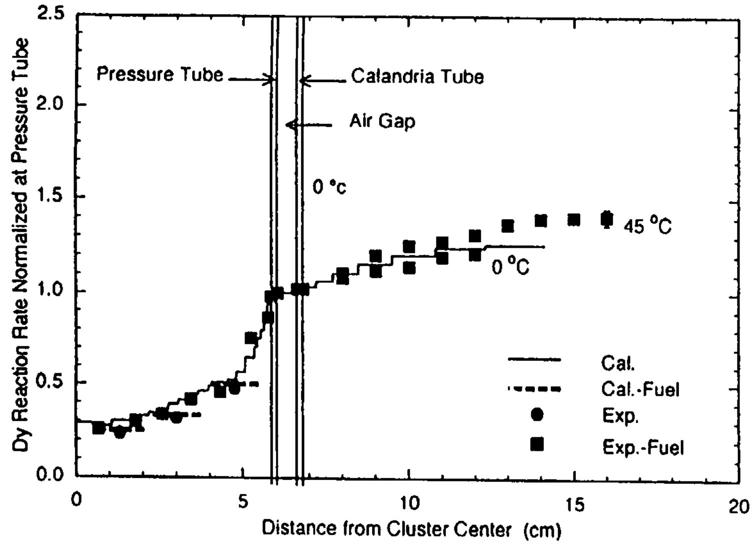
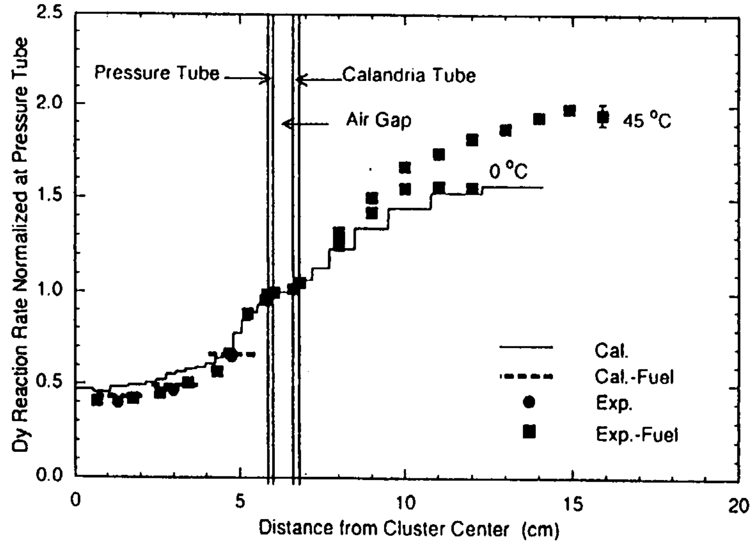


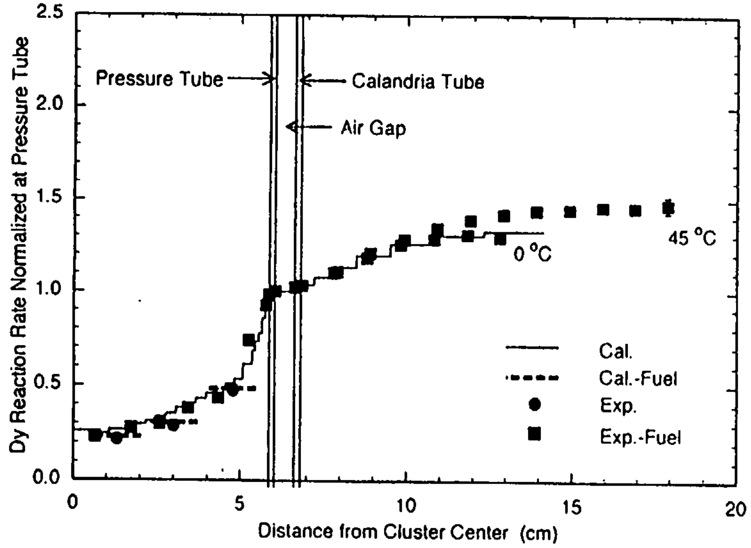
Fig.2 Comparison of q^{49}_{25} between calculations and experiments (0.87(R)wt% PuO_2-UO_2)



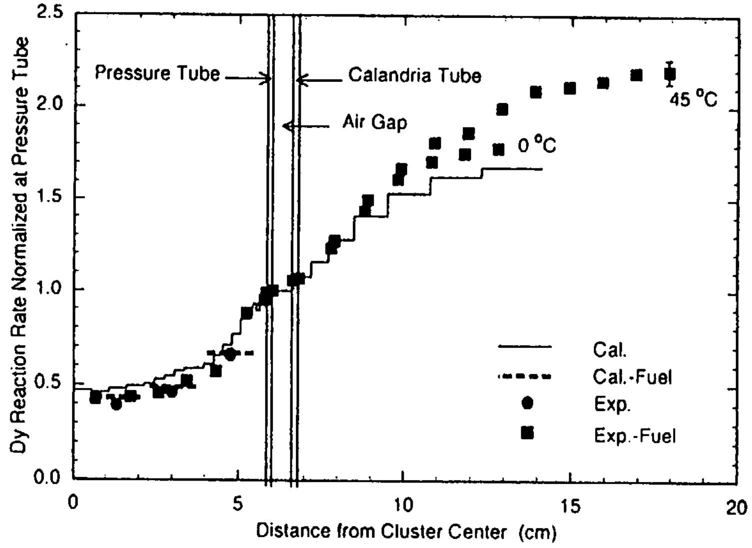
(1) 22.5cmPitch,0%Void



(2) 22.5cmPitch,100%Void



(3) 25.0cmPitch,0%Void



(4) 25.0cmPitch,100%Void

Fig.3 Intra-cell thermal neutron flux distribution
($0.87(R)wt\%PuO_2$-UO_2)

3. RESULTS AND DISCUSSIONS

3.1 Dependence of Coolant Void Reactivity on Σ_{a0}

The analytical results of the dependence of the coolant void reactivity in the infinite lattices for the ATR on the Σ_{a0} using the WIMS-ATR code are shown in Fig.4 (lattice with 22.5 cm pitch) and Fig.5 (lattice with 25.0 cm pitch) for the parameters of the kinds of fuel(UOX,MOX(S),MOX(R)). Also, the component ρ_v^f in the region for fuel and component ρ_v^c in the region for coolant contributed to the coolant void reactivity, that are separated according to the Eq.(2.4), were shown respectively.

The following are found from the results shown in Figs.4 and 5.

1) The coolant void reactivity ρ_v tends to exponentially shift toward negative side with increase in the Σ_{a0} to be saturated at around Σ_{a0} =0.5 cm^{-1}. This tendency is not dependent on the lattice pitch.

2) ρ_v^c shifts toward the more negative side with increase in the Σ_{a0}, but comes to the positive contribution for any value of the Σ_{a0} . This value, that depends almost on the fuel material concentration (corresponding almost to the Σ_{a0}), does not depend on the lattice pitch.

3) For the identical Σ_{a0} in each fuel lattice of the UOX and MOX, ρ_v^c comes always to the negative contribution for the MOX lattice rather than for the UOX lattice and comes slightly to the more negative contribution for the MOX(S) lattice.

4) ρ_v^f shifts toward the negative side with increase in the Σ_{a0} , but comes to the negative contribution for any value of the Σ_{a0} . However, it shifts to the positive side with increase in the lattice pitch.

5) ρ_v^f with the smaller lattice pitch comes to the negative contribution for the MOX rather than for the UOX, but there is seldom contribution difference between the S and R classes. ρ_v^f with the large lattice pitch comes to the same contribution for both the UOX and R class MOX fuel lattices, but comes to the more negative contribution for the S class MOX fuel lattice.

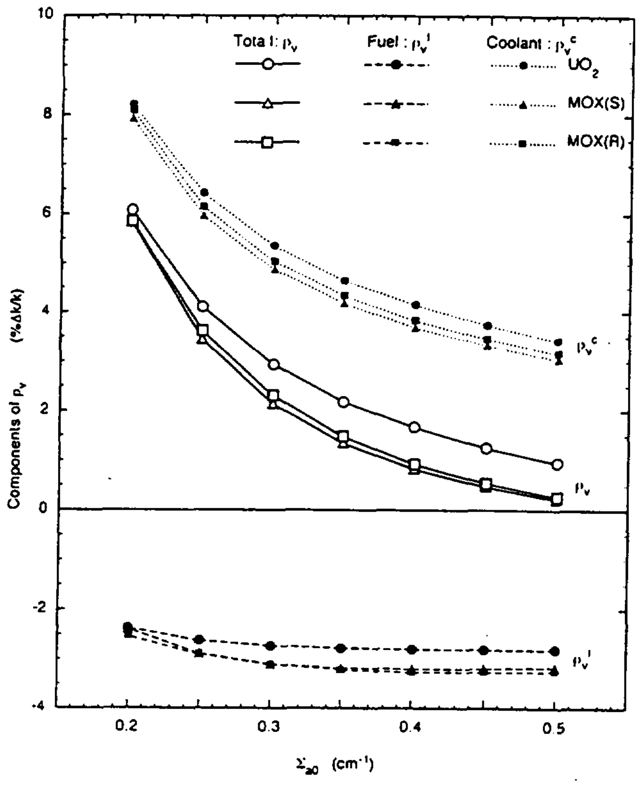


Fig.4 Components of coolant void reactivity (22.5cm square pitch lattice)

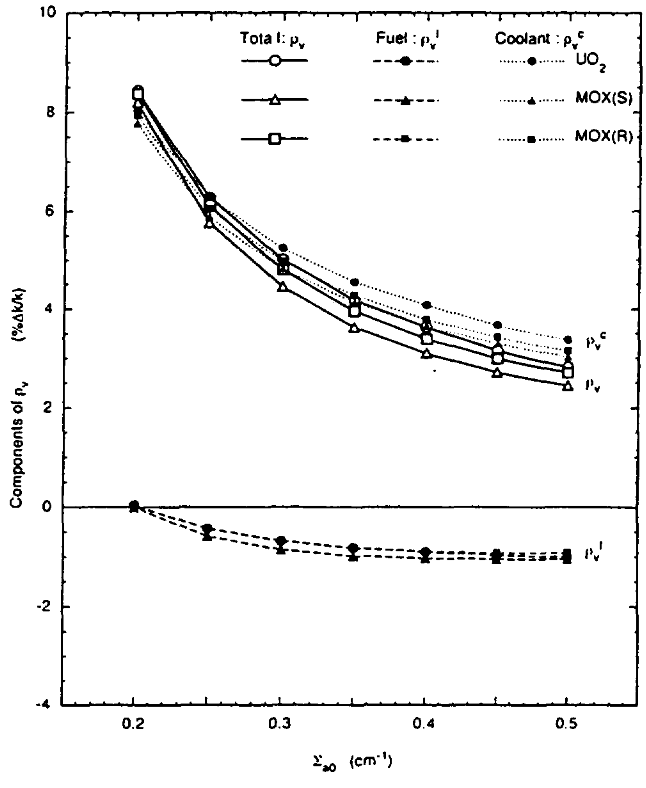


Fig.5 Components of coolant void reactivity (25.0cm square pitch lattice)

6) Therefore, the coolant void reactivity comes to the most negative side for the S class MOX fuel lattice and to the most positive side for the UOX fuel lattice.

3.2 Coolant Contribution to Coolant Void Reactivity

For clarification of the above-mentioned behavior, the analytical results of the dependence of the components with the hydrogen absorption contributions in the coolant to the coolant void reactivity on the neutron energy for each fuel lattice are shown in Fig.6. The thermal neutron absorption cross section of hydrogen in the coolant is almost of the 1/v type. Thus, ρ_v^c depends almost on the absorption effect of the sixth energy group for hydrogen, and the effects of oxygen and the like are negligibly small. From Fig.6, it is found that the contribution of the void reactivity of coolant is positive for any lattice and depends seldom on the lattice pitch.
And the larger the content of Pu-239 with the resonance absorption at around 0.3 eV is, the more it contributes to the negative side of the void reactivity.

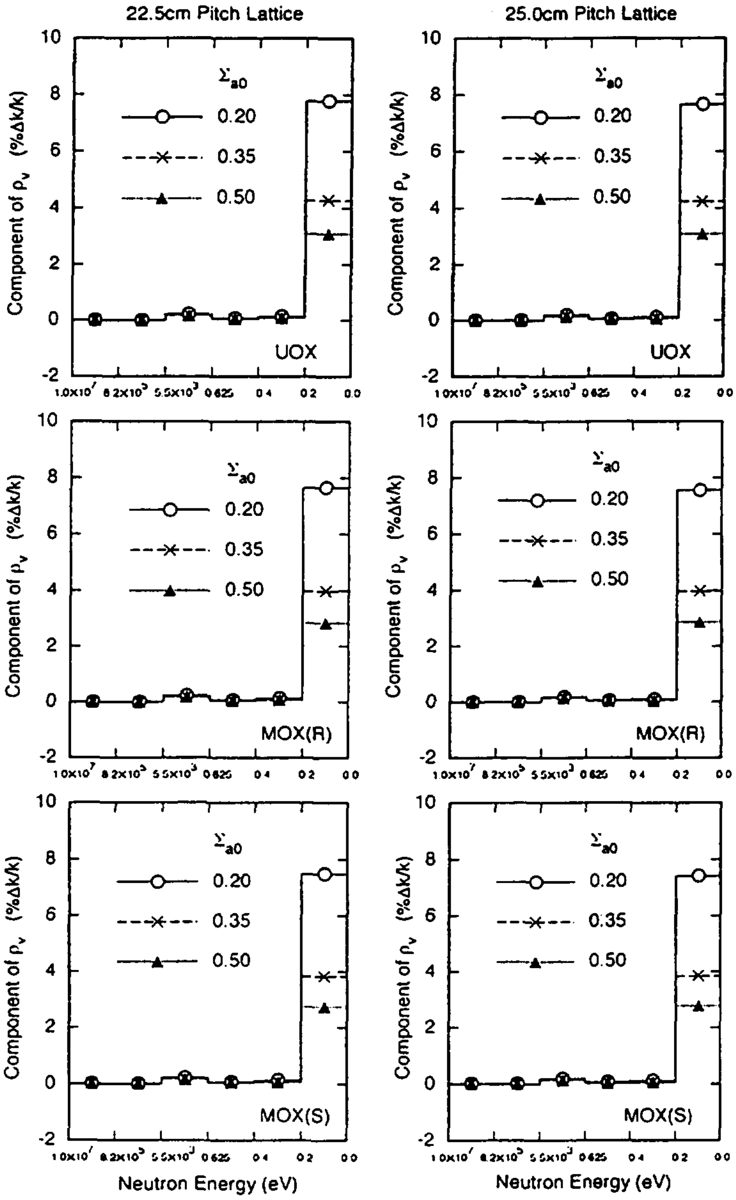


Fig.6 Contribution of hydrogen absorption in H_2O coolant to ρ_v

Fig.7 shows the analyzed thermal neutron flux distribution (0.5 eV and below) in the lattice normalized at the value of the outer surface of the pressure tube. The thermal neutron flux in the pressure tube is most depressive for the MOX(S) fuel lattice. And, with increase in the fuel material concentration(Σ_{a0}), the falling of the distribution increases. With increase in the lattice pitch,this falling increases. This is because as is evident from the thermal neutron spectrum in the region of coolant shown in Fig.8 the thermal neutron spectrum is softened more for the 25.0 cm pitch lattice and the thermal neutron shielding by the coolant becomes more effective. With increase in the effect of the thermal neutron shielding, the contribution to the void reactivity shifts to the negative side, but the effective thermal neutron absorption cross section increases due to the softer spectrum. Thus, the contribution of the void reactivity of coolant becomes almost equal in both pitch lattices.

3.3 Contribution of Coolant Void Reactivity by Neutron Absorption of Fuel

The contribution of the coolant void reactivity by each reaction rate of the separate fuel nuclide for each of energy group of the sixth group is shown in Fig.9 for examples of the MOX(S) fuel lattice. From the figure, the contribution of the neutron absorption rate for each fuel nuclide is almost a negative value above 0.625 eV, but the contribution can be a positive value below 0.625 eV of the thermal neutron energy region. As seen from Table 1, the content of U-238 in the fuel is more than about 85w/o in any Σ_{a0} to be remarkably high as compared to that for the other nuclide. Accordingly, for the giant resonance absorption cross section of U-238 in the third group (0.625 eV~5.53 keV) in the region for the resonance energy, the resonance capture rate increases due to the spectrum effect due to the increased void fraction and the component of this resonance absorption becomes a large negative value. But this effect shifts the coolant void reactivity to the positive side by the increase of Σ_{a0} value due to the decrease of the number density of U-238.

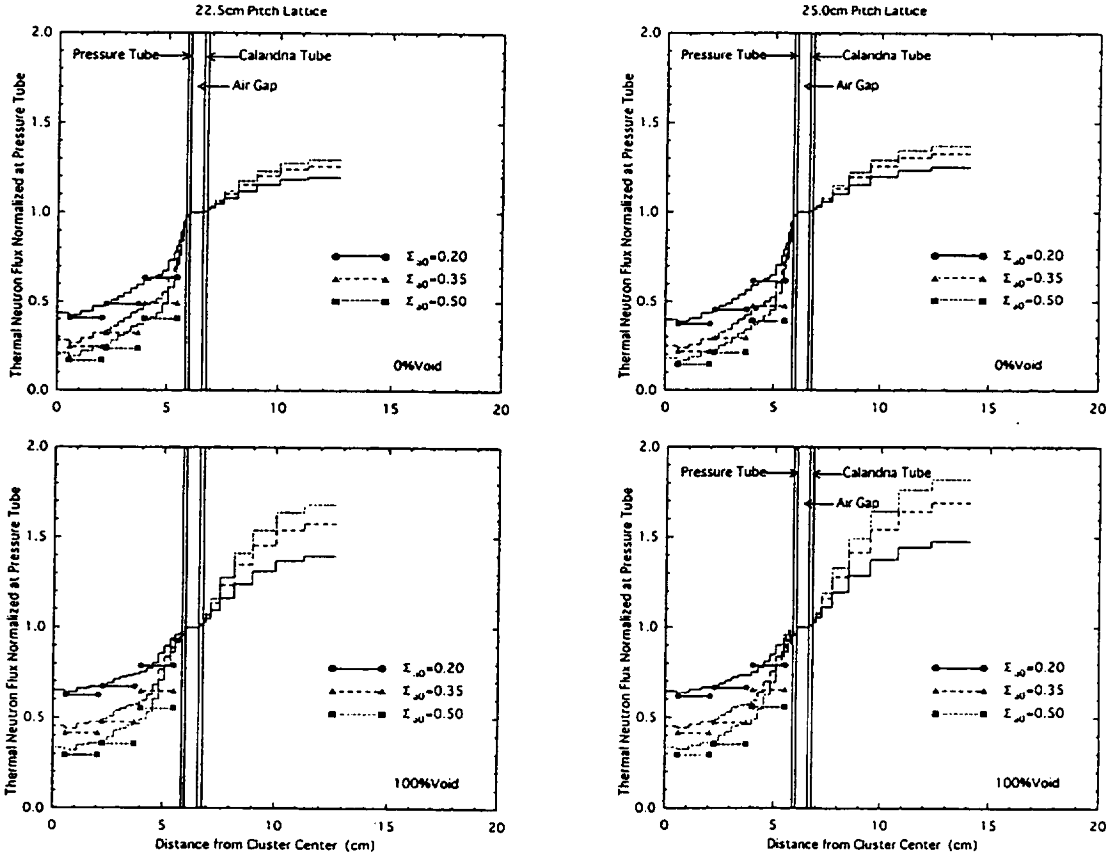
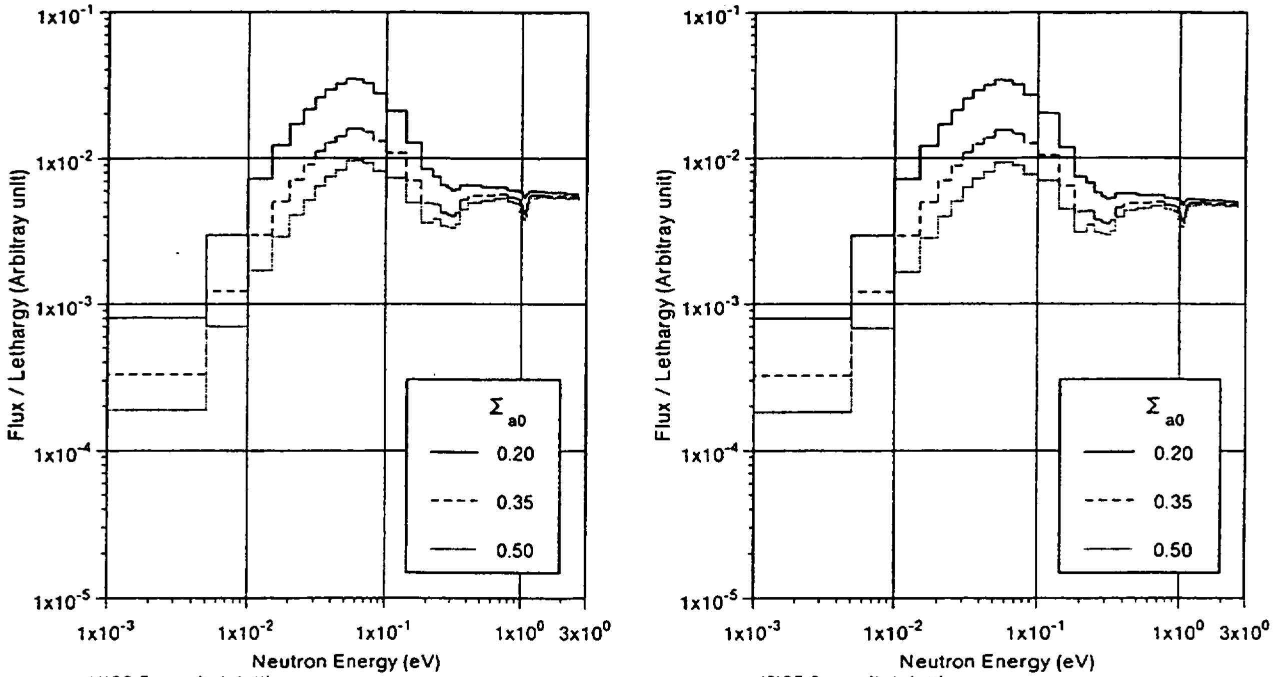


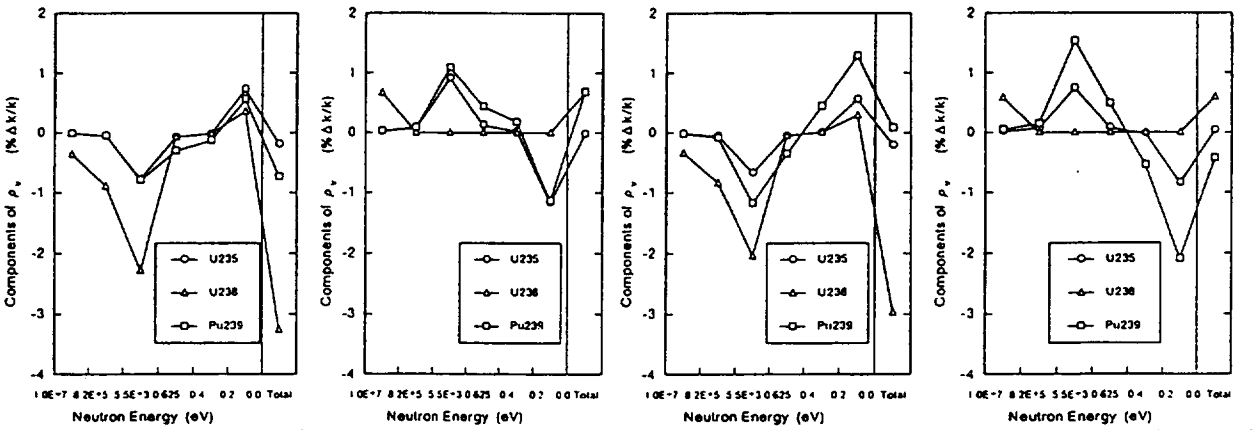
Fig.7 Intra-cell thermal neutron flux distribution
(MOX(S), Fuel)



(1)22.5cm pitch lattice

(2)25.0cm pitch lattice

Fig.8 Neutron spectra in the center region of fuel cluster (MOX(S),0% void)



(1) $\Sigma_a\phi$ Component (Σ_{a0}=0.35cm^{-1})

(2) $\nu\Sigma_f\phi$ Component (Σ_{a0}=0.35cm^{-1})

(3) $\Sigma_a\phi$ Component (Σ_{a0}=0.50cm^{-1})

(4) $\nu\Sigma_f\phi$ Component (Σ_{a0}=0.50cm^{-1})

Fig. 9 Component of ρ_v (22.5cm pitch lattice of MOX(S))

The contribution of the neutron absorption below 0.625 eV which is especially sensitive to the variation of the value Σ_{a0} is shown in Fig10 for each fuel nuclide. The contribution by the absorption for U-238 in the sixth energy group (0.0 eV~0.22eV)of the Maxwellian region shifts the void reactivity to the positive side with the increase of the value Σ_{a0}, but the value is small. On the other hand, in the case of U-235with the UOX fuel lattice the contribution of the absorption of the sixth group, with increase in the Σ_{a0}, shifts fairly to the positive side due to the restrain of the thermal neutron flux increase by the increase of the coolant void fraction. And the positive contribution in the sixth group for the MOX fuel lattice is considerably restrained mainly due to the parent material of the natural uranium. The contribution of the plutonium nuclides in the MOX fuel lattice can be considered as follows. Plutonium239 approximates uranium235 for the contribution in the sixth group and the value is almost the same between MOX(S) lattice and the MOX(R) lattice.

The contribution of the fifth group containing 0.3 eV energy region shifts largely the void reactivity to the positive side with the increase of value Σ_{a0}. This tendency is more striking for the MOX(S) lattice which has more plutonium239 content. This is because the effect of thermal neutron scattering due to coolant restores the depression of the neutron flux at around 0.3 eV and because the absorption decreases due to the exhaust of the neutron flux at this energy by the spectrum hardening with the increase of coolant void fraction. Plutonium241 is small in the content rate for the MOX(S) lattice, but it shows a similar tendency to plutonium239 for the MOX(R) lattice.

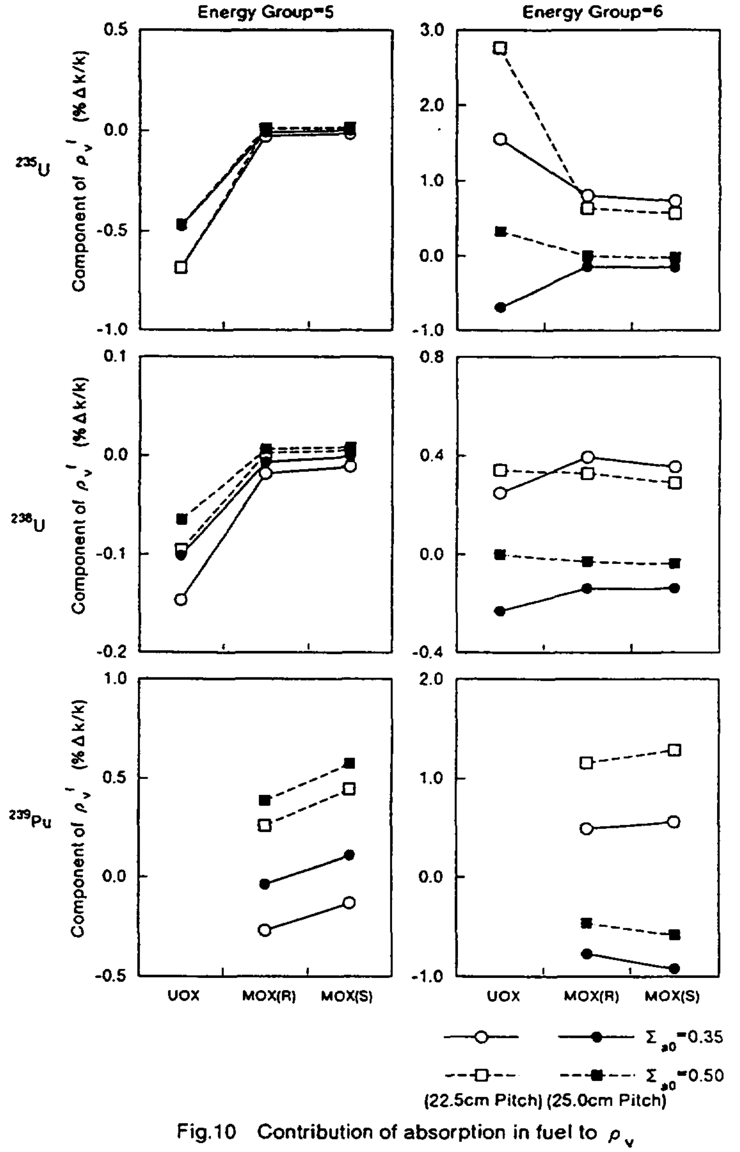


Fig.10 Contribution of absorption in fuel to ρ_v

3.4 Contribution of Void Reactivity by Fission of Fuel

The contribution of the fission in the thermal energy region is shown in Fig11. The contribution of the sixth group for U-235 of the UOX lattice, with the increase in the Σ_{a0}, shifts the coolant void reactivity to the negative side. This is because the fission reaction rate decreases due to the spectrum hardening by increase of coolant void fraction. In the case of the reaction rate in the sixth group for plutonium239 it approximates uranium235 for the contribution. However, in the case of the fifth group containing the resonance in 0.3 eV, the S class lattice shifts more to the negative side than the R class lattice. This is because the neutron flux at around 0.3 eV of Pu-239 resonance absorption becomes deeply depressed in the voided MOX fuel lattice.

In the case of reaction rate for plutonium241, because of the less content for the S class lattice, there is seldom contribution to the void reactivity, but for the R class lattice there are the small contribution in the sixth group. Uranium238, because of the only fast nuclear fission and the neutron spectrum hardened in the 100% void lattice, is natural to come to the positive contribution, but this effect decreases with increase of Σ_{a0} due to decrease of the number density of uranium238.

3.5 Contribution of Whole Reaction Rate for Fuel to Void Reactivity

The contribution of the whole reaction rate for the fuel to the void reactivity is as follows from the results of the study in the sections 3.3 and 3.4.

The negative contribution of the fuel is due to the giant resonance absorption of uranium238 (the second and third groups). However, these effects decrease greatly with increase in the Σ_{a0}. Thus, as

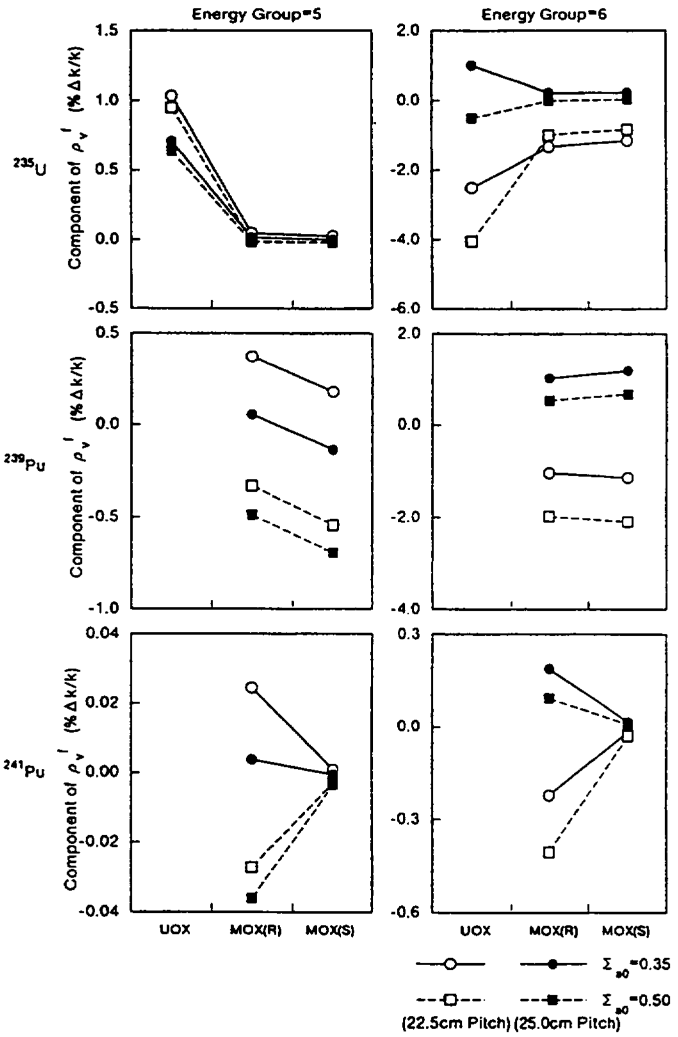


Fig.11 Contribution of fission in fuel to ρ_V

the Σ_{a0} increases, the contribution to the void reactivity of fuel is made more negative by the different contributed reaction rate for each fuel nuclide in the region of thermal neutron (the fourth,fifth and sixth groups).

In the UOX fuel lattice, with increase in the Σ_{a0}, the negative contribution of fission reaction rate for uranium235 fissions in the sixth group exceeds the positive contribution of absorption reaction rate to shift to the negative side to much extent and so the contribution to the void reactivity is made negative. This effect is smaller for the lattice with the soft thermal neutron spectrum of wider lattice pitch due to less effect of spectrum hardening by increase in the coolant void fraction. Accordingly, the contribution becomes more positive for the lattice with wider pitch. The same tendency can be also seen in regard to the contribution of the reaction rate for uranium235 in the MOX fuel lattice, but the effect is inclined to reduce compared to that of UOX fuel lattice because the spectrum hardening is smaller due to the parent material of natural uranium and due to the giant resonance at around 0.3 eV. The contribution for plutonium239 is almost the same for uranium235 in the sixth group. Especially due to the great change of the negative contribution for the fission in the fifth group containing 0.3 eV with increase in the Σ_{a0}, the MOX-S fuel lattice becomes more negative. This tendency becomes more remarkable for the lattice of wider pitch with the soft thermal neutron spectrum. And for the MOX-R fuel lattice with the wider pitch, the effect of plutonium241 is more remarkable because of the high content rate, eventually becoming the more positive contribution.

The reduction of the void reactivity to the negative side by is due to the presence of the resonance in 0.3eV for plutonium239. That is because the higher the content of plutonium239 is, the less the recovery effect for neutron density within the resonance energy in 0.3eV is with increase in coolant void fraction, so that the decreased resonance nuclear fission rate for plutonium239 contributes to the negative side for the void reactivity.

4. CONCLUSION

The mechanism of the effect for the reduction of the void reactivity to the negative side by plutonium239 is clarified by the detailed analyses using the improved WIMS code to be able to calculate the detailed thermal neutron spectrum containing the resonance in 0.3eV for plutonium239 and various components due to the reactivity relevant to the void reactivity. For the purpose of improving the accuracy of the calculated void reactivity, the nuclear data library for the WIMS code has been replaced with the JENDL-3.1 library. As the results of the analysis, for the identical macro thermal neutron absorption cross section for fuel, the void reactivity is reduced to the negative side more for the MOX with high content of plutonium239 in plutonium than in uranium due to the presence of the resonance in 0.3eV for plutonium239. That is because the higher the content of plutonium239 is, the less the recovery effect for neutron density within the resonance energy in 0.3eV is with increase in coolant void rate, so that the decreased resonance nuclear fission rate for plutonium239 contributes to the negative side for the void reactivity. This effect is more remarkable for the larger pitch lattice with softer thermal neutron spectrum.

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PROBABILISTIC SAFETY ANALYSIS

(Session 10)

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APPLICATION OF PSA LEVEL 1 FOR THE FUGEN PROTOTYPE ATR PLANT

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Abstract

This paper presents application of PSA Level-1 for Prototype Advanced Thermal Reactor Plant "Fugen" in consideration of the design characteristics of such reactor and verify the safety aspect of Fugen using thus established procedure. ATR resembles the boiling water reactor (BWR) in a number of points, but there are also some differences between the ATR and the BWR. Therefore, PSA procedure have been established by taking such difference into consideration and by referring to experience of PSA in USA and Japan. Moreover, the core damage frequency was calculated on Fugen by using thus established procedure. As the result, it was verified that results including the maximum value of the uncertainty estimation were found to be quite satisfactory against the target value of reactor damage frequency defined by International Atomic Energy Agency (IAEA).

1. INTRODUCTION

In Japan, a nuclear plant is licensed, in principle, through the process of Deterministic Safety Assessment. However, it has been known that Probabilistic Safety Assessment (PSA) which enables the quantitative evaluation of the safety aspect of a nuclear plant including the accidents exceeding design basis events is an effective method, complementing the deterministic safety assessment, and it is not unusual that the results of PSA are also taken into consideration in the course of issuing the final approval.

The objectives of this study are to establish the PSA procedure to be applied to an ATR in consideration of the design characteristics of such reactor and with the results of PSA implemented in the U.S.A. as well as PSA actually performed on a number of domestic light-water reactors for reference and, at the same time, to verify the safety aspect of the Fugen using thus established procedure. Safety researches for ATR are shown in the table 1-1. This study has been performed on the basis of the result of these safety researches. The part of these safety researches have been continued.

2. CHARACTERISTICS OF FUGEN

The Fugen is of a boiling water cooled and heavy water moderated Pressure Tube type reactor equipped with a safety system shown in Fig. 2-1 and Table 2-1. The Fugen is a direct cycle power generator plant and resembles the boiling water reactor (BWR) in a number of points such as that it uses about 7 MPa boiling light water for coolant and the steam generated in the reactor goes into steam drums and sent on to turbine generators after separated into liquid and vapor forms in the drum. But there are also some differences between the ATR and the BWR, therefore, PSA procedure has been established by taking such difference into consideration as follows:

TABLE 1-1 SAFETY RESEARCHES FOR ADVANCED THERMAL REACTOR

Category	Item
· Integrity of MOX fuel assembly structure and fuel rods	· Irradiation test of MOX fuel · Post irradiation examination · Irradiation test of segmented fuel for ramp test · Ballooning test of fuel cladding · Development of fuel failure detection etc
· Thermohydraulic aspects	· Test of Critical heat flux · Heat transfer test of natural circulation · Heat transfer test of sudden decrease of channel flow rate · Heat transfer test at LOCA · Safety analysis with statistical method etc
· System.Components	· Insertion test of control rods at earthquake · Hydrogen water chemistry · Post irradiation examination of test specimens of pressure-tube material · Development of remote-controlled pressure-tube monitoring equipment · Development of detector(microphone) for small break of pipes · Test of LBB(Leak Before Break) for inlet feeder pipes and outlet pipes · Test of pipe whip for inlet feeder pipes and outlet pipes · Development of remote-controlled monitoring equipment for inlet feeder pipes and outlet pipes etc
· Beyond design basis accidents	· Test of core coolability by coolant in primary loop under condition of beyond design basis accidents · Test of core coolability by heavy water moderator · Test of void formation in heavy water for evaluation of reactivity accidents under condition of ATWS · Test of integrity of the calandria tube and vessel after pressure tube rupture · Development of analysis method for thermohydraulic aspects in containment · Probabilistic Safety Assessment(PSA) · Accident manegement · Development of symptom-oriented Emergency Operating Procedures(EOP) etc

(a) The reactor core is to have two halves that shall be cooled separately with each independent cooling system.

The cooling system consists of two loops, each charging one half of the reactor core. Figure 2-2 shows the reactor core cooling system, which operates each of the two loops of Fugen coolant system at LOCA. If a LOCA occurs where the piping system may be ruptured in a point from which the coolant may be discharged, the system shall be scrammed and emergency measures shall be taken to shut off the Main Steam Isolation Valve and to inject coolant into the reactor core through the Reactor Isolated Cooling System (RCIC). Since coolant would be lost from the ruptured part of the damaged loop, the Emergency Core Cooling System (ECCS) shall be actuated to secure necessary coolant. The fuel in the other unruptured loop shall be injected coolant by RCIC and the decay heat removed by the Residual Heat Removal System (RHR). As seen from above, Fugen

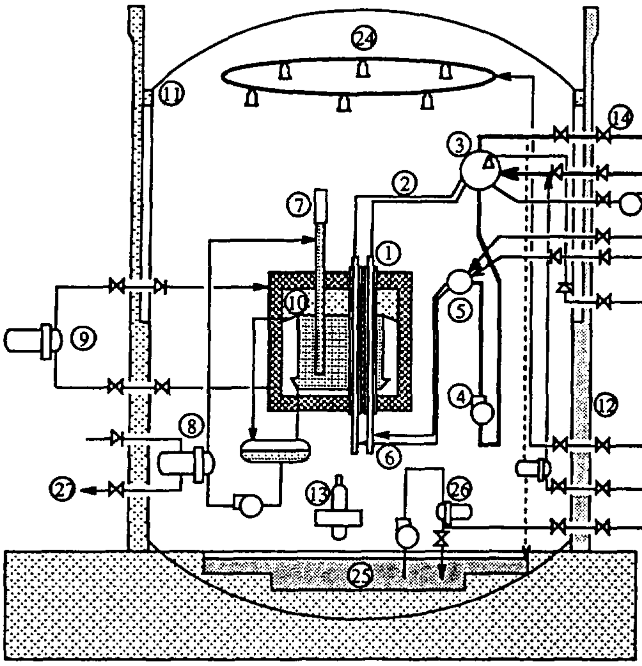


FIGURE 2-1 SCHEMATIC DIAGRAM OF COOLING SYSTEMS PROVIDED IN FUGEN

TABLE 2-1 THE SAFETY SYSTEM OF FUGEN

Reactor Shutdown System	Control Rod System (CRS) Helium Gas Circulation System (HEC)	
Number of Systems	2	
Number of DGs	2	
Composition of ECCS and Heat Removal System	Reactor Core Isolation Cooling System(RCIC)*	
	A-Residual Heat Removal System(RHR)	B-Residual Heat Removal System(RHR)
	A-High Pressure Coolant Injection System(HPCI)	B-High Pressure Coolant Injection System(HPCI)
	A-Low Pressure Coolant Injection System(LPCI)	B-Low Pressure Coolant Injection System(LPCI)
	A-Steam Release Pool Cooling System(SRPC)	B-Steam Release Pool Cooling System(SRPC)
	Accumulated Pressure Coolant Injection System(APCI)*	
	Moderator Cooling System(MCS)**	
	Containment Air Cooling and Cleanup System (CAC)**	

* There are two systems of dynamic devices or injection valves.
** MCS and CAC are planned to be effective on some specific initiating events only.

differs from BWR in that different cooling systems function and different series of processes take place in each of the two separate loops at the occurrence of an abnormal condition. At the occurrence of a transient event such as Loss of Offsite Power Accident, the same sequence of events take place in both loops with RHR and RCIC working concurrently.

(b) Heavy water used as moderator.

For moderator in Fugen, heavy water is used which is low in neutron absorption and able to burn fuel more efficiently as compared with light water. The Fugen is equipped with a vertical cylindrical tank called a calandria, which is filled with heavy water moderator and is separated from the coolant in the Pressure Tubes.

(c) The Fugen has 224 fuel channels called Pressure Tubes.

Fuel assembly used in the ATR is housed in 224 Pressure Tubes each one of which is independent from each other. Furthermore, each one of Pressure Tubes comprising this assembly is connected to the water- and steam-drums through an Outlet, respectively, Inlet tubes to carry coolant to each fuel component of the assembly. The existence of many more small size tubes than BWR is a particular feature. A safety systems have been designed for Fugen against the rupture of these tubes.

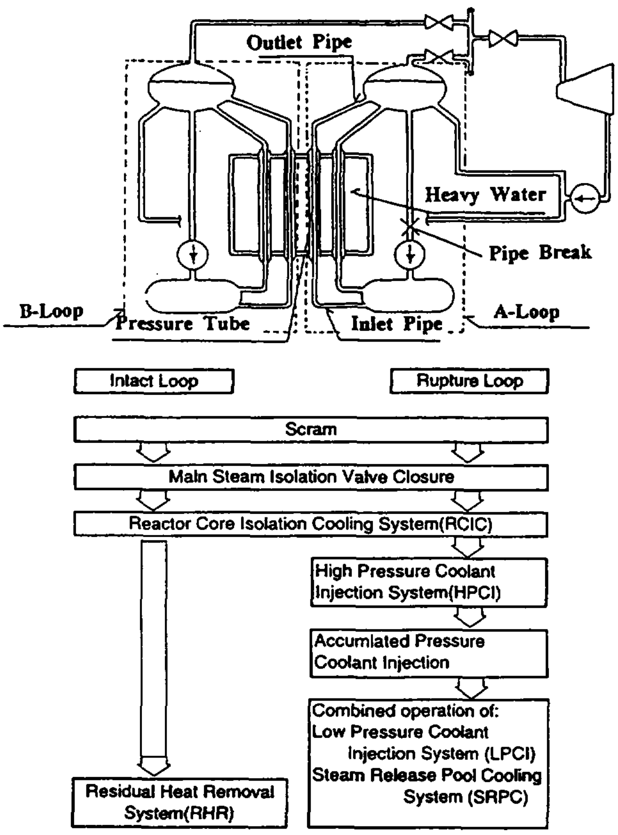


FIGURE 2-2 COOLING SYSTEM AT LOCA

3. DEVELOPMENT OF PSA LEVEL 1 PROCEDURE

3.1 Summary of Assessment Procedure

It has been first selected that a variety of events which initiate internal events that are the objectives of the intended assessment. It has been, then, drew out from the group of the Initiating Events the elements of safety functions required to prevent any damage to the reactor core and define the Success Criteria. Subsequently, Event Tree has been developed on the basis of the Success Criteria thus defined and performed a Fault Tree analysis (FTA) on the database of frequency of machine failure for the purpose of determining the branching ratio of the Event Tree. And, finally, it had been tried to determine the accident sequence with a probability to develop into a core damage along with the estimated level of core damage frequency through the Event Tree analysis. It shall be discussed in the following paragraphs how the characteristics of Fugen mentioned above can be reflected upon the evaluation method of Fugen.

3.2 Determination and quantification of Initiating Events

3.2.1 Selection of Initiating Events

At the selection of Initiating Events related to Fugen in this study, we have first developed the Master Logic Diagram of internal events. Then, these Initiating Events were divided into different groups such as the one to which the plant responses in the same way, the group of safety assessment events, the group of Initiating Events for prior PSA, etc.

The characteristics of ATR have been taken into consideration at the selection of the Initiating Events as follows:

(a) With respect to the reactor core divided into two halves with each loop for cooling, the characteristic of ATR is in the fact that different systems operate on each one of the two loops as mentioned in 2.(a) above, therefore, it was not taken reflected upon the selection of the Initiating Events but on the "Event Tree analysis."

(b) The use of heavy water for moderator was reflected on the Initiating Events. Such events as "Abnormal rise in the moderator temperature" or "Rupture of a heavy water tube" may be considered as events peculiar to ATR, therefore, were selected and grouped as Initiating Events: "Trouble in heavy water moderator system."

(c) ATR contains many small size tubes such as Outlet/Inlet tubes or Pressure Tubes which are some of the characteristic feature of Fugen and the possibility of the rupture of such tubes must be taken into consideration. Since the rupture of these tubes would be success criteria similar to the small LOCA, they would be grouped under the Initiating Events of the "Small LOCA" and shall be taken into consideration at the quantification of Initiating Events.

3.2.2 Quantification of Initiating Events

(1) Quantification of LOCA

The quantification of LOCA was done by reflecting the characteristics of ATR: the existence of a large number of small size tubes such as Outlet/Inlet tubes and Pressure Tubes. The method for quantification of the frequencies of rupture among general tubing and among the tubing peculiar to ATR is discussed below. It should be noted that the rupture of Outlet/Inlet and Pressure Tubes was handled as the small LOCA and included in the initiating frequency of the small LOCA.

1) Rupture of similar tubing to BWR

The frequency of rupture occurring to similar tubing to BWR other than those characteristic of ATR was quantified from actual results obtained from light water reactors (especially BWR) in Japan and U.S.A. (about 510 reactor years in total) with reference to prior PSA since they are basically the same in the composition.

2) Rupture of tubing characteristic of ATR

The frequency of rupture among tubing characteristic of ATR such as Outlet/Inlet pipes and Pressure Tubes was quantified from the results of PSA level 1 Study on prior LWRs[1] and also the results of another assessment based on probabilistic rupture mechanics.

(2) Quantification of transient events

1) Similar events to BWR

The frequency of transient events excluding Unusually heavy water moderator system that are characteristic of ATR was quantified from the results obtained from domestic light water reactors (especially BWR) and also from the actual operating results of the Fugen, (about 245 reactor years in total).

2) Events characteristic of ATR

"Unusually heavy water system" are events characteristic of ATR, therefore, were quantified on the basis of operating results of Fugen (13 reactor years in total) only.

The quantified frequency of Initiating Events thus obtained are shown in Table 3.2-1. The sum of the Initiating Event frequencies in this study for ATR determined from the above described source data came to be 4.3 x 10^{-1} reactor years which reflects the successful operating results of domestic LWRs and Fugen. The estimation of the intervals of Initiating Events was made using χ^2 distribution.

TABLE 3.2-1 INITIATING EVENTS FREQUENCY

Initiating Event	Frequency (1/reactor. years)
Large LOCA	4.4×10^{-5}
Intermediate LOCA	1.4×10^{-4}
Small LOCA	5.3×10^{-4}
Turbine Trip	2.7×10^{-1}
Failure of Heavy Water Moderator System	7.7×10^{-2}
Main Steam Isolation Closure	7.3×10^{-2}
Loss of Offsite Power	8.2×10^{-3}
Total	4.3×10^{-1}

3.3 Determination of Success Criteria

After extracting safety functions required to prevent core damage in reactors from Initiating Events selected as described in 3.2 above, the success criteria have been defined for each safety function. These success criteria were defined by investigation of design information and also by analyzing data on the basis of the authorization code of nuclear plants and other available information, as necessary. The definition of core damage in this study shall be deemed to have occurred when the clad surface temperature reaches over 1200°C which is in accordance with the current judgment criterion of the Deterministic Safety Assessment. The characteristics of Fugen were reflected upon the determination of the success criteria as follows:

(a) The fact that the core is divided into two halves each one of which is cooled by separate loop of coolant was dealt with by setting up a pair of success criteria, one for each loop. The state of the ruptured loop and unruptured loop will be different at the time LOCA, therefore, different criteria shall apply to the respective loops. The unruptured loop will display the same behavior at the time of LOCA as in the transient event, therefore, shall be subject to the same success criteria as in the transient event.

(b) The success criteria in consideration of another of the Fugen characteristics: the existence of many small size pipes and tubes such as Outlet/Inlet and Pressure Tubes, are handled in the way similar to that of the small LOCA since they are similar in the cross sectional area as well as the cooling system required.

(c) The use of heavy water for moderator was considered upon by incorporating the cooling effect on the core of the heavy water moderator system in the ATR-specific success criteria. The cooling function of the heavy water moderator system corresponds to about 28 MWt and there is a possibility that the reactor core may continue to be cooled through the following mechanism even after such other cooling systems as RCIC, RHR and ECCS were rendered unserviceable. Fig. 3.3-1 shows the principle of the heavy water core cooling system. After a reactor comes to scram as the result of LOCA or a transient event and the core cooling functions breaks down, the coolant within Pressure Tubes shall be lost through leakage or decay heat. Despite such loss of coolant from Pressure Tubes, however, the decay heat will be transferred from fuel rods to Pressure Tubes by conduction and from Pressure Tubes to calandria tubes by radiation and, then, on to the moderator, or heavy water, to complete the core cooling process. Furthermore, the core cooling effect shall further increase as the heat transfer to heavy water is further intensified by heat transfer through contacts between the Pressure Tube and Calandria tube coming into contact by ballooning of the Pressure Tube. A number of tests have been conducted by PNC (Power Reactor and Nuclear Fuel Development Corporation) to verify the above described mechanism[2] and we have analyzed the cooling effect of heavy water in ATR on the basis of the results obtained through the tests. Fig. 3.3-2 shows an example of such analyses made on the cooling mechanism of the reactor core with the heavy water moderator system. It was shown by the analysis of a situation: "Small break LOCA resulting from 2% ($0.002152m^2$) break of the downcomer flow area($0.1076m^2$) + Failure of ECCS", it is also expected that the required criteria shall be met since the water pressure within the recirculation system is so strong that Pressure Tubes will be subject to the ballooning effect which eventually increases heat transfer to the heavy water moderator system. As a result of investigation of application for PSA. On the basis of the results of the analysis, we have decided to incorporate the core cooling function of the heavy water cooling system into the success criteria in the case of the small LOCA and transient events (excluding the failure of the heavy water moderator system).

Table 3.3-1 shows the success criteria reflecting above mentioned considerations.

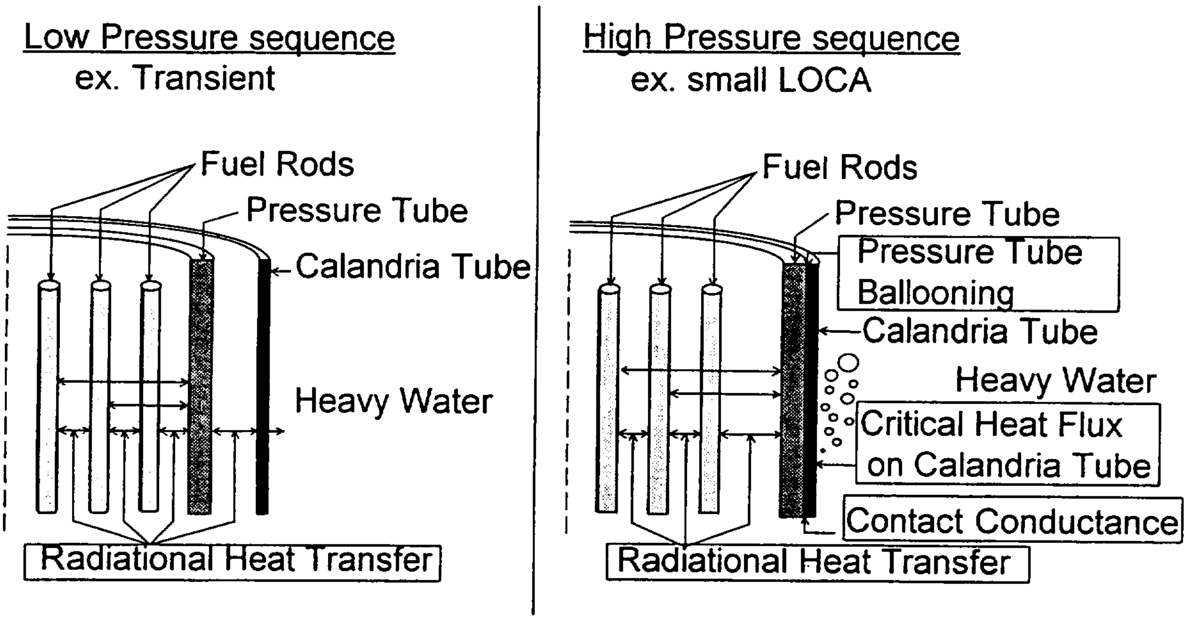


FIGURE 3.3-1 THE CONCEPT OF CORE COOLING BY MCS

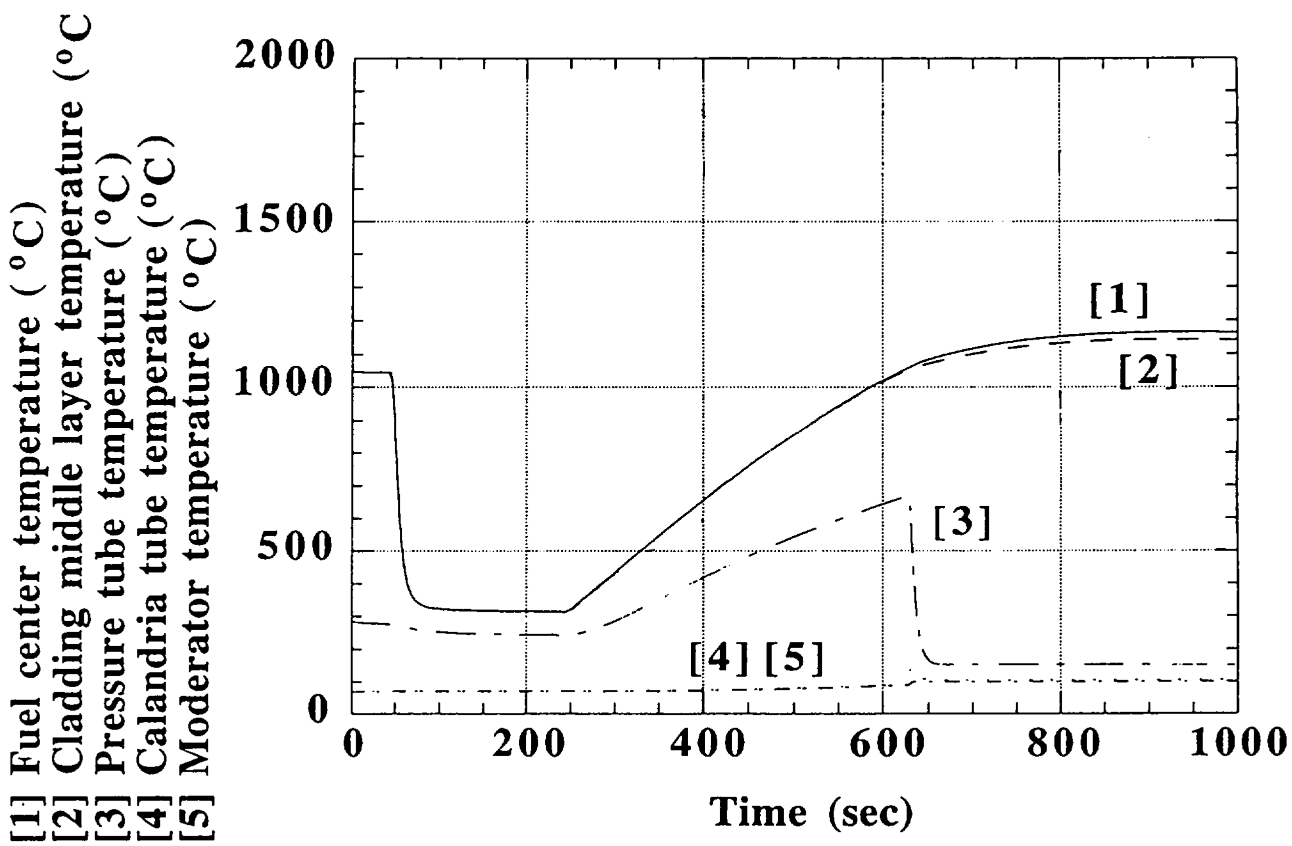


FIGURE 3.3-2 RESULT OF CORE COOLING BY MCS ANALYSIS

TABLE 3.3-1 SUCCESS CRITERIA FOR FUGEN *1

Protective function	Large LOCA*2		Intermediate LOCA*2		Small LOCA *2		Transient*2*3	
	Rupture Loop	Normal Loop	Rupture Loop	Normal Loop	Rupture Loop	Normal Loop		
Reactor Shutdown	[CR] or [HEC]		[CR] or [HEC]		[CR] or [HEC]		[CRD] or [HEC]	
Pressure Reduction + Coolant Injection	[LPCI]	[RCIC] or [HPCI] or [PR]*4	[LPCI]	[RCIC] or [HPCI] or [PR]	[HPCI + LPCI] or [PR+LPCI]	[RCIC] or [HPCI] or [PR]	[PCS]*5 or	[RCIC] or [HPCI] or [PR]
Heat Removal function	[SRPC]	[SRPC] or [RHR]	[SRPC]	[SRPC] or [RHR]	[SRPC]	[SRPC] or [RHR]		[RHR] or [SRPC]
Note	* 1 The core damage was defined at over 1200℃ of the peak clad temperature. * 2 MCS is expected as a mitigating system. * 3 CAC is expected as a mitigation system. * 4 PR means Pressure Reduction with the line by passing heat exchanger of RCIC. * 5 PCS is not expected at the Pressure boundary breaking such as failure of Safety Valve reclosure.							

3.4 Development of Event Tree

An Event Tree of seven Initiating Events was developed to delineate accident sequences to be considered for analysis purposes. The characteristics of Fugen were taken into consideration at the development of the Event Trees as follows:

(a) The notable feature of the Event Tree for Fugen is that it has been developed in such a way that two loops displaying two different sets of behaviors can be evaluated concurrently. Furthermore, as shown in Fig. 3.4-1, when the cooling function fails on the unruptured loop due to RCIC or RHR it will become necessary to cool the reactor core by ECCS not only by the ruptured loop but also by the unruptured loop so that a sequence shall develop where cooling by ECCS is required on both loops at the same time. Under such situations, ECCS, where RCIC and RHR have proved successful on the unruptured side, shall be capable of cooling if only one of the two systems is operative, however, both systems will have to be operative if the ECCS should be required also on the unruptured side. As can be seen from the above, the sequence of events will be more complicated with Fugen where the core is cooled by two loops of cooling system and the specific criteria required vary according to the progressed sequence of events . In order to correctly evaluate varying criteria in this study, we have developed the Event Tree by providing each tree with the heading indicating the related frontal system and, as a result, have come to end up with rather a large Event Tree to make us adopt "a large Event Tree with small Fault Trees" approach for Fugen assessment.

(b) A rupture to the large number of Outlet/Inlet pipes and Pressure Tubes that are characteristic of ATR has been effectively reflected by including it in the initiating frequency and small LOCA.

(c) The characteristics of Fugen of using heavy water for the moderator has been reflected as the core cooling effect of the heavy water moderator system in the success criteria. Consequently, the effect of the heavy water moderator system is referred to as the core cooling system in the heading of the Event Tree.

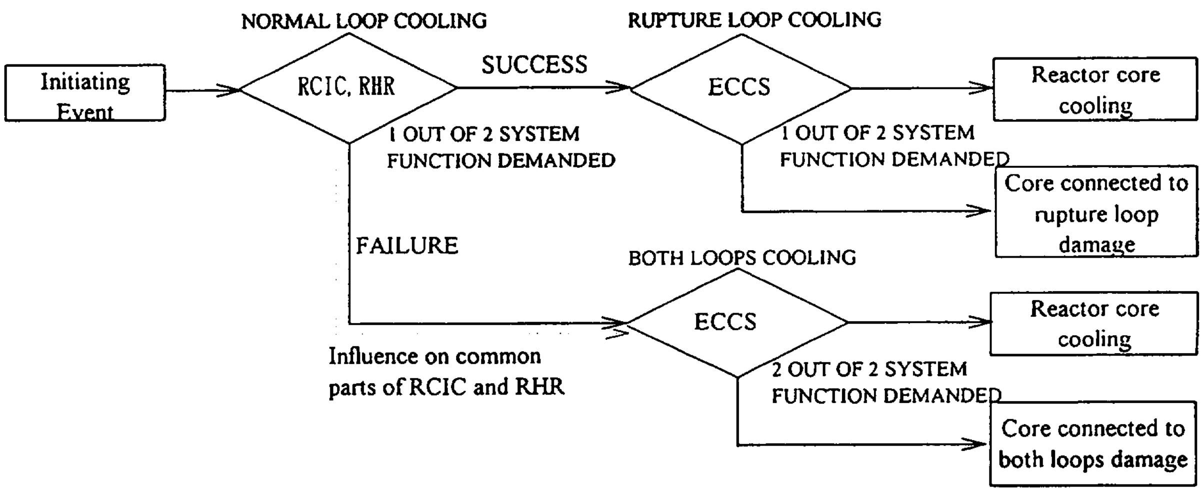


FIGURE 3.4-1 OUTLINE OF EVENT TREE ANALYSIS FOR TWO LOOP SYSTEMS

3.5 Development of Fault Tree

A Fault Tree has been developed with reference to functions required by each system in order to determine the branch probability on Event Tree. As seen in Paragraph 3.4, there are two loops of core cooling system in Fugen and the functions required of each system vary according to the progress of events on each loop. Consequently, the respective branch probabilities on the Event Tree vary depending on anticipated functions even on one and the same system. Under such circumstance, the Fault Tree has been developed using each function emerged on each system in the course of analyzing the Event Tree as the top event.

3.6 Preparation of Database

3.6.1 Component Database

A database for reliability assessment has been developed for the purpose of Fault Tree analysis. It was prepared under the following policy:

1) Whenever there are any data available on the failure of facilities and equipment which reflect the actual results of operation in Japan, such data shall be used before anything else.

2) If there is no domestic database available, ASEP Common Data[3] used for the assessment of NUREG-1150[4] in connection with machinery parts shall be used and, for electrical parts, IEEE Std-500[5] rich with data related to equipment type and failure mode shall be utilized as the basic data.

3) When any necessary data cannot be found in the above mentioned published databases or data found in above were not adequate for the intended purpose, then, LERs (License Event Reports) [6] or WASH- 1400[1] shall be applied.

3.6.2 Common Cause Failure

There may be considered two types of common cause failure: (1) concurrent failures of a number of machines or systems due to a fact that such machines or systems are supported by a common device or support system, and (2) concurrent failures of a number of machines that have been produced under the same specification and manufacturing process and used in the same environment. For the Fault Tree or Event Tree assessment of the machines or systems fallen under (1) above, analysis shall be made by the minimal cut set method or Zion method[7] in consideration of their dependency. Machines fallen under (2) above shall be analyzed by using the β-factor method applied to NUREG-1150[4].

3.6.3 Maintenance and examination

The unavailability of a system for assessment due to a maintenance service shall be quantified by using the following formula:

$$Pm = \Sigma\ (fm.Tm)$$

where:
fm = The frequency of maintenance service (10 times the failure ratio of the subject equipment) [8]
Tm = Average restoration time of the subject machine

Unavailability due to any examination shall be handled in the same way as above.

3.6.4 Human factors

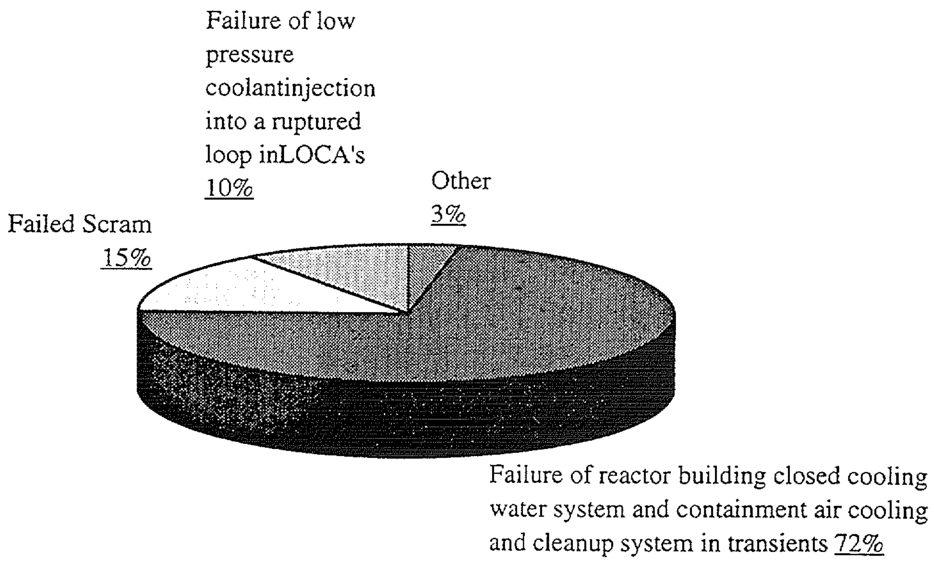
We have also taken into consideration human errors in operation by operators before and after an accident in this study. Human errors have been quantified by using the database of NUREG/CR- 1278[9], THREP procedure and the reliability curve used in NUREG/CR- 4772[10].

3.6.5 Functional Recovery

In case of a machine failure, it is possible to repair the machine to recover its functions if there is ample time available. We have carried out assessment with consideration of such possibility for recovery for some machines such as diesel generator. The success ratio of the functional recovery of a machine shall be assessed by using the following formula:

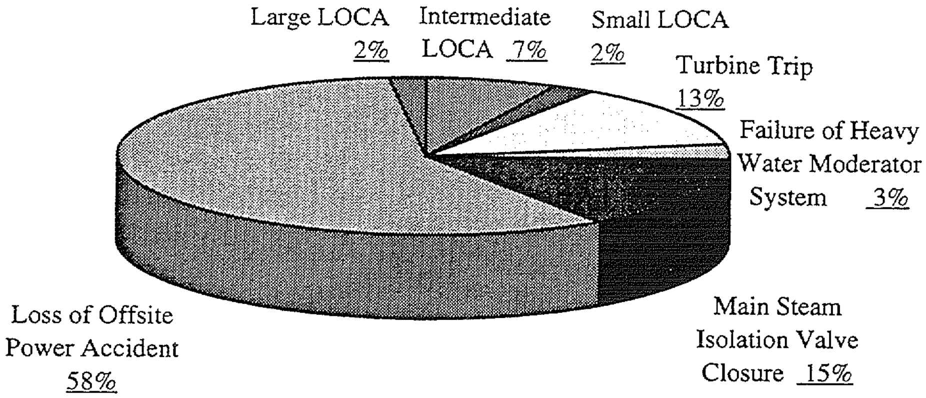
$$Pr = \exp(-T/t)$$

where:
T = time available for functional recovery
t = the average repair time of the subject machine



(2) DETAILED CLASSIFICATION OF CORE DAMAGE FREQUENCY BY ACCIDENT SEQUENCE

FIGURE 4-1 RESULT OF PSA LEVEL 1 FOR FUGEN



(1) DETAILED CLASSIFICATION OF CORE DAMAGE FREQUENCY BY INITIATING EVENT

3.7 Assessment of the frequency of reactor damage

The reliability of each system was calculated by Fault Tree analysis on the basis of the developed database. An Event Tree analysis was then carried out with the values obtained through Fault Tree analysis or other process for analyzing the branch probability of the Event Tree. Both point estimation and section estimation based on Monte Carlo method were carried in this analysis.

4. ASSESSMENT RESULTS

As the result of the trial calculation of the reactor damage frequency in accordance with the results obtained through Event Tree analyses, it was verified that results including the maximum value of the uncertainty estimation were found to be quite satisfactory against the target value of reactor damage frequency defined by International Atomic Energy Agency (IAEA).

Fig. 4-1 shows the reactor damage frequency grouped by Initiating Event and also by accident sequence. Events initiated by Transients account for 89% of the reactor damage frequency by Initiating Event. Significant reasons for the reactor damage frequency by accident sequence were found to be the failure to remove decay heat in transiens (72%). In a few years, the assessment results will be changed by the system improvement.

5. CONCLUSION

The PSA procedure for ATR Plant was established in this study on the basis of PSA performed on plants in U.S.A. and domestic LWR. Furthermore, the reactor core damage frequency was calculated on ATR by using thus established procedure. As the result, it has been proved that the safety aspects of ATR are appropriately secured when compared with other LWRs.

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OPERATING EXPERIENCE OF HWRs

(Session 11)

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OPERATIONAL EXPERIENCE WITH THE FIRST EIGHTEEN SLIGHTLY ENRICHED URANIUM FUEL ASSEMBLIES IN THE ATUCHA-1 NUCLEAR POWER PLANT

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Abstract

Atucha I is a 357 Mwe nuclear station, moderated and cooled with heavy water, pressure vessel type of German design, located in Argentina. Fuel assemblies (FA) are 36 active natural UO2 rod clusters, 5.3 meters long and fuelling is on power. Average FA exit burnup is 6 MWd/kg U. The reactor core contains 252 FA. To reduce the fuel costs about 6 MU$S/yr a program of utilization of SEU (0.85 %w U235) fuel was started at the beginning of 1995 with the introduction of 12 FA in the first step. The exit burnup of these FA is ≈ 10 MWd/kgU. It is planned to increase gradually the number of them up to having a full core with SEU fuel, with an expected FA average exit burnup of 11 Mwd/kgU. The SEU program has also the advantage of a strong reduction of spent fuel volumes, and a moderate reduction of fuelling machine use. This paper presents the satisfactory operation experience with the introduction of the first 12 SEU fuel assemblies and the planned activities for the future. The fresh SEU fuel assemblies were introduced in six fuel channels located in an intermediate zone located 136 cm from the center of the reactor and selected because they have higher margins to the channel powers limits to accomodate the initial 15 to 20 % relative channel power increase. To verify the design and fuel management calculations, comparisons have been made of the calculated and measured values of the variation of channel ΔT, regulating rods insertion and flux reading in in-core detectors near to the refuelled channel. The agreement was good and in most of the cases within the measurement errors. Cell calculations were made with WIMS-D4, and reactor calculations with PUMA, a fuel management 3D diffusion program developed in Argentina. With SEU fuel with a greater burnup in the central, high power core region, new operating procedures were developed to prevent PCI failures in fuel power ramps that arise during operation. Some fuel rod and structural assembly design changes were introduced on the first 12 SEU FA. Other specific design changes are also being implemented in the next SEU FA series in order to improve the fuel behaviour at higher burnups and in power ramps.

1 INTRODUCTION

Atucha I is a 357 Mwe (gross) nuclear station, pressure vessel type designed by Siemens (Germany), moderated and cooled with heavy water, located in Argentina, 120 km NW of Buenos Aires. Fuel assemblies (FA) are 36 active natural UO_2 rod vertical clusters, 5.3 meters long and fuelling is on power. Average FA exit burnup is 6 MWd/kg U and maximum rod burnup (m.r.b) ≈ 8.4 MWd/kgU. With the objective of reducing the fuel costs about 6 MU$S/yr (app. 25 %) a program of utilization of slightly enriched uranium (SEU) (0.85 w% U235) fuel was started at the beginning of 1995 with the gradual introduction of 18 FA during the year. Six of them were taken out of the reactor with an average exit burnup of ≈ 10 MWd/kgU.

It is planned to increase gradually the number of them up to having a full core with SEU fuel, with an expected FA average exit burnup of 11 Mwd/kgU (m.r.b.≈14 MWd/kgU). The SEU program has also additional advantages of a strong reduction (almost one half) of spent fuel volumes, and a moderate reduction of fuelling machine use. This paper presents the operation experience with the irradiation of the first 18 SEU fuel assemblies and the planned activities for the future.

2 BRIEF DESCRIPTION OF THE ATUCHA-I REACTOR

The Atucha I reactor has a gross electrical power of 357 MWe and a thermal power of 1179 MWth. The reactor core contains 252 coolant channels which contain the FA and separate the coolant from the moderator. The average coolant temperature is 280°C and the average moderator temperature is 190°C. A section of the core can be seen in fig.1.

Power regulation is made through six absorber rods, three made of hafnium, usually called black, and three made of steel, called gray. The rods are inserted with an angle of 15 to 21° with the vertical. They normally have an insertion corresponding to an excess reactivity of 6.5 mk in normal operation. Additional 21 hafnium rods are used for shutdown purposes. Power measurements for the regulation are obtained with four out of core compensated ion-chambers.

The fuel movement scheme is radial, and the core is divided in three approximately concentric annular regions. The fresh fuel is introduced in the intermediate zone, left there until it reaches 2.7 MWd/kgU, transferred to the central zone until it reaches 5.1 MWd/kgU, then moved to the outer zone, from where it is taken out at 6 MWd/kgU. In some cases fuel is moved through 4 positions instead of three to reduce power ramps in the fuel.

The coolant flow in the fuel channels is reduced from the center to the periphery of the core according to the channel powers, in such a way as to have approximately constant outlet channel temperatures. The temperature increase in the channels at full power is about 35 °C. No coolant boiling is allowed at the channel outlets. To obtain that there are 8 "hydraulic regions" with different nozzles, numbered 1 to 8, from the periphery to the center. To keep an adequate margin to outlet boiling, channel power limits are defined for each hydraulic region.

The reactor has outlet temperature measurements in 28 channels, and 41 in-core vanadium flux detectors give indications of local flux to the operators.

3 PROGRAM OF INTRODUCTION OF SEU FUEL

A program of introduction of SEU FA was prepared, increasing gradually the fraction of the core with SEU fuel up to having a full core with this type of fuel.

Phase 1 consists in the introduction of three groups of six SEU F.A. not exceeding twelve at any time in the core. It began in January 1995 and is expected to last until May 1996. From then, in Phase 2 the number of SEU FA in the core will gradually be increased up to sixty and it is expected to cover until around July 1997. In Phase 3 the number of SEU FA in the core will be gradually increased from sixty to a full core. This period is expected to last for about three years.

The fresh SEU FA were introduced in six predetermined channels (shown in fig.1) selected because thay have the following features:

- A larger margin to the channel power limits which is important to accomodate the higher power increase when introducing fresh SEU fuel.

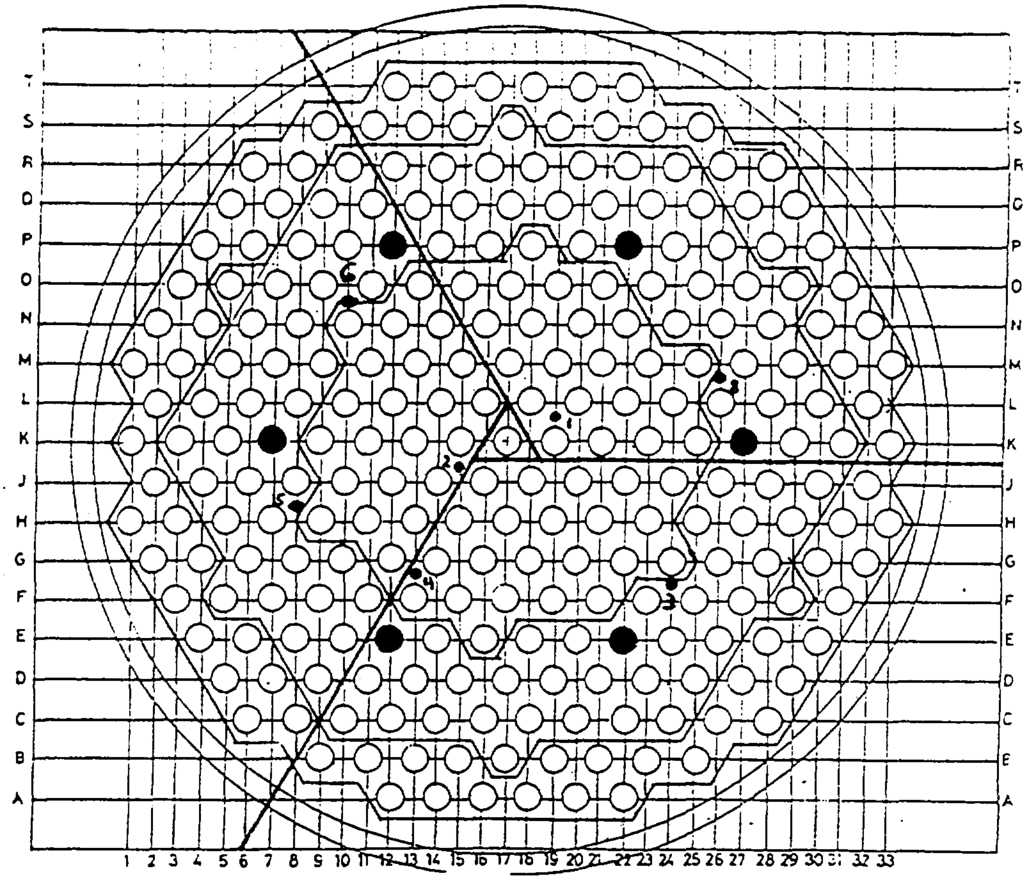

Fig. 1: Section of the Atucha-I core indicating refuelling zones and fuel channels selected for the introduction of fresh SEU fuel in Phase 1 of the irradiation program

● Flux detector assemblies

- The channel power is relatively high, which reduces the irradiation time, until the FA is transferred to another position.

- They have outlet channel temperature measurements and five out of the six have in-core detectors in the vicinity. This allows taking measurements of outlet temperatures and in-core detectors to compare with calculations.

The main objectives of the Phase 1 of the irradiation program are:

- To verify the performance of the SEU fuel in the core with exit burnups close to the values expected for the equilibrium full SEU core. In particular, to verify the behaviour in power ramps arising in refuelling operations, reactor power increases, and startups from low power.

- To verify predictions of neutronic calculations like reactivity gain, channel power increase and detector flux increase when introducing SEU fresh FA.

- To test operating procedures developed for SEU fuel.

4 ECONOMIC ASPECTS

It is well known that in heavy water reactors, initially designed for natural uranium fuel, a slight increase in enrichment from 0.711 w% U235 to values from 0.85 to 1.2 % has a strong improvement in fuel exit burnup and fuel economy. This has already been reported for Siemens pressure vessel type and CANDU type heavy water reactors (see for example refs. 1 and 2).

In Argentina the Atucha-I fuel is manufactured by CONUAR, an Argentine Company and because of the type of design, assembly, dimensions and small scale its fabrication cost is high compared with other reactors of natural uranium and heavy water. For that reason any improvement in the extension of the life of the fuel in the reactor has an important economic impact in the unit energy cost.

Basically the advantages of using SEU fuel are reflected in four areas:

4.1 Reduction in the uranium mineral consumption

Considering an average load factor of 0.85 the number of F.A. consumed per year is 397. The uranium mass per FA is 153.4 kg which implies a consumption of ≈62.2 Mg of U in UO2 (or 63.5 Mg of U in U3O8).

For the condition of an equilibrium full SEU core, under the same assumptions, the annual fuel requirements are reduced to 216.8 FA, corresponding to 33.9 Mg of SEU 0.85. At the present time the 0.85 % enriched UO2 was obtained by mixing 1.75 Mg of U in 3.4% enriched UO2 (bought in the international market), with 32.2 Mg of U in natural UO2 produced in Argentina. This implies a 48% savings in the natural UO2 consumption (considering the purchasing of the 1.75 Mg of the 3.4 % enriched UO2).

4.2 Reduction in the Unit Energy Cost

Presently the component of the unit energy cost due to fuel (front end fuel cycle cost) is about 8.9 U$S/MWh. The reduction of the number of FA required per year, implies a corresponding decrease in manufacturing costs. This reduction is not linear with the reduction in the number of FA mainly because of two reasons.

i) The SEU UO_2 cost has a higher cost per unit mass than natural UO_2

ii) The fixed costs of the fuel manufacturing plant affect the manufacturing costs per FA when changing the production scale.

Considering these factors the front end fuel energy cost for the SEU equilibrium core is estimated at about 6.6 U$S/Mwh which implies a reduction of about 25 %.

4.3 Reduction in the Spent Fuel Volume

The reduction in the consumption of FA implies a decrease in the irradiated fuel volume of about 45 %. Although detailed estimations of final spent fuel storage costs have not been done, if values of 100 U$S given for CANDU fuel in [3], are used as a lower bound, we obtain minimum annual savings of 2.8 MU$S for this concept.

5 FUEL MANAGEMENT AND SAFETY STUDIES

5.1 Reactor Physics and Fuel Management Studies for Phase 1 and 2

In Phase 1 (up to 12 SEU FA in the core) the situation can be considered as a relatively small perturbation of the natural uranium core, except for the larger channel powers in the six channels where the fresh SEU FA are introduced.

The reactor physics studies were done with the British lattice code WIMS-D4 [4], and the 3D reactor code PUMA [5], developed in Argentina, and used for the fuel management calculations of the Atucha-I and Embalse operating stations.

Basically the objective was to review and analyze the effect of the SEU fuel in stationary and transient operating conditions.

The main results can be summarized as follows:

Fuel management studies were done in two steps. The first step was with "time-average" type calculations for a given fuel movement strategy. This calculations provide the burnups at which the fuel is moved from one region to other and the exit burnup. It also provides time averages of the flux and power distributions. It does not give fluctuations due to refuelling operations. These types of studies are later completed with detailed refuelling simulations for 6-12 months which permitted to make final adjustments.

Before the beginning of Phase 1, a six month detailed simulation was made to confirm the fuel strategy chosen. Time average studies for configurations with 12, 30, 60 and 252 FA (full core) have been completed. A detailed simulation was done for the transition from 12 to 60 SEU FA.

5.2 Safety Related Studies

To study the effect of the presence of the SEU fuel in parameters related to safety like void reactivity coefficients, fuel temperature reactivity coefficients and the effective delayed neutron fraction β_{eff}, regulating and shutoff rod differential and total reactivity worth, calculations were done with WIMS and PUMA using representative fuel configurations for each Phase of the program (see table 1).

For the safety studies a complete revision of the Safety Report (SR) was prepared and sumbitted to the Argentine Licensing Authority. An internal independent revision group was established in the Project to review the SR and the technical reports associated to it. As the SR has a deterministic methodology, the stationary, transient, and accidental conditions of operation were analyzed considering the following differences and similarities between natural and SEU FA.

a) The global core activity and the core or FA decay heat, immediately after a plant shutdown (calculated with ORIGEN [7]) for a core with 60 SEU FA is practically the same as in a natural uranium core or FA with the same power.

TABLE 1:EFFECT OF THE PRESENCE OF SEU FUEL ON PARAMETERS RELATED TO SAFETY ASPECTS

PARAMETER	NATURAL CORE	12 SEU FA	60 SEU FA	252 SEU FA	UNIT
REACTIVITY CHANGE DUE TO COOLANT VOID	12.67	12.66	12.69	12.71	mk
FUEL TEMPERATURE REACTIVITY COEFFICIENT	-0.664	-0.661	-0.640	-0.611	.01 mk/°C
MODERATOR BORON REACTIVITY COEFFICIENT	-5.35	-5.32	-5.25	-5.05	mk/ppm B
EFFECTIVE FRACTION OF DELAYED NEUTRONS	5.664	5.646	5.589	5.337	0.001
SHUTOFF RODS REACTIVITY WORTH	-94.6	-94.3	-93.8	-88.9	mk

b) The reactivity coefficients, the kinetics parameters, and the shutdown rod effectiveness, for a representative core with 60 SEU show a very small, but in some cases unfavourable variation with respect to the natural uranium core.

Considering a) and b), the Safety Report Revision divides the postulated accidents in three categories:

- Events for which the fuel type modification will not alter the evolution of the plant. For these cases detailed studies were not done. It was shown that the behaviour of the plant would be similar to the natural uranium case. As an example this would apply to events such as loss of electricity, loss of main heat transport pumps and loss of cooling in the fuelling machine.

- Events in which the fuel type change has an effect on the evolution of the plant. Basically this includes the reactivity accidents, mainly the pressure assisted ejection of a control rod, and large loss of coolant events. These events were reanalized with detailed neutronic and thermohydraulic simulations.

- New postulated accidents that arise with the use of SEU fuel. In these category the erroneous introduction of a fresh SEU FA in one of the central channels was analyzed. The results of the studies show that although the operation limits for channel powers are exceeded the acceptance values for accident conditions of DNB critical or fuel rod centerline temperature are not exceeded. The possible risks of criticality when using SEU fuel were also analyzed, but according to the rules of the Argentine Regulatory Authorithy, with fuel enrichments below 1%, there are no risks of criticality with moderators not more effective than light water.

6 OPERATING EXPERIENCE

6.1 General Aspects

The irradiation of SEU fuel at the plant started in January 1995, with the introduction of six SEU FA in the six preselected channels described before in the period from the 9th to the 31st. During each introduction, relevant operating data like control rod positions, inlet heat transport system temperatures, outlet heat transport temperatures at the six channels, in-core detectors, etc, were collected. Between May 22nd to June 26th, the first six FA were moved to the central region of the reactor, and another set of 6 fresh SEU FA was inserted. Between October 30th and December 5th, a third group of 6 fresh SEU FA was introduced, and the first group of six taken out of the reactor with an average exit burnup of about 10 MWd/kgU.

The first SEU fuel assemblies remained in the channel they entered the core until they reached an average burnup of 5.5 to 6 MWd/kgU, and were later moved to the central region, were they remained until the average burnup reached ≈ 8.0 MWd/kgU. From there they were transferred to the central region until they reached 10 MWd/kgU.

The core reactivity gain when introducing fresh SEU fuel was about 0.7 mk (compared with 0.35 mk for natural fuel), while the channel power increase was about 15 - 20 % (compared with a very small change for natural fuel).

One particular concern about SEU fuel was the greater susceptibility to pellet cladding interaction (PCI) failures in power ramps, considering that it has a higher burnup than the natural uranium fuel.

6.2 Comparison between Calculations and Measurements of Relevant Parameters

The data was used for comparisons with neutronic calculations, similar to the ones used for design and fuel management using WIMS-D4 and PUMA. In particular, the relative increase in ΔT in the channels with the SEU fuelling was compared to the relative increase of the calculated channel power, and the increase in the vanadium detector readings close to the refuelled channels were compared with calculated values. The consistency of the calculated and measured reactivity change due to the SEU refuelling was done comparing the calculated core reactivity before and after the refuelling with the corresponding rod positions in each case, which should be the same.

The main results of the comparisons are shown in table 2. It can be seen that the agreement is good and in most of the cases within the uncertainty of the measurement errors.

6.3 Fuel Performance

The performance of the 18 SEU FA during the irradiation period was good, without any indication of failures. On the other hand the exit burnup of the natural uranium bundles showed an increase of about 2 %, due to the positive contribution to the core reactivity of the SEU FA, an effect that had been anticipated before.

After the 6 SEU FA that completed their cycle were taken out of the core, examinations and measurements were performed showing no abnormalities and that the elongation of the fuel rods was within the expectations considering the larger fuel burnup.

6.4 Effects of the SEU Fuel on the Plant Fuel Consumption

During 1993 and 1994 the average fuel consumption per full power day (fpd) was 1.307 FA/fpd. In 1995 the average fuel consumption was decreased to 1.215 FA/ fpd. In 1995, 18 SEU and 394 natural fuel assemblies were loaded, and 339 fpd were generated.

TABLE 2:COMPARISON BETWEEN CALCULATIONS AND MEASUREMENTS FOR THE FIRST TWO INTRODUCTIONS OF SEU FUEL IN ATUCHA-I

CASE	MEASUREMENTS		CALC.	UNIT
SEU FUELLING IN K27	AVGE	SIGMA		
RELATIVE INCREASE IN MEASURED CHANNEL DELTA-T	20.4	2.55		%
RELATIVE CHANGE IN CALCULATED CHANNEL POWER			20.8	%
RELATIVE CHANGE IN DETECTOR READINGS (ASSEMBLY 8)				
DETECTOR 1	6.66	1.15	4.96	%
DETECTOR 2	7.74	1.18	5.90	%
DETECTOR 3	7.95	1.23	6.72	%
DETECTOR.5	7.97	1.32	6.35	%
DETECTOR 6	8.42	1.34	6.11	%
DETECTOR 7	8.38	1.57	6.33	%
INCREASE OF THE AVERAGE OF THE ACTIVE DETECTORS	7.90		6.19	%
INCREASE OF THE INSERTION OF REGULATING RODS				
GRAY BANK	26.0	2.00		cm
BLACK BANK	4.0	1.60		cm
CORE REACT. CALCULATED BEFORE THE INTRODUCTION			-2.00	mk
SAME AFTER THE FUELLING KEEPING THE RODS POSITION			-1.27	mk
SAME AFTER THE FUELLING UPDATING THE RODS POSITION			-2.02	mk
REACT. DIF. BETWEEN AFTER AND BEFORE THE FUELLING			-0.02	mk
REACTIVITY CHANGE DUE TO THE FUELLING			0.73	mk
SEU FUELLING IN P12				
RELATIVE INCREASE IN MEASURED CHANNEL DELTA-T	20.94	3.21		%
RELATIVE CHANGE IN CALCULATED CHANNEL POWER			20.3	%
RELATIVE CHANGE IN DETECTOR READINGS (ASSEMBLY 6)				
DETECTOR 4	6.62	1.17	6.19	%
DETECTOR 5	6.25	1.18	5.57	%
DETECTOR 6	7.33	1.30	4.75	%
DETECTOR 7	7.43	1.30	5.55	%
INCREASE OF THE AVERAGE OF THE ACTIVE DETECTORS	6.86		5.52	%
INCREASE OF THE INSERTION OF REGULATING RODS				
GRAY BANK	28.7	2.0		cm
BLACK BANK	5.7	1.6		cm
CORE REACT. CALCULATED BEFORE THE INTRODUCTION			-2.07	mk
SAME AFTER THE FUELLING KEEPING THE RODS POSITION			-1.37	mk
SAME AFTER THE FUELLING UPDATING THE RODS POSITION			-2.23	mk
REACT. DIF. BETWEEN AFTER AND BEFORE THE FUELLING			-0.16	mk
REACTIVITY CHANGE DUE TO THE FUELLING			0.70	mk

Assuming that the design value of 0.70 SEU FA / fpd corresponds to the SEU fuel, these gives a value of 1.242 natural FA / fpd ($\approx$ 5 % improvement).

6.5 Prevention of PCI Failures

The existing PCI prevention criteria were developed for natural uranium fuel and was based on approximate estimations of maximum core linear power based on the readings of in-core detectors. For SEU fuel new criteria were developed for fuel of higher burnup based on the verification of the final linear powers and change in linear powers for all the fuel assemblies in the core using data files with burnups and linear powers from the fuel management calculations. These revised criteria was applied to power ramps arising during operation as a result of fuel movements, reactor power increases, and associated control rod movement and as a result of them the time to reach full power in a start up was increased from 28 to 35 hours. The overall experience with the new criteria seems good as no fuel failures were observed.

7 FUEL DESCRIPTION AND IMPROVEMENTS IN THE FUEL DESIGN

FA for Atucha-I consist of 36 fuel rods with natural UO_2 pellets and one supporting tube in the outer ring. Fuel rods are set in their positions by solid spacer grids. About 3 cm long bearing pads welded to the fuel sheath provide interaction with the spacers. Sliding shoes welded to the spacers or mounted to the supporting tube provide the interaction with the fuel channels and keep the fuel assembly in the position. Some sliding shoes are elastic.

Each fuel rod has a 5300 mm UO_2 pellet stack with isolated pellets at both ends. The fuel rod has also at the top of the pellet stack a gas plenum and a compression spring. Fuel rods are prepresurized to 18 bar.

Several changes have been introduced to the design of the fuel rod to use UO_2 pellet with SEU. The aim of these changes was to provide more space for gas release and to assure reliably interaction between spacers and bearing pads during the whole life of the fuel. Other changes, like modifications to the specification of the fuel sheath material, were directed to reduce the fuel rod susceptibility to PCI failures. A very thin coating of graphite over the internal surface of the fuel sheath will be introduced in the future with the same objective.

The original material for the sliding shoes (Zircaloy) was replaced by INCONEL 718 to compensate the higher relaxation produced by the increase of the fuel life into the reactor.

8 CONCLUSIONS

In January 1995, a program to irradiate SEU fuel in the Atucha 1 nuclear station was started. During 1995 three sets of 6 FA were introduced in the reactor. The first set was taken out after approximately 9 months with an average exit burnup close to 10 MWd/kgU, at the time the third set was introduced. The operating experience during this period was good, mainly reflected in the following aspects:

a) The operation of the plant showed no abnormalities attributable to the use of SEU fuel. The fuel consumption showed a decrease from 1.307 FA/fpd in 1994 to 1.215 FA/fpd in 1995, with the loading of 18 SEU and 394 natural FA.

b) The performance of the SEU fuel and the response to power ramps was good. PCI prevention criteria was revised to adapt it to SEU fuel with higher burnup.

c) The comparison of calculated channel power variations, reactivity increases and detector reading changes with the introduction of fresh SEU fuel with measured values showed good agreement.

d) Design improvements are being implemented to improve the fuel behaviour, particularly in power ramps.

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OPERATING EXPERIENCE OF THE FUGEN HWR IN JAPAN

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Abstract

Fugen is a 165 MWe Prototype heavy water reactor which mainly uses plutonium-uranium mixed oxide (MOX) fuel. Power Reactor and Nuclear Fuel Development Corporation (PNC) has taken responsibility for the Advanced Thermal Reactor (ATR) project, with its name "FUGEN" taken from the Buddhist God of Mercy. The project started in October 1967 for the purpose of development and establishment of the technology for this new type of reactor and clarification of MOX fuel performance in the reactor. Site construction began in December 1970 at Tsuruga and the plant commenced commercial operation on March 20, 1979. Since then, Fugen has been operated successfully for nearly 17 years. The plant performance and reliability of this type of reactor has been demonstrated through the operation. All these operational experiences have contributed to the establishment of the ATR technology. ATR is an unique reactor with outstanding flexibility regarding nuclear fuel utilization, because it has superior properties concerning the utilization of plutonium, recovered uranium and depleted uranium. Furthermore MOX fuel can be loaded in full core. In August 1995, Japan Atomic Energy Commission decided to cancel a construction program of demonstration-ATR plant which has been promoted so far, for the economical reasons regarding its construction cost. However, the future program of Fugen will be introduced for more advanced utilization of MOX fuels and other fields.

1. Introduction

Fugen is a 165MWe prototype Advanced Thermal Reactor (ATR) located in Tusruga, on the sea of Japan coast. In May 1966, the Japan Atomic Energy Commission (JAEC) established a national policy encouraging development of the heavy water moderated, boiling light water cooled, pressure-tube type power reactor, what we call ATR, and Fast Breeder Reactor (FBR). The reactor type for ATR was selected from several thermal reactor designs on the principles outlined below.

(1) ATR can use a variety of fuels, rather than only enriched uranium, including plutonium and depleted uranium.
(2) ATR has high neutron economy which can reduce the demand for natural uranium.
(3) ATR can effectively control plutonium build-up prior to the introduction of a commercial FBR.

ATR has been intended to use plutonium and recovered uranium extracted from the spent fuel of conventional light water reactors (LWRs), thereby helping Japan secure a stable supply of energy and at the same time contributing to the establishment of plutonium utilization technology. Research and development projects have been carried out mainly in the full-scale development facilities at PNC's O-arai Engineering Center in Ibaraki Prefecture [1]. The prototype ATR Fugen was designed and constructed based upon the results of this R&D activities and with the support of the Japanese government, electric utilities and research organizations [2].
Fugen started its commercial operation in 1979 [3] and it has been operated successfully for these 17 years. The excellent performance for a use of MOX fuel have been demonstrated and a great deal of valuable operation and maintenance experiences have been accumulated during this period.

2. Plant Description

The prototype ATR Fugen is similar to CANDU PHWR reactor except that its coolant is boiling light water and the core is vertically situated. A main specification of Fugen and a bird's-eye view of the main part of the reactor and the reactor coolant system are shown in Table I and Fig. 1 respectively. Four kinds of standard fuels of 28-rod fuel assemblies have been used in Fugen: MOX fuels of different fissile content and UO_2 fuels of different enrichment. Besides these standard fuels, four sets of special fuel assemblies of UO_2 which accommodate some specimens of pressure-tube material are also loaded. Furthermore, Fugen has been used as an irradiation bed of experimental fuels and materials. The fuels are loaded and unloaded by a remote-controlled refueling machine situated below the reactor core. The configuration of standard and special fuel are shown in Fig.2.

TABLE I Main specification of Fugen

Reactor type :		Pressure tube	
Heavy water moderated, boiling light water cooled, pressure-tube type		Material	: Zr-2.5%Nb alloy
		Inner diameter	: 117.8 mm
Gross output		Thickness	: 4.3 mm
Thermal output	: 557 MW	Calandria tube	
Electrical output	: 165 MW	Material	: Zircaloy-2
Core		Inner diameter	: 156.4 mm
Height	: 3.7 m	Thickness	: 1.9 mm
Diameter	: 4.05 m	Calandria tank	
Lattice pitch	: 240 mm	Material	: Stainless steel
Number of fuel channels	: 224	Inner diameter	: 8 m
Fuel		Height	: 5 m
Fissile material	: UO_2 1.5%, 1.9%	Control rod	
(^{235}U and Pu-fiss)	MOX 1.4%, 2.0%	Number of rods	: 49
Pellet diameter	: 14.4 mm	Material	: B_4C in stainless steel tube
Pellet height	: 18 mm	Reactor coolant system	
Fuel cladding		Coolant	: Light water
Material	: Zircaloy-2	Coolant temperature	: 284 ℃ (in steam drum)
Outer diameter	: 16.46 mm	Coolant pressure	: 6.7 MPa (in steam drum)
Thickness	: 0.8 mm	Coolant flow rate	: 7600 t/h
Moderator		Number of loops	: 2
Inventory	: 160 t(Initial charged)		
Temperature	: 70 ℃ (max.)		

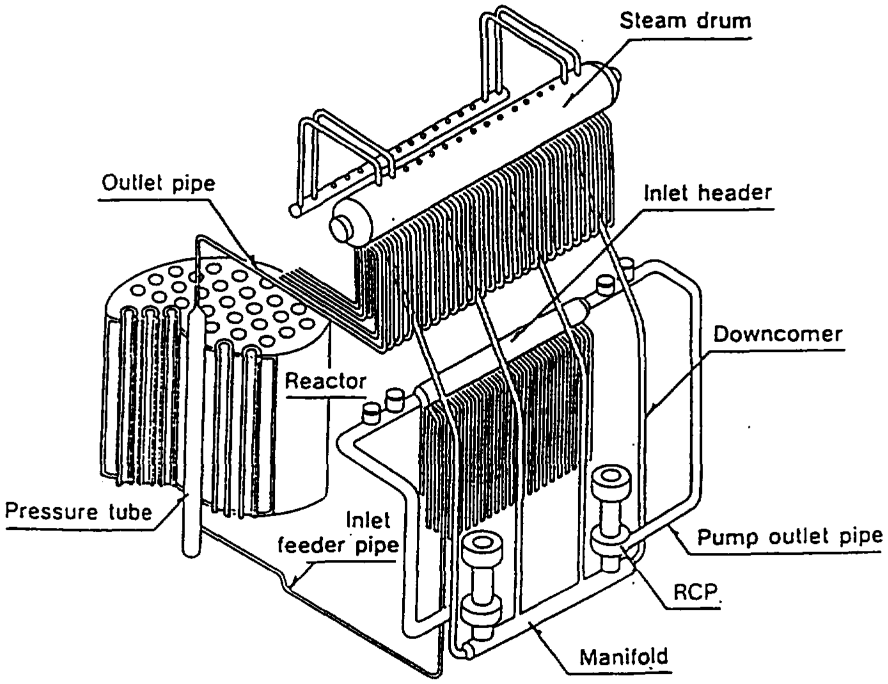


Figure 1 Reactor and reactor coolant system of Fugen

The core reactivity is controlled by moterdriven control rods and liquid poison (^{10}B) in the heavy water moderator. In usual operation, eight control rods are partially inserted in which the outer four rods are for manual fine control and the inner four rods are for automatic power control, and the other forty-one rods are all withdrawn from the core. The short-term reactivity change is controlled by the control rods and the long-term reactivity loss caused by fuel burnup is compensated by the adjustment of ^{10}B concentration in moderator. The core configuration is shown in Fig.3.

3. Overall performance of the plant

Fugen has been successfully operated since March 1979 when its commercial operation started [4][5][6]. The operating history is indicated in Fig.4. In 1980, stress corrosion cracking (SCC) was detected in some stainless steel pipes of reactor heat transfer system, and wide spreaded material replacement works and the other countermeasures for SCC were carried out mainly in 1980, 1981 and 1985. For this reason, the electrical load factor recorded in these years are lower than the others. However, the average electrical load factor throughout 17 years is about 65%. This fact justifies the overall reliability of Fugen is excellent. Because the Japanese government requires to perform a big inspection every operating year, which means the average electric load factor naturally cannot beyond 80% no matter how a perfect performance is gotten.

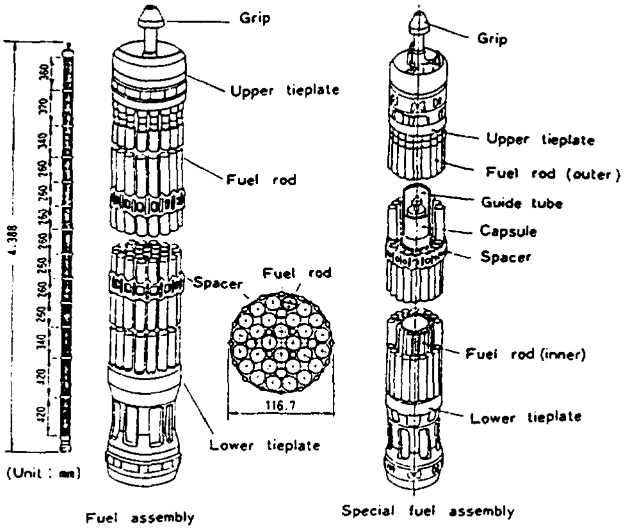
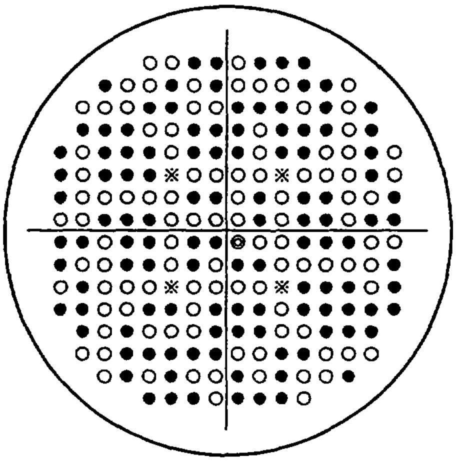


Figure 2 Schematic view of fuel assemblies



	TYPE	Number		TYPE	Number
○	UO2	103	※	Special	4
●	MOX	116	◎	Gd-1	1

Figure 3 Core configuration (24th cycle)

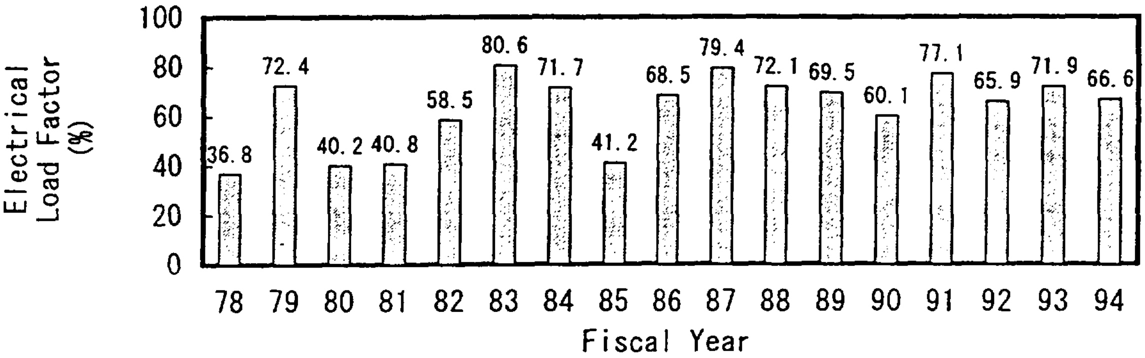


Figure 4 Operating history of Fugen

4. Core characteristics and experiences of MOX fuel utilization and development

4.1 Core characteristics

A lot of core performance data of Fugen such as core life time, reactivity coefficient have been accumulated and evaluated throughout the seventeen years' operation. The main characteristics of Fugen's core are as follows [7]:

(1) MOX fuel can be used in almost the same manner as UO_2 fuel with little effect of resonance absorption by Pu isotopes. Because neutrons are significantly slowed down in heavy water moderator.

(2) Flat power distribution can be attained without the insertion of control rods. Because the excessive reactivity is adjusted by poison concentration in the moderator and the migration area for thermal neutron is large in heavy water.

(3) Coolant void reactivity coefficient is nearly zero.

4.2 Experiences of MOX fuel utilization

Fugen has utilized MOX fuels along with UO_2 fuels since its initial loading [8] and was loaded with 620 assemblies of MOX fuel and 541 assemblies of UO_2 fuel to date. The history of MOX fuel utilization is illustrated in Fig.5. Fugen is originally designed to be able to load MOX fuels in full core, but in the actual result the maximum and minimum ratio of MOX fuel in the whole core in the individual core cycle are 72 % and 34 % respectively. Since the 4th cycle, type B fuel assemblies which have higher fissile content have been loaded to reduce the fuel cycle cost by attaining higher burnup. The maximum burnup of discharged fuel reaches 19.6GWd/t for MOX fuel and 19.9GWd/t for UO_2 fuel assembly [9][10].

The fuel-cycle flexibility of Fugen derived from the excellent neutron economy has been also demonstrated by means of loading several kinds of fuels such as:

(a) MOX fuels containing Pu recovered in the Tokai reprocessing plant (October 1981)
(b) UO_2 fuels containing U enriched in the Ningyo enrichment plant (January 1983)
(c) MOX fuels containing U recovered in Tokai reprocessing plant (June 1984)
(d) MOX fuels containing Pu recovered from Fugen's spent MOX fuels (June 1988)

These activities have contributed to demonstrate to close nuclear fuel re-cycle in Japan.

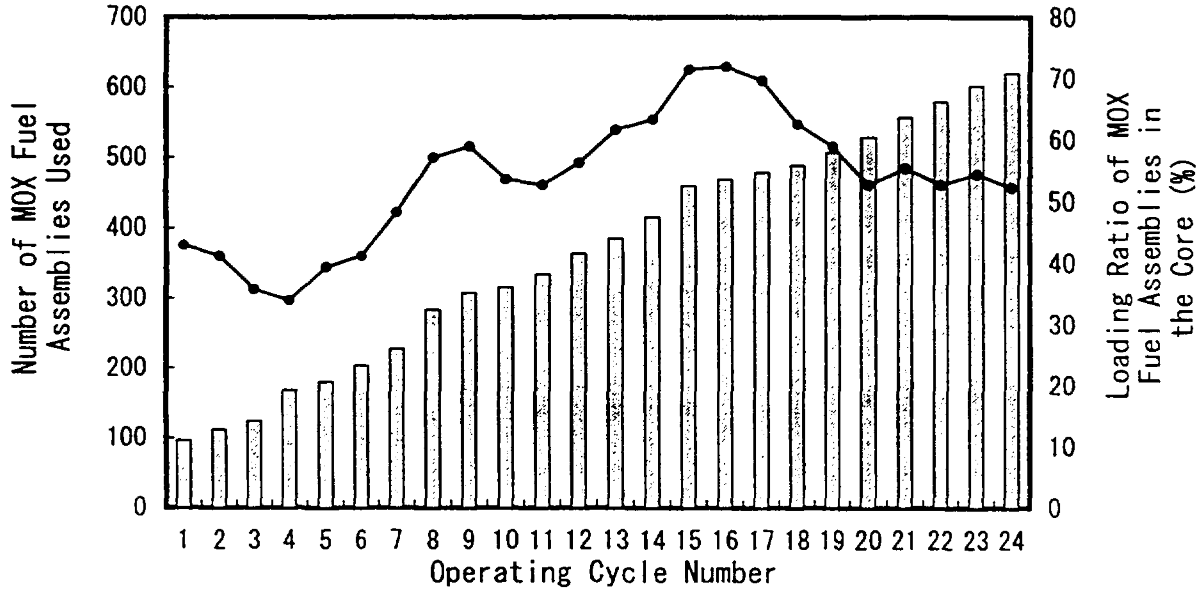


Figure 5 MOX fuel utilization in Fugen

4.3 Development of high performance MOX fuel

Some experimental fuel assemblies which are 36-rod bundle fuels have been irradiated in Fugen's core since 1984 to develop high performance MOX fuels; mainly focusing on high burnup. The design burnup of the present standard 28-rod fuel is 20 GWd/t as an average value of assembly, and two kinds of 36-rod fuels which aimed 35 GWd/t for standard fuel and 40 GWd/t for Gd added fuel have been irradiated. The former test irradiation and post irradiation examination were completed and the latter experimental fuel is still under irradiation in Fugen. Besides these high burnup fuel development activities, two sets of 36-rod fuel assemblies which contained short fuel rods using Zr-lined cladding and hollow MOX fuel pellets have been also irradiated to evaluate their resistance for Pellet-Cladding-Interaction. Some of the irradiated short rods (15-22 GWd/t) were used for power ramp test in Halden HWR in 1994-95 and no rod was broken during the power ramp tests up to about 68 kW/m (max.). Table II shows the specification of the fuel assemblies.

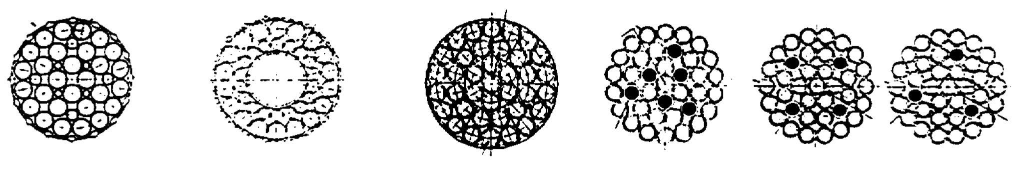
5. Inspection and evaluation of the integrity of pressure-tube

The pressure boundary components of the reactor cooling system of Fugen are inspected periodically. The pressure-tube assemblies are the most important components of all and they are subjected to in-service inspection (ISI) during the annual inspection to confirm the structural integrity. 224 sets of pressure-tube assemblies are arranged vertically and through the calandria tubes individually. Each pressure-tube assembly consists of a pressure-tube made of heat-treated zirconium-niobium alloy (Zr-2.5wt%Nb), and upper and lower extension tubes made of stainless steel. The pressure-tube is about 5 m long with 117.8 mm inner diameter and 4.3 mm wall thickness, and the full length of the assembly is about 9 m.

TABLE II Specification of fuel assemblies

Name:	Standard fuel		Special fuel		36-rod fuel	Segmented fuel	Gd. fuel	
Number of rods:	28		36		36	36	36	
	(4/8/16)		(18/18)		(6/12/18)	(6/12/18)	(6/12/18)	
Material:	MOX/UO2		UO_2		MOX	MOX	MOX + UO_2-Gd	
Pellet (Type)	(A)	(B)	(A)	(B)			(I)	(II)
Fiss. content:	1.4%	2.0% (MOX)	1.8%	2.3%	2.4%	3.0%	3.5%	3.2%
(^{235}U+Pu-fiss) :	1.5%	1.9% (UO_2)						
Diameter:	14.4 mm		8.29 / 12.78mm (inner/outer)		12.4 mm	12.4 mm	12.4 mm	
Cladding:								
Material:	Zircaloy-2		Zircaloy-2		Zircaloy-2	Zircaloy-2 and/or Zr-lined-Zircaloy-2	Zircaloy-2 and/or Zr-lined-Zircaloy-2	
Outer diameter:	16.46mm		9.70 / 14.72mm		14.5 mm	14.5 mm	14.5 mm	
Liner heat rate:	17.5 kW/ft		15.0 kW/ft		15.0 kW/ft	12.0 kW/ft	15.0 kW/ft	13.0 kW/ft
	(574 W/cm		492 W/cm		492 W/cm	394 W/cm	492 W/cm	427 W/cm)
Burnup:	20 GWd/t		30 GWd/t		35 GWd/t	30 GWd/t	40 GWd/t	30 GWd/t

Cross section:



5.1 Pressure-tube monitoring equipment

For the purpose of ISI for the pressure-tube assemblies, remote-controlled pressure-tube monitoring equipments (PTM-I(UT-ID), PTM-I(VT) and PTM-II) were developed [11] and have been used in Fugen since 1984. The design conditions are (a) to achieve fast and high precision inspection in a severe radiation environment (3×10^5 R/h-gamma rays) and in water (10℃-40℃), and (b) to minimize the personnel exposure by enabling remote-controlled handling. The inspection equipment is inserted inside the target pressure-tube assembly by using the refueling machine. The functions of the equipments are ultrasonic flaw detection, measurement of inside diameter and cross-sectional form for PTM-I(UT-ID), visual inspection of the internal surface for PTM-I(VT) and ultrasonic flaw detection for upper and lower rolled joint parts and axial length measurement of pressure-tube for PTM-II. Before the actual inspection, the total systems were tested on a mock-up pressure-tube assembly to confirm their functions and reliability. The specification and the test results are shown in TABLE III.

48 pressure-tubes were inspected by using PTM-I and 9 pressure-tubes by PTM-II so far. No defect has been detected and Fig.7 and Fig.8 show an example of the inside diameter measurement data. The inside diameter measurement is made over the whole region for the circumference from 0° to 360° and along the axial direction of the pressure-tube in upward spiral movement with the pitch of 3mm. Irradiation creep and growth strains are induced by fast neutron exposure, but the measured values using the equipment corresponded well with the design values.

TABLE III Specification and test results of the pressure-tube inspection equipment

Items		Specifications	Out-of-pile test results
Ultrasonic inspection device (UT)	- Flaw detection type:	Pulse-echo	
	- Frequency, angle of incidence:	10MHz, 26°	
	- Number of probes:	1 channel for longitudinal 1 channel for circumferential	
	- Detection performance:	5.0mm(L)×0.1mm(D)×0.1mm (W) with S/N >10dB	5.0mm(L)×0.05mm(D) ×0.06mm(W) with S/N=15dB
Inside diameter/form measurement device (ID)	- Measurement type:	Ultrasonic (15MHz)	
	- Measuring accuracy:	± 20 μm (10℃ - 40℃)	±15 μm (10℃ - 40℃)
	- Measuring range:	117.8mm - 120 mm	
Internal surface visual inspection device (VT)	- Type:	Black-and-white TV	
	- Differentiation performance:	2.0mm(L)×0.2mm(D) ×0.1mm (W)	5.0mm(L)×0.05mm(D) ×0.1mm (W)
	- Visual field	Height 30mm × width 40mm	
Elevation drive mechanism	- Type:	Rack-and-pinion	
	- Speed:	UT-ID: 30 - 300mm/min VT : 200 - 800mm/min	
	- Position detection accuracy:	± 3mm	± 1.3mm
Rotation drive mechanism	- Type:	Gear	
	- Speed:	UT-ID: 0.1 - 1 rev/s VT : 1 rev/min	
	- Angle detection accuracy:	± 2°	± 0.5°
Equipment operating condition	- Environment:	Reactor cooling water	
	- Temperature:	10 ℃ - 40 ℃	
	- Pressure:	Maximum 0.4 MPa	
	- Radiation:	Gamma rays 3×10^5 R/h	3×10^5 R/h - 80 hours

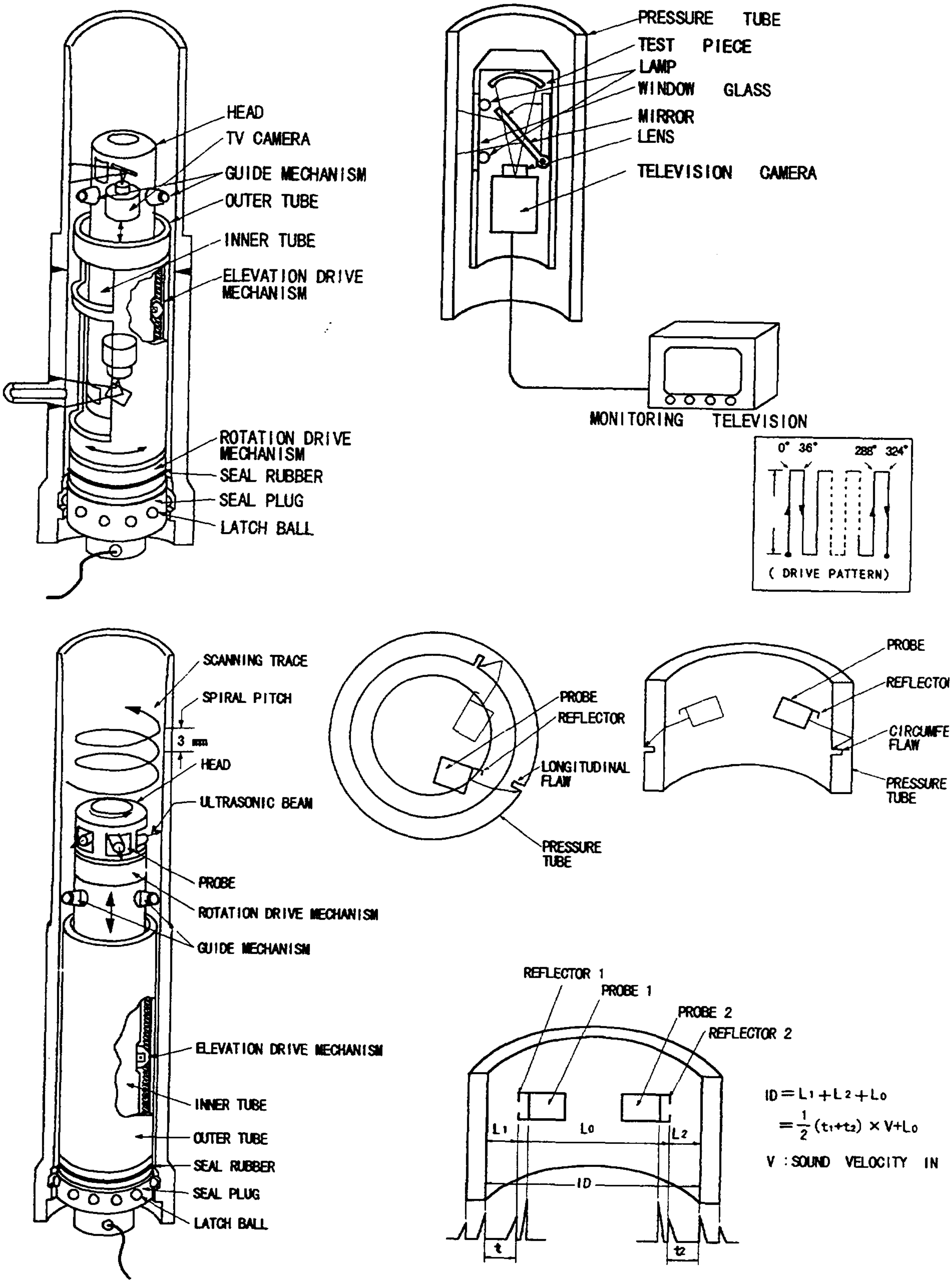


Figure 6 Schematic view of PTM-I

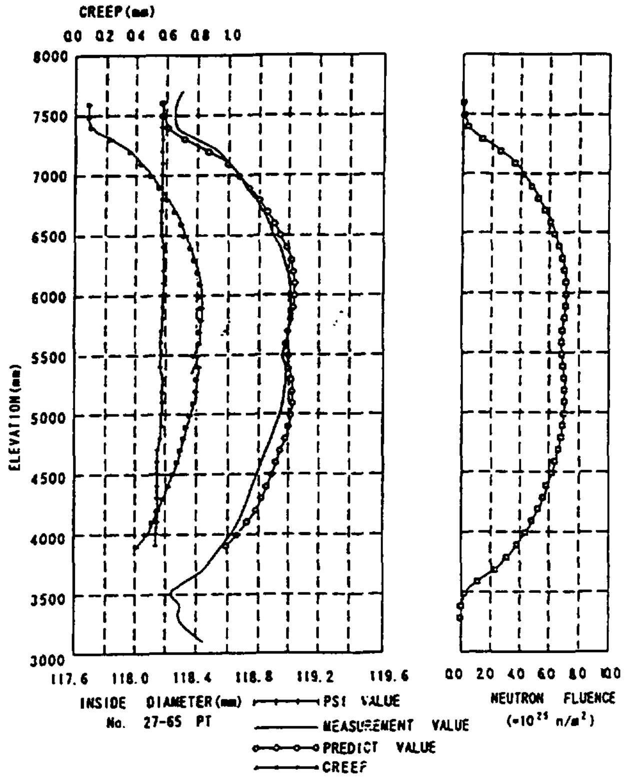


Figure 7 Results of inside diameter measurement

5.2 PIE of test specimens of pressure-tube material

Pressure-tube surveillance specimens were assembled inside the special fuel assemblies from the initial stage of the operation and were exposed to irradiation. The post irradiation examination (PIE) of the first and second surveillance specimens were performed. The PIE included tensile tests, bending tests, corrosion tests and hydrogen analyses, and resulted as follows:

(a) The ultimate tensile strength and 0.2% proof stress of irradiated materials showed values of about 30% increase during the initial period of irradiation in comparison with those of the unirradiated material, and became almost constant with the fluence more than 2×10^{25} n/m^2.

(b) The fracture toughness at 300 ℃ of small bending specimen showed value of about 20% decrease during the initial period of irradiation in comparison with that of the unirradiated material, and became almost constant with the fluence more than 2×10^{25} n/m^2.

(c) The corrosion is very small as 1 μm/year.

(d) The hydrogen pickup rate is very small as 1 ppm/year.

6. Radiation control

The dominant exposure source of Fugen is radioactive crud deposited on the inner surface of components and pipes. The major radioactive species in the crud are Co-60, Mn-54 and Fe-59. Because the internal tritium exposure is controlled less than 1 man mSv a year in Fugen by the measures such as follows:

(a) All components, valves, and piping of the heavy water and helium system are designed from the point of view of protection against tritium leakage. Canned-rotate pumps, below-seal valves, and welded tube connection are applied to the systems.

(b) Tritium diffusion with heavy water leakage is restricted by isolated room structure which has a special ventilation system for tritium removal.

(c) Early detection of leakage is performed with high sensitivity tritium monitors developed in PNC.

Ninety per cent of annual station dose is caused by inspections, maintenance and repair works which are carried out in annual inspection outages. Several kinds of measures including application of automated ISI machines, chemical decontamination and water chemistry methods have been applied to reduce the station dose rate. The dose rate of the surface of the components and pipes has been increasing with the reactor operation as shown in Fig.10, however the station doses for routine inspection are kept in constant range of 2-4 man-Sv an inspection period in spite of buildup of radioactivity.

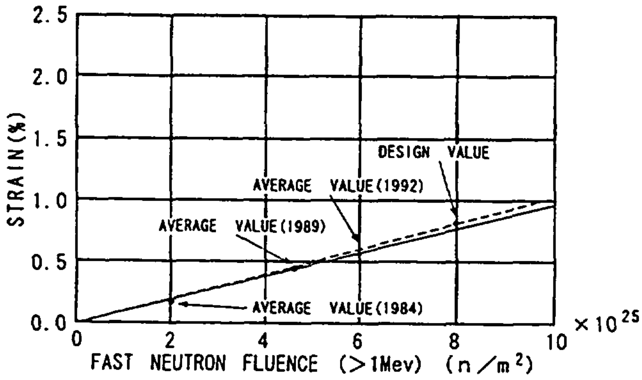


Figure 8 Results of diametrical irradiation creep and growth strain

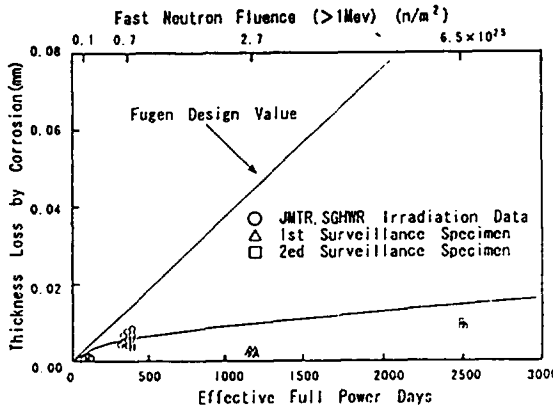


Figure 9 Corrosion test results of pressure-tube specimen

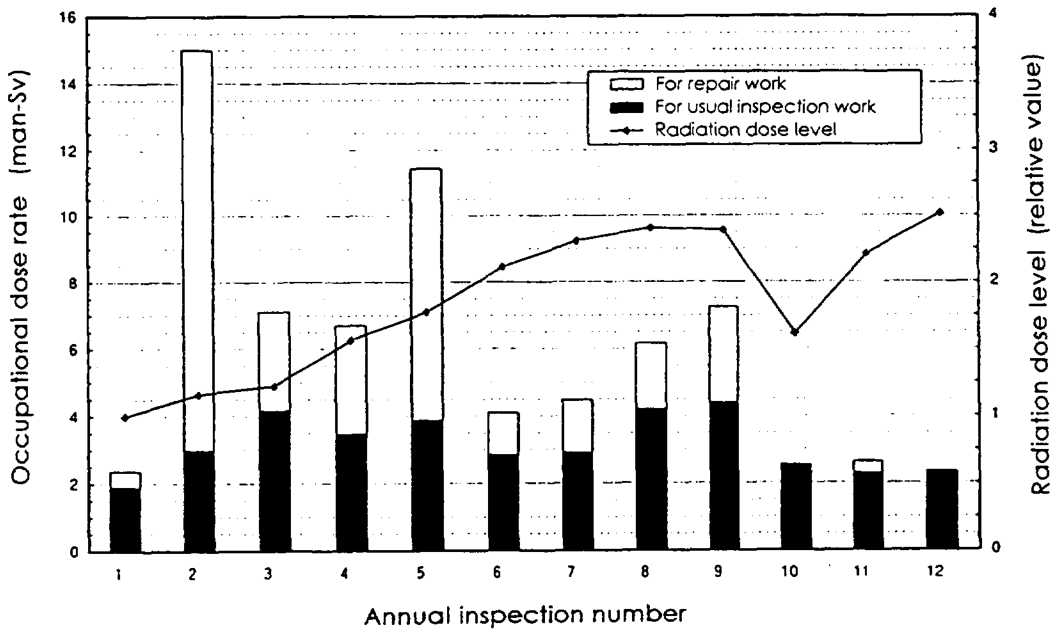


Figure 10 Radiation dose level (outlet of recirculation pump) and station dose history

The most effective measure for dose reduction was the system chemical decontamination which was performed in 1989 (for A-loop of coolant circuit) and 1991 (for B-loop) when large material replacement works for SCC were performed. A series of R&D activities have been carried out since 1977 and has confirmed material integrity during and after the decontamination and characterized the dilute chelate decontamination reagent [12]. All fuel assemblies of each loop of the coolant system were discharged before the decontamination work to avoid a large amount of activity dissolution from the fuel surface. After the decontamination, the dose rate on the surface of the reactor coolant system equipments and pipes were reduced to half to one tenth as shown in Fig.11. The average decontamination factors of 3.4 and 5.1 were obtained and the occupational dose of 6.6 and 7.8 man-Sv were saved by the two times of decontamination respectively [13][14].

7. Water chemistry

The light water coolant in Fugen is kept neutral with no chemical additives as in BWRs'. Specification of the coolant is shown in TABLE IV, and the measured values indicate good chemistry. One of philosophies of the controlling the coolant chemistry is to suppress iron transport to the core as low as possible, since the reduction of iron input is effective to reduce surface dose rates of the coolant circuits. From this point of view, oxygen injection into the feed water and operational improvement of the condensate demineralizers were employed [15][16].

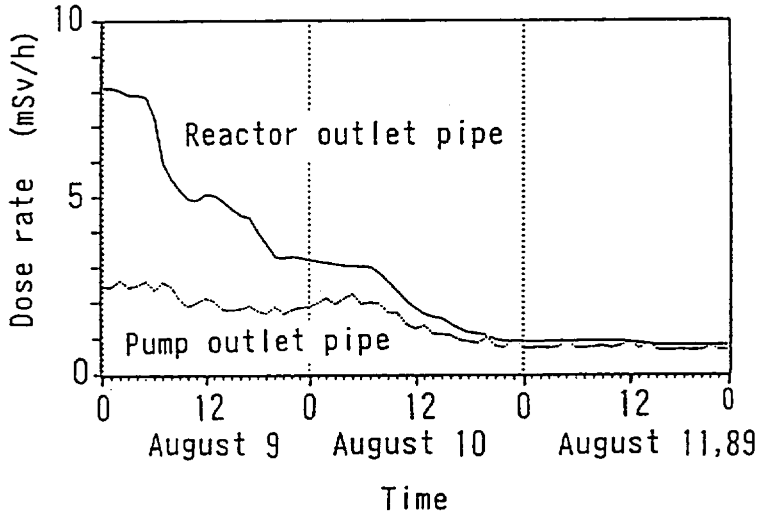


Figure 11 The surface dose rate change during decontamination

TABLE IV Chemistry parameters of light-water-coolant and heavy-water-moderator

	Light-water-coolant		Heavy-water-moderator	
	Specification	Measured value	Specification	Measured value
PH:	5.5 - 8.5	7.0 - 7.8	4.5 - 8.5	6.2 - 7.6
Conductivity:	< 1.0 μS/cm	0.10 - 0.29	< 5.0 μS/cm	0.27 - 1.2
Cl^-:	< 0.2 ppm	< 0.001 ppm	< 1.0 ppm	0.001 - 0.014 ppm
SiO_2:	< 2.0 ppm	0.01 - 0.03ppm	< 1.0 ppm	0.006 - 0.034 ppm
BO_3:	< 2.0 ppm	< 0.005 ppm		
Dissolved oxygen:	< 0.4 ppm	0.012 - 0.032ppm		
Suspended solid:			< 0.5 ppm	< 0.1 ppm

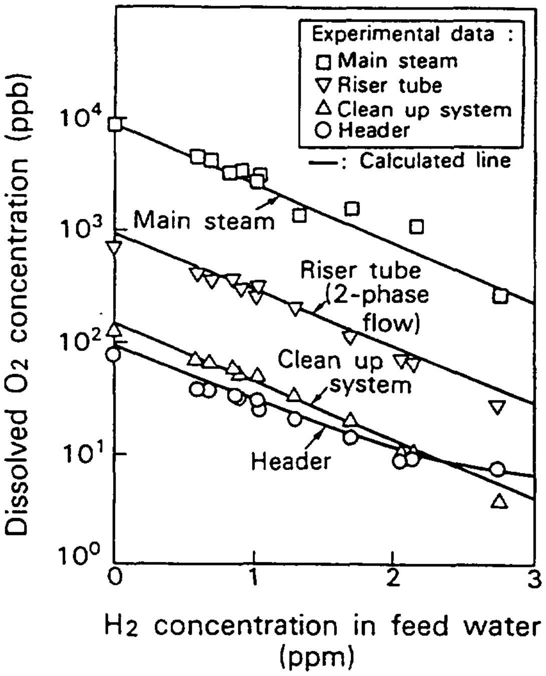


Figure 12 Oxygen concentration change with hydrogen injection

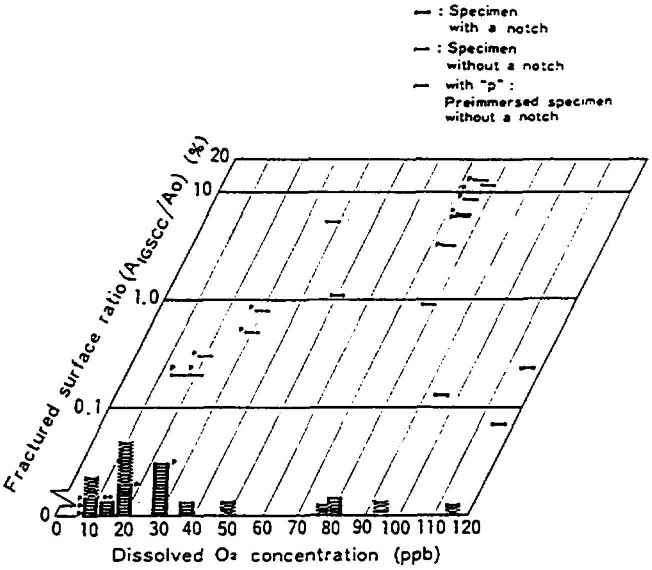


Figure 13 SSRT test results

Heavy water moderator contains no chemical additives except boron. Purification of heavy water is achieved with the resin beds in order to prevent the material of the system from corroding and minimize the accumulation of deuterium in the helium used as blanket gas of the moderator. The typical chemistry parameters are also listed in TABLE IV. In early days of the start-up test in 1978, unexpected deterioration was found in weak basic ion exchange resin beds used in the heavy water purification system, which was caused by deuterium peroxide, and this resulted in the increase of the conductivity and the radiation dose rate in the heavy water system. Studies to improve the quality of the heavy water, such as nitrogen gas reduction from helium covering gas, cooling of the resin beds and improvement of the resin, have been carried out. The conductivity reduced by half and the increasing trend of radiation dose rate was interrupted consequently.

Hydrogen Water Chemistry (HWC) was introduced to Fugen in 1984 and has acquired satisfactory achievements for the protection of type 304 stainless steel against SCC for more than ten years [17][18]. The main part of the reactor coolant piping of Fugen consists of type 304 stainless steels including weld joints which are relatively susceptible to SCC. After the basic study, hydrogen was injected to the upstream the feed water pump. Dissolved oxygen (DO) in the coolant decreases significantly with hydrogen injection as shown in Fig.14. The desired hydrogen injection condition was derived from such test results. In parallel with the above mentioned tests, the studies concerning the mitigation effect of HWC on SCC susceptibility have been carried out. Uniaxial constant load tests, slow strain rate tensile tests and the electrochemical potential measurements were conducted in the in-plant autoclave and the susceptibility of SCC was evaluated as shown in Fig.13.

8. Heavy water management

Heavy water purity is maintained at 99.7wt% in the moderator system of Fugen. Deuteration and dedeuteration of resin beds and maintenance work generate degraded heavy water. The degraded heavy water is enriched to over 99.80wt% with two types of heavy water upgraders and returned to the system again. Upgrader (I) is an electrolysis batch process plant, and upgrader (II) is a continuous processing plant using isotopic exchange reaction between water and hydrogen.

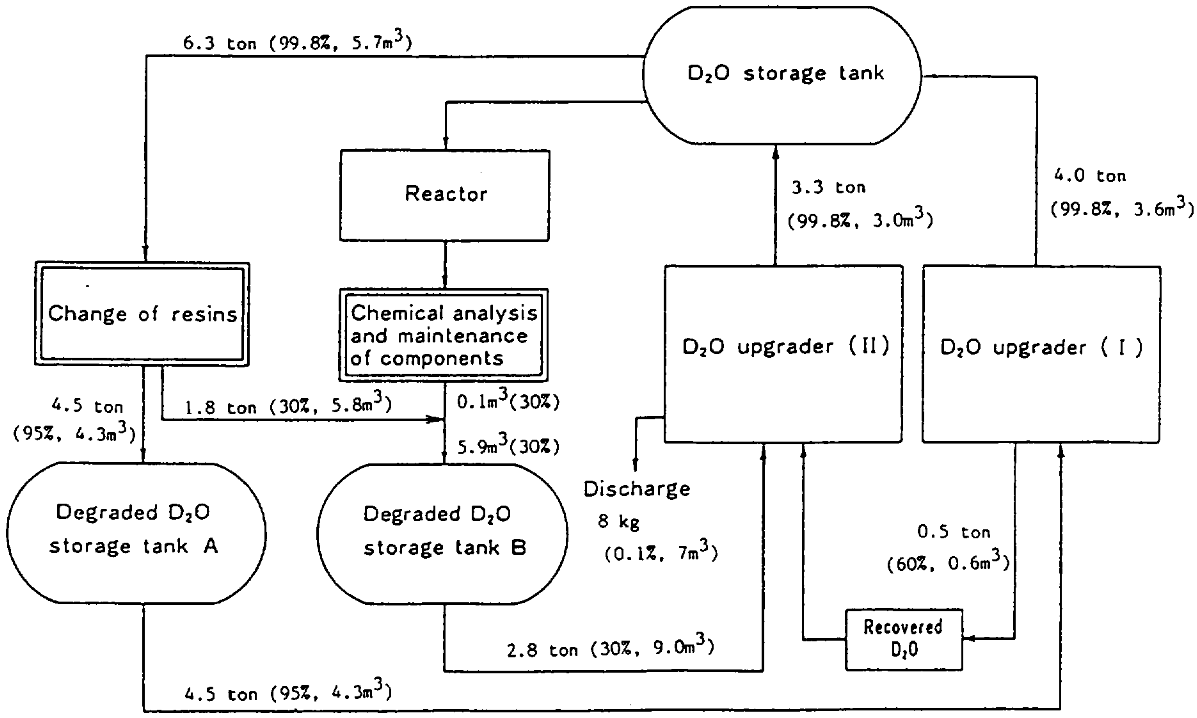


Figure 14 Heavy water recycling system in Fugen

The upgrader (I) treats 5 t/y of degraded heavy water of 95 wt% D_2O content, and produces 4.4 t/y of nuclear grade heavy water. It started in operation in 1979. The upgrader (II) processes 10 t/y of degraded heavy water of which D_2O content is lower than 95 wt%. It started in operation in 1986, and had produced approximately 36.7 tons of nuclear grade heavy water up to the end of 1995. Fig.14 shows the heavy water recycling system in Fugen.

9. Future programs

In August of 1995, JAEC decided to discontinue the ATR demonstration plant development program. Accordingly, the future program of Fugen has been reviewed and direction of the new mission of Fugen has been set forth. The detailed program is just under discussion, but the following is the outline of the possible programs.

Fugen has played a pioneering role in demonstrating the nuclear fuel recycling technology including MOX fuel utilization as a thermal reactor as shown previously. The most important part of the new mission of Fugen is to drive forward the further development in this area. The development program of more advanced MOX fuels of 54-rod type with higher burnup of approximately 60 GWd/t and MOX fuels containing gadolinia is in preparation. Another subject is concerned with the utilization of MOX fuels containing increased content of higher plutonium isotopes and the minor actinide elements.

Noticeable technology relating to the operation and maintenance of nuclear power plants has been developed by using Fugen [19]. The further programs are being planned to develop and demonstrate more advanced technology concerning plant operation support system, prediction and diagnosis system for abnormal events, operation support system for emergency events, maintenance support system, and occupational dose reduction technique by zinc injection into feed water circuit.

Fugen has contributed to the international cooperation to improve the safety of nuclear plants in the world. For an example, the technique of coolant leakage detection by an acoustic method is going to be applied to a RBMK type nuclear power plant in Russia and a number of researchers and engineers in Asian countries joined the work in Fugen. It is expected that the achievements of the future technology development programs described here will contribute to the enhancement of the safety and reliability of nuclear power plants in the world and the participation in the programs from other countries will be considered.

Fugen is a pressure-tube type reactor, which has a unique capability of utilizing some parts of pressure-tube assemblies as test beds under irradiation and water environment of fuels and materials. The possible idea is to use a pressure-tube assembly for studying water chemistry and developing on-line monitoring technique of material behaviors in a controlled water condition under irradiation. Other emptied pressure-tube assemblies will be used as test bed for the development of advanced materials for nuclear uses as well as non-nuclear uses.

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FAULT DIAGNOSIS THROUGH WEAR PARTICLE ANALYSIS (FERROGRAPHY) IN INDIAN PHWRs AND BWRs

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Abstract

Ferrography is a diagnostic predictive maintenance tool that is available to alert problems early in rotatory and reciprocating machinery. The particle of abnormal and accelerated wear generated by each of the mechanism can easily be recognised by their shape, size and texture appearance. Thus, wear particle analysis can avoid failures and can also pinpoint the root cause of problem. It is capable of indicating the rate of wear, severity of wear, type of wear and location of wear for the equipment well in advance.

This presentation will provide an overall view on the application of ferrography for identifying primary modes of failure in critical equipments of Indian Nuclear Power Plants. The trending of particle quantification (PQ) index has been found suitable for trend analysis of the wear process. The normal, alarming and severe levels of wear in terms of PQ index is being established. This aim have been realized on the strength of extensive experimental work on oil samples drawn periodically/regularly from machine of NAPS, KAPS, MAPS, and TAPS.

Trend and results of regular lube oil sample of all Indian Nuclear Power Plants are encouraging.

1.0 INTRODUCTION

1.1 The Origins of Wear Debris

When the lubricant film breaks down, seizure occurs and scuffing takes place with consequential increase in surface damage and generation of wear products.

In a free flowing (typically oil) and well lubricated system, this debris is carried by the lubricant to other parts of the system. This affords opportunity to monitor the wear condition through direct, "on-line" measurement of debris quantity. The main objective of this technique is to establish characteristic wear patterns of all plant equipments. This technique gives indication of failure well in advance as shown in Fig.1.

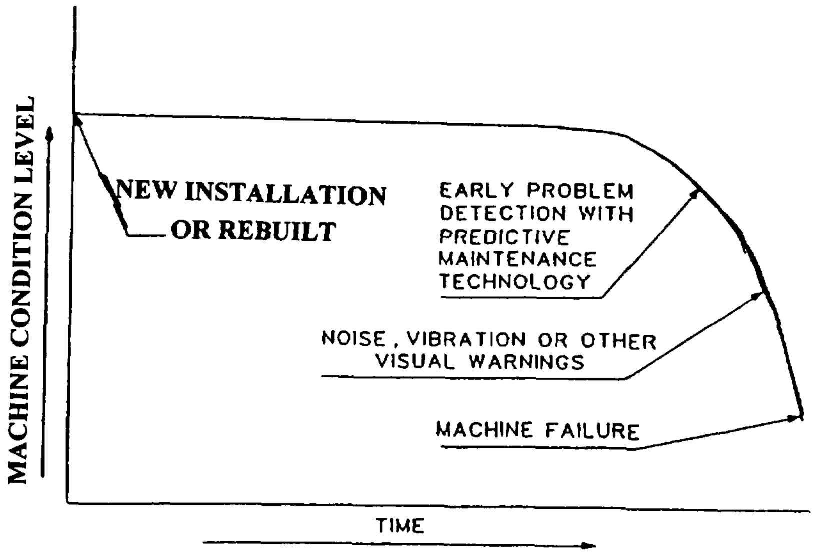


Fig.1 Condition monitoring

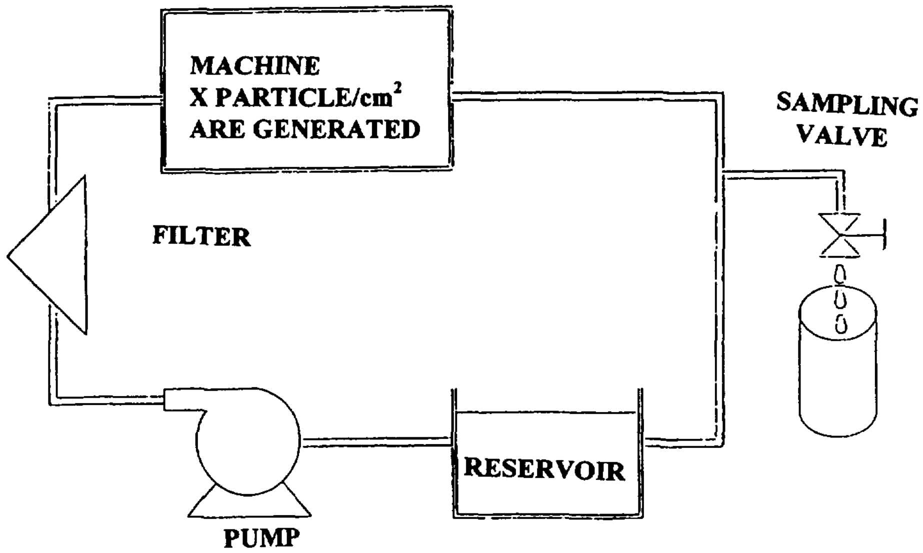


Fig. 2 Typical lubrication schematic

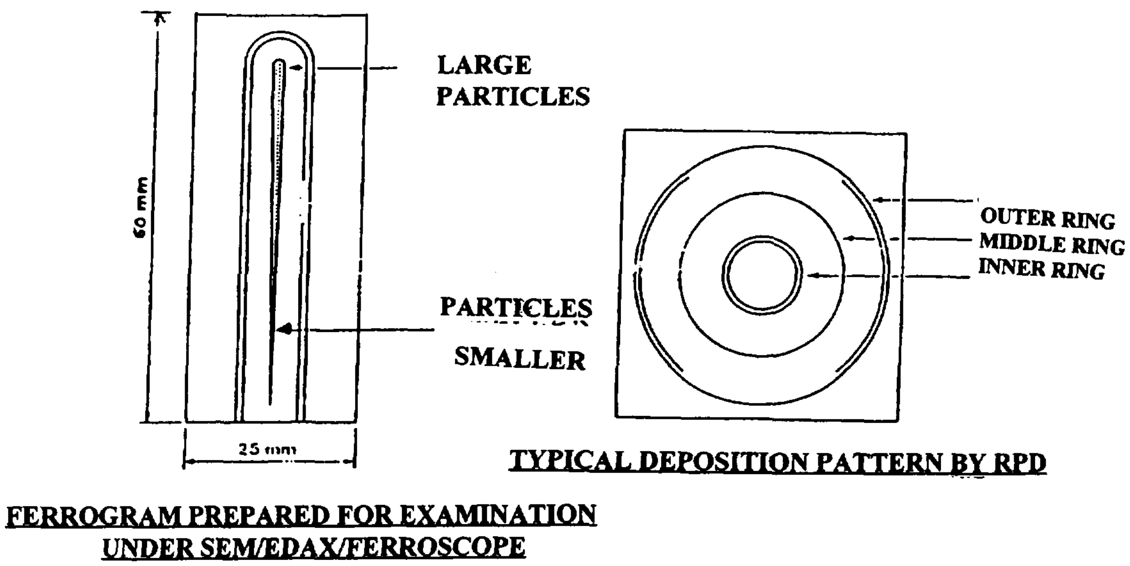


Fig. 3 A ferrogram and a typical deposition pattern

2.0 EXPERIMENTAL PROCEDURE

The size, shape, quantity, colour and composition of the particles can revel the mode of the wear in a machine. Increase in sizes, quantity or both of particles well be there when severe wear occurs. It can be applied to all oil, grease and hydraulic lubricated systems.

2.1 Sampling Point

The lubricant in critical equipments in PHWR and BWR are not easily accessible for sampling. Initially the samples are being collected from either existing drain-point or main oil tank. But sample should always be taken before in-line filter and sample should be taken from a single location in the system. A true representative oil sample is an absolute must for proper analysis and diagonosis. This is shown in Fig.2.

2.2 Sampling frequency

Experiment in other plants has shown that taking a monthly sample ensures that the onset of abnormal wear is detected in time to minimise the consequences. In Indian Nuclear Power Plants samples are collected in every three months from the equipments to closely monitor the machine health and the rate of wear. Monthly sampling will be preferred once in-house setup of ferrography instrument is ready. Action has already been started in this regard.

3.0 INSTRUMENTS, MEASUREMENT AND ANALYSIS

3.1 Instruments

The instruments being used for wear particle analysis are :

i) Particle quantifier - It measures ferromagnetic debris in lubricating oils, greases, etc. It has excellent sensitivity for particle sizes from, 1 to > 2000 microns and is used to monitor trends as in indicator of wear condition knwon as a "PQ Index".

ii) Automatic Particle Counter (APC) - An oil smaple is passed through a magnetic field in which photo sensors measure the change in transmitted light, yielding a digital read out proportional to the amount of refraction. The resultant number is an indication of the quantity of metal particle in the oil and it can be trended.

iii) Rotary Particle Depositor (RPD) and Standard Linear Ferrogram - The RPD instrument is used for preparing a slide for ferrogram for subsequent examination by optical miscroscope of ferroscope.

Non-ferromagnetic materials like Aluminium, Bronze, etc. which have rubbed against ferromagnetic material, and thereby picked up some of the particles from such materials, would also get deposited on the slide which is shown in Fig.3.

iv) Ferroscope System and Ferrography Analytical Software Technique (FAST) - The ferrography techniques is used by two different methods and instruments. In first method the Direct Reading Ferrography (DRF) then separates the debris into small and large particle and gives quantitative measurement by photometric method.

In second method the Analytical Ferrography (AF) which separates the wear debris from used oil by magnetic methods producing a glass slide for subsequent examination by microscope.

3.2 Trending of Results

Presently, samples from our nuclear power plant is being trended for PQ index and few samples were also examined by Automatic particle counter for measuring the range of particles and if PQ index is on higher side only then detailed ferrography examination is being carried out.

3.3 Selection of equipments :

The selection of main equipments and systems from PHWR and BWR for the purpose of wear particle analysis are done based on the criticality of equipments. Among all, turbo generators have the cleanest lubrication arrangements.

4.0 PQ Index

The severity of machine condition is judged by observing the trend of PQ values in successive samples and not on the basis of single reading. To be significant, the PQ index must change by at least 10% relative in successive sample. PQ values are affected by oil change, topping or any maintenance activity in the machine. Even for a given machine, the PQ value for normal wear may get changed as the machine becomes aged or worn out. Secondly, the PQ permits the adoption of condition monitoring techniques at a fraction of the cost associated with spectrographic and actual ferrographic methods. PQ indexing instrument is shown in Fig.4.

5.0 Wear Particle Equilibrium :

A running machine continuously generates the wear particles, but the life of particles circulating around the system is quite short. The particle concentration for normally operating equipment reaches a dynamic equilibrium after the initial break in period is over.

The behaviour of large particle and small particle is slightly different as regard to forming equilibrium. The rate of increase in concentration of small particles is quite small during normal wear mode whereas it increases rapidly once the abnormal wear mode begins. If the oil is changed, the initial concentration of small particles will be zero but will not return to its original configuration. Instead it will follow the rate of increase previously established. Fig.5 shows the behaviour of small particles.

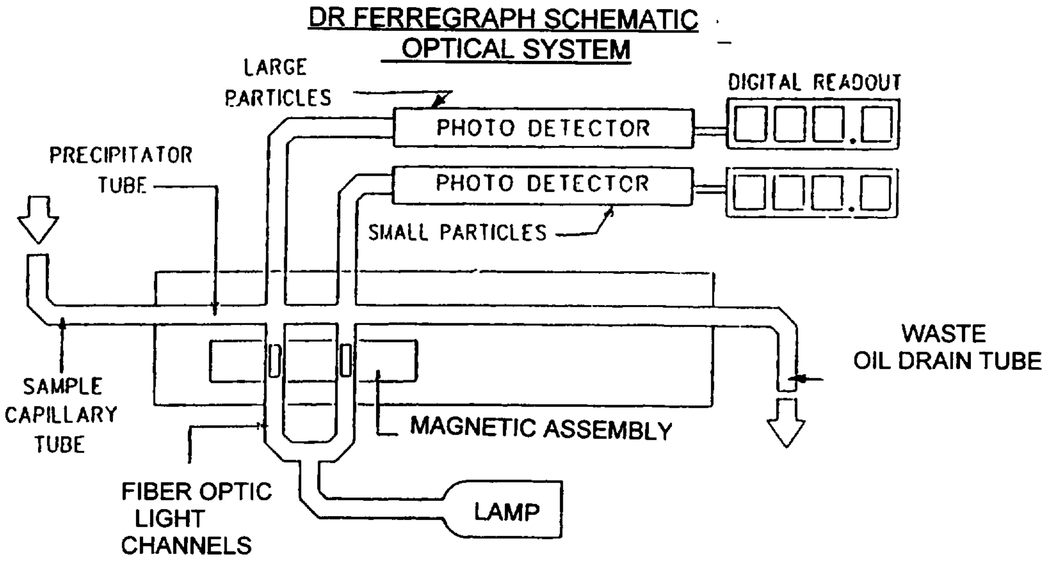


Fig. 4 Particle quantifier: Direct read-out of PQ index

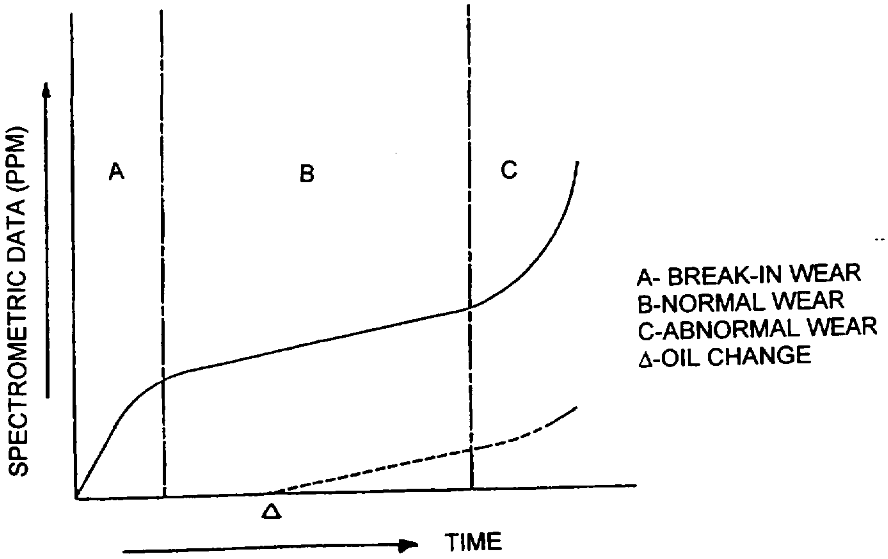


Fig. 5 Behaviour of small particle concentration

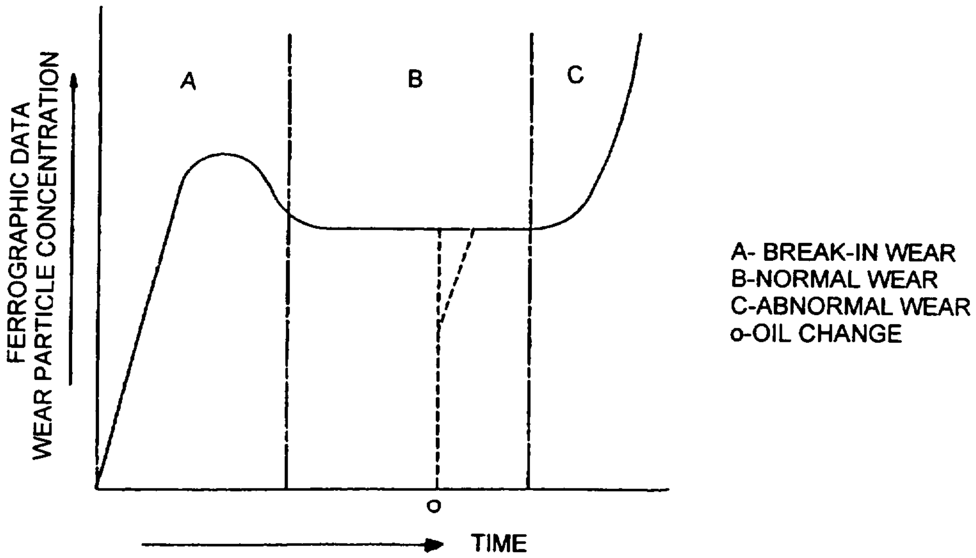


Fig. 6 Behaviour of large particle concentration

Fig.6 shows the behaviour of large particle concentration during the life time of a typical machine. The recovery time in case of large particle is quite short and it is a function of various parameters such as particle size, filtration efficiency, oil volume, oil circulation rate etc.

6.0 **Trending and Measurement of Severity of Wear Index :**

DL = Desnsity of large wear particles i.e. > 5 um.

DL = Density of small particles i.e. < 5 um.

6.1 The ratio DL/DS is a measure of wear condition. A rapid increase in DL/DS shows abnormal wear. This is shown in Figure.7.

6.2 Total wear Q = DL+DS and 'Severity terms' S=DL-DS.

Sharp rise in DL-DS indicates increased generation of large particles, leading to sudden increase in total wear thus imminence of catastrophic failure is indicated.

6.3 Mean = The average of Q values determined to be normal wear.

6.4 Severity of wear index Q x S.

$= (DL+DS) \times (DL-DS) = DL^2 - DS^2$

6.5 Percentage of large particles (PLP)

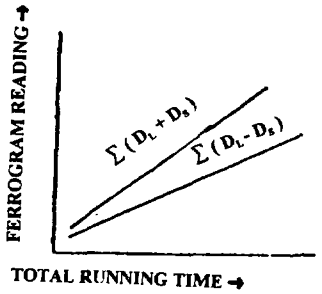
$= \frac{(DL-DS)}{(DL+DS)} \times 100$

An increase in both the wear particle concentration and percentage of large particles is an indication of abnormal wear condition. This is shown in Figure.8.

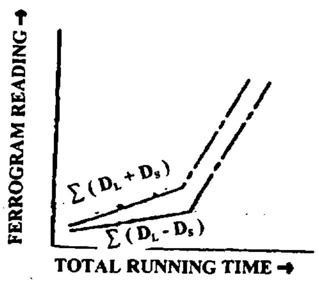
7.0 **Sampling, Result and Discussions :**

Details of samples collected from different sites of nuclear power plant in one year are given in table :

Sr. No.	Sampling Detail	NAPS Date	NAPS Total Sample	KAPS Date	KAPS Total Sample	MAPS Date	MAPS Total Sample	TAPS Date	TAPS Total Sample
1	First Sampling	Dec.94	23	Feb.95	48	Apr.95	35	Apr.95	44
2	Second Sampling	Mar.95	32	May.95	61	Dec.95	40		
3	Third Sampling	Jun.95	37	Aug.95	63				
4	Fourth Sampling	Nov.95	40	Dec.95	50				



(1) Satisfactory situation, gradual increase in wear and severity of wear



(2) Catastrophic situation developing

Fig. 7 trend analysis by ferrography

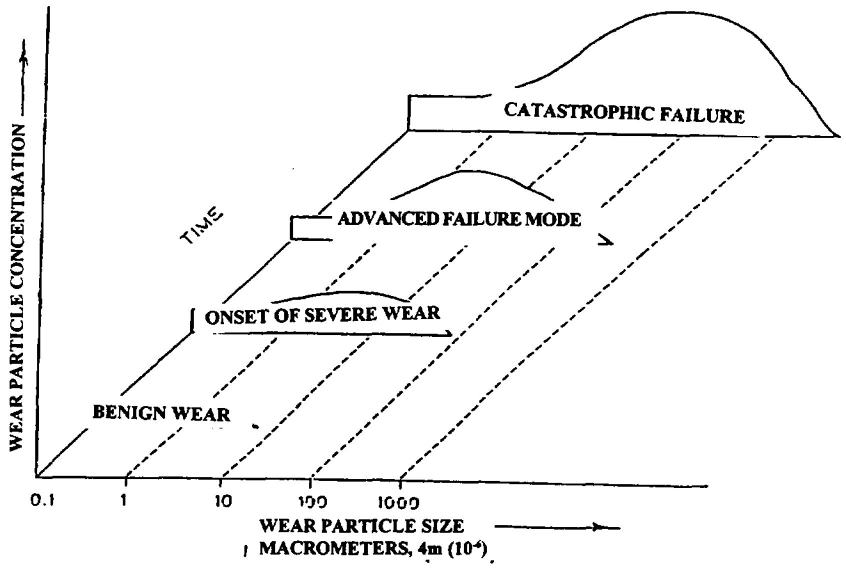


Fig. 8 Wear progress mode

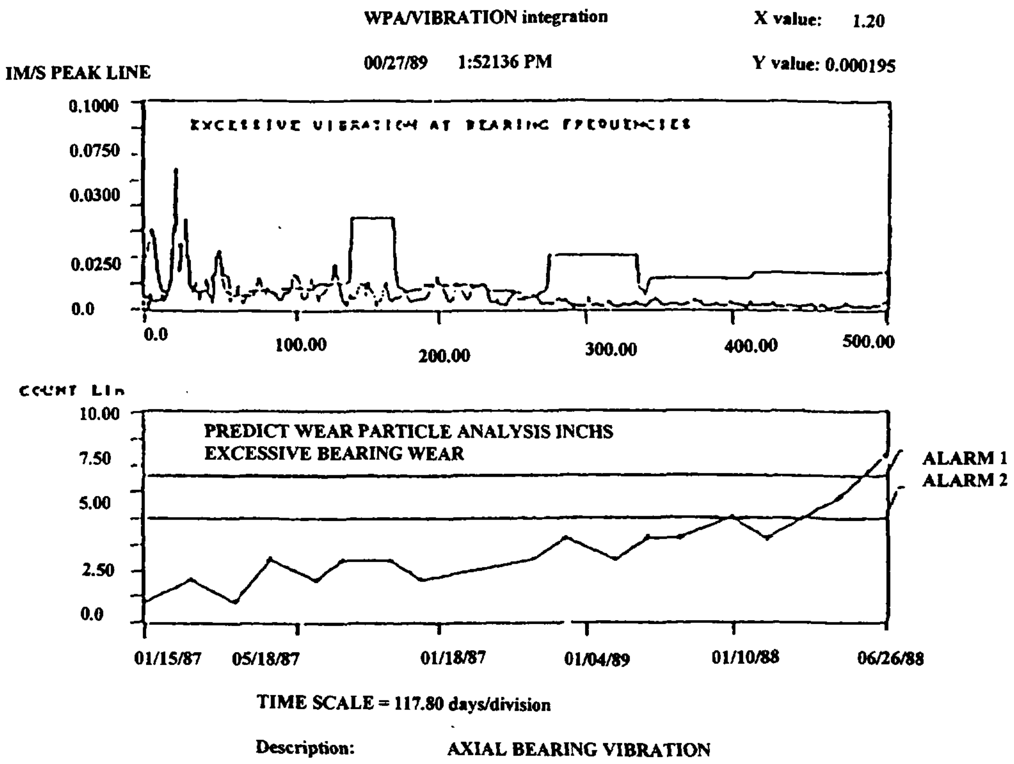


Fig. 9 Bearing failure indicated on both vibration spectrum and wear particle analysis trend plots on same screen

All samples were analysed for Particle Quantification (PQ) Index. Few samples were also analysed for particle size by Automatic Particle Counter and some samples were analysed for detailed ferrography.

Results of above samples have been analysed and based on the data from particle quantification test it is observed that wear condition is normal in most of the samples of NAPS, KAPS, MAPS, and TAPS.

PQ index values are found low for all the samples for all three batches of NAPS and maximum value observed in 39. Particle counting by Automatic Particle Counter (APC) over a range of 0 - 150 microns showed that maxium number of particles found in the range of 0 - 20 microns in most of the samples. Absence of larger wear particles further confirm that there is no abnormal wear conditions. It was then decided not to analyse the sample for Automatic Particle Counter.

However, samples from all six DG sets of MAPS and KAPS were found opaque and dirty on visual inspection in first batch sampling but PQ index found to be low in these samples. However, following chemical test (i) viscosity and viscosity index (ii) Neutralisation value (iii) Pour point (iv) emulsification characteristics (v) rust prevention characteristics (vi) copper strip corrosion test and (vii) foaming characteristics were carried out for above samples and found well within the limits. These oil are also replaced after operation of 4000 hrs.

Detailed ferrography for 8 samples of KAPS and 4 samples of TAPS have been carried out due to their high P.Q. Value. Therefore, to establish the characteristics wear pattern in these machines and to monitor the progressive wear damage, it is necessary to have few more samples from these machines and sampling frequency of three months is justified.

However, two FWTR, one NAHP and one NALP of KAPS, two compressors of KAPS, one condensate extraction pump of MAPS are considered to be critical due to high PQ value as indicated in the Annexure. But immediate repititive sampling from these equipment showed normal condition.

The PQ index of turbo generator lubricating oil shows that the lube oil is more or less clean and generally meet the NAS cleanliness standard i.e. NAS class 7. No definite trend has been observed in the behaviour of PQ index with respect to time for other equipments so far.

8.0 RECOMMENDATIONS

It is recommended to modify the sampling points of few equipments by fitting a half inch gate valve at the location where oil is in agitation and also contained most of the recent wear particles. Samples from the new locations will provde consistent and repeatable results. Sampling point should also be provided in all equipments being commissioned at NPC's power projects. Also adequate precaution to be taken in providing proper sampling points in advance PHWRS from design stage itself.

9.0 CONCLUSION :

The result has been presented to show that when Wear Particle Analysis (WPA) indicates machine is in good operating condition, periodic maintenance may not be necessary. Therefore, based on the WPA results, maintenance may be deferred or the periodic frequency may permanently be adjusted. During last fifteen months, no sampled equipment has failed so far in any of the NPC's plant. However, two pumps failed at NAPS as lube oil sample could not be collected due to non-availability of the sampling points. Immediately, sampling points were provided in both the pumps for sampling.

10.0 WEAR MECHANISM

This analysis also indicates exact wear mechanism from the presence of ferrous rubbing, cutting and laminar wear particle in the lubrication oil sample by measuring the size i.e. major dimension and shape i.e. length and thickness ratio of the particle. Ferrography is very much useful for reciprocating machines where vibration analysis has not much scope.

11.0 Integrating Wear Particle and Vibration Analysis :

Two of the most powerful predictive maintenance diagnostic techniaques employed today are wear particle and vibration analysis. Ferrography system when coupled with vibration analysis proved effective in detecting and diagnosing faults. Sometimes vibration data does not indicate any immediate need for corrective action but routine oil sample after ferrographic analysis indicates even minor traces of ferrous rubbing/sliding wear. Wear particle and vibration analysis offer combined benefits to the user.

12.0 Wear Particle Analysis for Light Water and Heavy Water lubricated Systems :

Ferrography is also being effectively applied to monitor the light water and heavy water lubricated bearings, gear assembly and other components in western countries. The appropriate method is to utilise special wash aqueous solutions formulated to analyse the light or heavy water sample.

13.0 ACKNOWLEDGEMENT

The work presented is the outcome of efforts by all maintenance personnel of the Indain Nuclear Power Plants and Headquarter. The authors gratefully acknowledge the continued guidance, encouragement and support provided by Shri Y.S.R. Prasad, Managing Director, NPCIL and Shri Ch. Surendar, Executive Director (Operations), NPCIL.

SL. NO.	LIST OF EQUIPMENT	PARTICLE QUANTIFICATION (PQ) INDEX						
		NAPS			KAPS			MAPS
		1	2	3	1	2	3	1
		Dec. 94	Mar. 95	Jun. 95	Feb. 95	Jun. 95	Sep. 95	Apr. 95
1	CHILLER-1				8			
2	CHILLER-2		13		8			
3	CHILLER-3	3	9					
4	TURBO GENERATOR 1 (MOT)	12	4		6	7		
5	TURBO GENERATOR 2 (MOT)	3	10		4	13		
6	DG SET I-1	18	8	5	9	13	10	
7	DG SET I-2	3	7	5	10	13	10	
8	DG SET I-3	6	20	4	10	12	6	
9	DG SET II-1	23	10		8	11	3	
10	DG SET II-2	39		6	6	9	3	
11	DG SET II-3	15	7	5	12	14	6	
12	COMPRESSOR-1	27	18	4	5	13	2	
13	COMPRESSOR-2		26	8	9	15	3	
14	COMPRESSOR-3	11	9	3	6	7	3	
15	COMPRESSOR-4	17	10	5	14	9	4	
16	COMPRESSOR-5	15	7	10	4	17	4	
17	COMPRESSOR-6	25	5	3	7	16	3	
18	BOILER FEED PP-1 -I							
19	BOILER FEED PP-2 -I							
20	BOILER FEED PP-4 -I		9					
21	BOILER FEED PP-5 -I		8	8	11	5	6	
22	BOILER FEED PP-6 -I	5	9	5		11	2	
23	BOILER FEED PP-7 -I				3	21	168	
24	BOILER FEED PP-1 -II							
25	BOILER FEED PP-2 -II							
26	BOILER FEED PP-4 -II	14	5					
27	BOILER FEED PP-5 -II	9	7	5		14		
28	BOILER FEED PP-6 -II	11	9	3	6			
29	BOILER FEED PP-7 -II				4	12		
30	AUX. BFP-8 -I (DE SIDE)		5	10		16	3	
31	AUX. BFP-8 -I (NDE SIDE)					15	6	
32	AUX. BFP-9 -I (DE SIDE)					14	13	
33	AUX. BFP-9 -I (NDE SIDE)					5	3	
34	AUX. BFP-18 -I		14				4	
35	AUX. BFP-8 -II (DE SIDE)	24	10	4		7		
36	AUX. BFP-8 -II (NDE SIDE)					6	8	
37	AUX. BFP-9 -II (DE SIDE)					12		
38	AUX. BFP-9 -II (NDE SIDE)					11		
39	AUX. BFP-18 -II	5	8				8	
40	PY COOLANT PP-101	13						
41	PY COOLANT PP-102	25					3	
42	PY COOLANT PP-103	13					6	
43	PY COOLANT PP-104	10						
44	OSU 01 -I		15	7	8	8		
45	OSU 02 -I		7	6	14	8	3	
46	OSU 03 -I		8	10	14	8	5	
47	OSU 04 -I		7	7	14	9	4	
48	OSU 01 -II		7	6	2	18	5	
49	OSU 02 -II		8	5	4	7	4	
50	OSU 03 -II			8	5	6	5	
51	OSU 04 -II		8	6	2	6	2	
52	FMP 1 -I				16	25		
53	FMP 2 -I				18	33	6	
54	FMP 1 -II				9	16	16	
55	FMP 2 -II				7	20	3	
56	DOM. WATER PP #1001				46		8	

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INTERNATIONAL ADVANCES IN DESIGN, MANUFACTURE, TESTING, OPERATION AND MAINTENANCE OF FUELLING MACHINES AND FUEL HANDLING EQUIPMENT FOR HEAVY WATER REACTORS

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1. INTRODUCTION

Heavy Water Reactors have performed exceedingly well by achieving high capacity factors, and currently make a substantial contribution to the generation of nuclear power. Most of these reactors use natural uranium as fuel and need regular on-power refuelling to maintain the power level. Thus the high availability and capacity factors achieved by these reactors are largely dependent on successful and consistent performance of on-power fuel handling systems.

On-power refuelling is a challenging task as it involves opening and closing the reactor pressure boundary and maneuvering hot fuel bundles while they are producing energy by fission process. A combination of complicated mechanisms working with high precision and extremely high degree of reliability are employed to perform a variety of functions in quick succession for accomplishing the objectives of on-power refuelling.

A typical refuelling cycle consists of as many as a thousand precise steps to be executed in a sequential manner, and are to be repeated shift after shift, day after day. Careful planning and management of operation and maintenance activities is required to keep the system working in such a busy schedule. Design of equipment for fuel handling systems encompasses various fields of specialisation. The large sized equipment like the fuelling machine bridge are designed as rigid steel structures while their intricate mechanisms resemble high precision machine tool components. The fuelling machine head is a Class I pressure vessel. Various mechanisms employed in it work in water environment, without any conventional lubrication, and still have a high level of wear resistance. The whole system operates on a remote controlled automatic operation from the control room, requiring a host of sensing and remote actuation devices and a complex logic system to ensure proper sequencing and safety through interlocks. All major equipment and piping are analysed for seismic qualification and various scenarios arising out of postulated abnormal conditions like common cause failures are analysed in detail. Adequate provisions are made in the design for prevention of accidents and mitigation of their consequences. Thus design of fuel handling system involves assimilation of knowledge of a wide range of engineering sciences and analytical methodologies. The manufacture of various equipment employs use of a variety of special materials and modern processes almost on the frontiers of technology.

The evolution of fuel handling systems and equipment has been an on-going process and improvements are being made in it all the time like other fields of modern technology, as the experience in design, manufacture, operation and maintenance is gathered, screened and analysed. Break throughs achieved in other fields like analytical techniques, material sciences and computer technology have been gainfully utilised in enhancing the quality and reliability of fuel handling systems. Changing requirements of reactor safety are also reflected in the design of fuel handling systems.

Canada holds the predominant position in the world of heavy water reactor technology. Most of the commercial pressurised heavy water reactors are of CANDU type, a number of them are successfully operating in Canada. Reactors supplied from Canada have been operating in India, Pakistan, Argentina and South Korea, and another CANDU reactor has been recently commissioned in Romania. Design, construction, commissioning and operation of these reactors and their refuelling systems has generated a wealth of experience. Review of such experience gives an insight in various aspects and also ideas for future developments for future PHWRs.

An overview of evolution and advances made in design, manufacture, operation and maintenance in fuel handling systems of heavy water reactors in various parts of the world is presented in this report.

The first International Conference on CANDU Fuel Handling Systems was held in Toronto on May 13 and 14, 1996 under the aegis of Canadian Nuclear Society [Ref. 1]. Presentations were made in this conference on different topics related to design, manufacture, testing, operation and maintenance and vision for future are indicative of the directions in which the progress is currently being made and provided much of the information for this appendix. A workshop on Fuel Handling Systems of Nuclear Reactors was held earlier at Kalpakkam, India in 1986 [Ref. 2]. Information gathered from proceedings of these and other seminars, symposia, conferences, etc. has been summarized in this appendix. A brief chapter on evolution is added to provide the historical background and logical sequence of evolution.

2. EVOLUTION OF FUEL HANDLING SYSTEMS

A typical on-power fuelling machine must have the following features to perform its function:

1. Means to position in alignment with a selected coolant channel.
2. Mechanisms to make a leaktight joint with the channel end fitting before its end closure is opened
3. Mechanisms to handle the closure devices of the channel and the fuel bundles.
4. Capacity to temporarily store a number of new or spent fuel bundles, the plugs removed from the channel and the tools required for refuelling operation.

A typical fuelling machine designed to fulfil all of these requirements consists of a carriage and a head containing a snout, a ram and a magazine assembly. Geometrical configurations of these are distinctly different in different reactors, but they essentially perform the same functions. A fuelling machine has to be obviously compatible with the reactor coolant channel assembly and hence design of its certain features is dictated by the design of coolant channel.

Basically two distinct types of combinations of fuelling machines and coolant channel assemblies have been evolved [Ref. 3, 4]. In one type the string of fuel bundles is supported in the axial direction and prevented from movement due to the drag produced by coolant flow with help of spring loaded fuel latches, fixed at the down stream end of the channel assembly. In this concept on-power refuelling is done in the direction opposite to the coolant flow. New fuel bundles are inserted from the downstream end and spent fuel bundles are removed from the upstream end.

The fuelling machines contain fuel carrier tubes which get inserted into the channel upto the latches and open them to facilitate movement of fuel bundles. As the spent fuel bundles are taken out by the fuel carrier from the upstream end, the remaining string of fuel bundles in the channel is held in position by the fuel latch at the downstream end. Breech type shielding and closure plugs are used in this concept. These are latched in position and torqued by rotary motions of fuelling machine mechanisms. Fuelling machines of this type are used in the Nuclear Power Demonstration (NPD) reactor, KANUPP, and in the Bruce and Darlington stations.

In the other type of the reactor channel - fuelling machine combination, the string of fuel bundles is held inside the channels against the coolant drag force by means of the shielding plugs at the downstream end. The shielding and the sealing plugs have latching mechanisms and are held in position by means of jaws expanded into grooves cut in the liner tube and the end fitting respectively. Fuelling is done in the direction of flow in this concept. So when new fuel bundles are inserted at the upstream end and spent fuel bundles travel to the fuelling machine at the downstream end there is a need to hold the fuel column by some device because the shielding plugs are not in their latched position at this stage. A pair of sensor, stop and pusher assemblies, also known as 'separator' assemblies, are used to perform this function. It separates the pair of bundles received in the magazine tube from the rest of the string, creates sufficient gap for the magazine to rotate freely and supports the rest of the string of fuel bundles against the hydraulic drag force of the coolant flow. Concentric rams with linear movements are used for latching and unlatching of the sealing and shielding plugs. As a fuel carrier is not used in this concept, the space between the smaller bore of the coolant tube and the larger bore used for holding the sealing plug is filled by a guide sleeve which is stored in one of the magazine positions and advanced in position after removal of the sealing plug.

This type of fuelling machine - channel combination is used in Douglas point, Pickering, CANDU-6 plants and all Indian PHWRs. Fuel bundles were moved only by use of the rams in the earlier designs like Douglas Point and Indian PHWRs. In subsequent designs the force of coolant flow itself is utilised for movement of fuel bundles towards the discharge machine. In peripheral coolant channels where the flow is not sufficient to produce this force, a flow assisted ram extension tool is used to generate the axial force sufficient to move the string of bundles upto the fuelling machine at the downstream end.

The fuelling machine carriage design has also been evolved along two distinct concepts. In Douglas Point, RAPP and MAPP a large carriage supported on wheels moves horizontally across the reactor face. A small gimbal frame supporting the fuelling machine head moves vertically along two vertical columns mounted on top of the carriage and moving alongwith it. In all other reactors the columns are rigidly fixed to the building structure. A large mobile bridge provides the vertical movement and a small carriage moves underneath the bridge across the reactor face. However, a wheel mounted carriage similar to Douglas Point is again being contemplated for CANDU 9 design.

The design of fuel handling system is also influenced by the multi-reactor considerations. All reactors up to Pickering were provided with dedicated fuelling machines and fuel transfer systems. Designs like CANDU 6 and CANDU 9 which are developed as stand alone units also have dedicated systems for each unit. However, in Bruce and Darlington Stations having four reactors in each station the fuel handling system is also designed as a composite centralised facility shared by individual reactors.

The fuelling machine heads in these stations travel from one unit to the other on transport trolleys. Dedicated columns, bridges and carriages are provided at each reactor on both sides. The fuelling machine heads are picked up from the transport trolleys by the carriages in the reactor where refuelling is to be done. New fuel loading, spent fuel discharge and fuelling machine maintenance areas are common for all reactors.

The fuel transfer systems have also undergone considerable amount of evolution. All the mechanisms for loading of new fuel into fuelling machines on both sides and accepting spent fuel from them were located in a common fuel transfer room in Douglas Point and RAPP design. These were separated and two sets of dedicated equipment were provided to interact with the respective fuelling machine heads on either sides in Pickering and Indian PHWRs of new generation. Spent fuel is transported through a shuttle or a conveyor to the receiving bay in these reactors. In Darlington and Bruce stations the new fuel port and the spent fuel port are further separated from each other and located in common areas outside the reactor buildings and the fuelling machines visit them for fuel transfer operations. This travel is avoided in CANDU 9 design by fixing the ports on the containment wall. Development of the control systems used for fuel handling has also kept pace with the speedy advancements taking place in the fields of instrumentation and computers. The control system in the initial design was fully made up of hard wired logic units. Advanced computers of appropriate generations are used and sophisticated softwares have been progressively developed for each of the subsequent unit.

3. DESIGN

Evolution of design of pressurised heavy water reactors started in Canada with the NPD reactor and reached its high point in setting up large multiunit generating stations like the Bruce and Darlington. Thereafter as demand for energy in general and nuclear power in particular reached stagnation levels, more large scale nuclear power stations were not set up in Canada. The design of single unit nuclear stations of about 680 MWe capacity, the CANDU 6, was also developed in parallel considering prospects of export to other countries. Initially these units were built in Canada at Gentilly-2 and Point Lepreau. Similar units have been installed at Embalse in Argentina and Wolsong in South Korea. The high level of perfection accomplished in the design of these units has been well demonstrated by successful operation of these units. The CANDU-6 is now well established as a proven design as more such units are being installed and commissioned in South Korea and Romania.

Further advance in PHWR design in Canada was directed towards development of a smaller version, CANDU 3, and a large reactor CANDU 9. The fuel handling system in CANDU 3 is significantly different. A single ended unidirectional refuelling scheme has been envisaged in this reactor. However, this reactor system was probably not economically competitive in comparison with fossil fuel thermal power stations of similar size based on natural gas, which were developed in the same time period. Hence, it remained at the conceptual level and did not reach the stage of setting up of commercial units. Consequently further progress on detailed design work was slowed down.

More concentration has been hence focused on the design of CANDU 9 which is a stand alone single unit version of Darlington with a number of passive safety features added. Significant changes are made in the design of fuel handling system for this concept.

Fuelling machines and fuel transfer equipment based on different geometrical configurations as compared to the first project at RAPP (built with Canadian Collaboration) were designed for standardized Indian PHWRs of 220 MW(e) capacity built in India in the eighties and nineties. Design of fuel handling system for a 500 MW(e) unit based on further refinement of these concepts is in progress. Safety analyses for various postulated emergency situations and seismic analyses have been carried out for these reactor designs.

Design of the fuel handling system and various equipment is also required to meet the high standards of nuclear safety apart from fulfilling the functional requirements to a high degree of reliability. Various safety aspects to be considered in the design, criteria for selection of materials or processes, rules to be followed for design and analysis, documentation to be generated and preserved etc. have been identified and systematically compiled in International Codes and Guides, such as IAEA safety series [Ref. 5, 6] and ASME code [Ref. 7]. Regulatory authorities in various countries have also brought out national codes and guides. Preparation of a guide for design of fuel handling and storage systems in India is in an advanced stage. The process of an independent and systematic review of design from the point of view of safety has been already established in most of the countries.

A summary of the progress in the areas of design is given in following paragraphs by summarizing the key points from papers presented at the first International Conference on CANDU fuel handling systems.

The CANDU 9 Fuelling Machine Carriage is described in Ref. 8. The combination of a robust mobile carriage and the standardized proven fuelling machine head of CANDU 6 is contemplated for CANDU 9. Refuelling will be done by flow assist method in CANDU 9 . The departure from the bridge concept used in CANDU 6 is done for achieving the goals of seismic qualification for future sites with potentially higher seismicities, reduced construction time and improved on-power maintainability of the carriage and the head. Simplicity and maximum use of standard commercially available components are also part of design intents. The key project requirements are as follows :

1. Target life time capacity factor of 90 % for the reactor. Expected refuelling rate: 32 channels per week.

2. Access to all channels, transfer ports for new and irradiated fuel, the rehearsal facility and an ancillary port.

3. Drive mechanisms compatible with computer control.

4. Meeting requirements of CAN/CSA N 285 and ASME Section III sub-section NF.

5. Application of QA programs during design and construction.

6. Lincensability in Canada as per Canadian Codes.

7. Seismic qualification for a Design Basis Earthquake with acceleration greater than 0.2 g.

8. Providing safe and reliable transportation of fuelling machine head.

9. Meeting environmental qualification requirements for normal operating conditions and postulated accidents.

10. Satisfying current human factors requirements (safety and comfort for O & M personnel) and overall minimisation of man-Rem dose for maintenance.

11. Minimise overall construction time (by achieving a high degree of preassembly in shop).

12. Minimise overall cost.

13. Design life of 60 years.

The fuelling machine carriage travels between the fuelling machine vault and a shielded maintenance lock located by its side. The fuel transfer ports and the rehearsal facility are located in the maintenance lock. A shielded door which can be closed only when the machine is in the maintenance lock separates it from the fuelling machine vault. The carriage consists of a base structure, two vertical columns and a vertically mobile frame. The base structure consists of a framework fixed with two wheels and a large turntable rotatable through 90°. The 'X' motion of the carriage across the reactor face is guided by guide rollers travelling along rails fixed to the floor and a guide beam at the top. Two columns of fabricated closed box sections and mounted over the turntable support the ballscrews at the top and bottom and guide the vertical movement of the elevator.

The framework used to support the fuelling machine head consists of a cradle with pivots, a yaw turntable, an inner elevator and an outer elevator. The outer elevator is moved in the vertical direction by 2 ballscrews of 120 mm dia. for 'Y' travel. The 'Z' travel is provided by an acme screw drive fitted between the inner and the outer elevators. Sufficiently long stroke of Z drive is provided to facilitate removal of fuelling machine head on a service cart by using Z drive. Swings in X and Y planes are provided by the yaw turntable and pitch pivots respectively to provide the necessary flexibility for homing. Standard CANDU 6 mechanisms are used for centering. All the drives are provided by electric motors. Each drive has a main drive and a backup. Oil hydraulic system is not used for carriage for any device. Seismic locks are provided at top and bottom to lock the carriage with floor as well as top guide beam before the fuelling machine head is clamped on a channel. These wedge style locks accommodate differential thermal expansion. They are driven by electric motors through acme screw drives and have force control feedback to limit the depth of engagement to just establish a positive contact between the mating faces without causing a lateral movement. Fail safe brakes with a seismically activated and qualified switch protect the carriage from an earthquake occurring while it is in transit. Antilift hooks are provided to prevent lifting of carriage base structure and tensioning springs to avoid buckling of ballscrews in compression due to vertical forces acting upwards. The overall cost is kept to the minimum by maximum use of standard commercially available products and processes for manufacture. Excessive demand for accuracies in machining are avoided by relaxation of tolerances to commercially achievable limits and liberal use of shims to achieve overall dimensional and alignment requirements.

The CANDU 9 Fuel Transfer System is described in Ref. 9. Fuel transfer system for CANDU 9 has been evolved from the systems used in CANDU 6 and Darlington station with further modifications to match with the fuelling machine design and other requirements of CANDU 9. The new fuel transfer system in CANDU 9 is similar to CANDU 6 with an important difference in the point of loading of the new fuel.

In CANDU 6 the new fuel is manually brought inside the containment and loaded on a loading trough. It is pushed into a rotary new fuel magazine by a ram. From the new fuel magazine it is transferred into the fuelling machine head through the new fuel transfer port by remote controlled operation from the control room. Shield plugs and gate valves are used to provide radiation shielding and check spread of contamination.

In Darlington the new fuel ports are located in the fuelling facilities auxiliaries area (outside the containment) and the new fuel port penetrates the containment wall. The integrity of containment is maintained through gate valves. The fuel bundles are loaded into a trough and pushed into the magazine by a ram. They are transferred from the magazine into the fuelling machine through a transporter located in the new fuel port by using a two stage ram.

In CANDU 9 the new fuel loading trough and the magazine are located outside the containment but are similar in design to CANDU 6. The rams consist of stainless steel racks driven by pinions and servomotors. The containment integrity as well as checking spread of contamination is achieved by a system of duplicated valves.

Irradiated fuel transfer in CANDU 6 consists of discharge and transfer equipment located in the reactor building and transfer of spent fuel bundles to the reception and storage bays through the discharge canal. Irradiated fuel is transferred from fuelling machine head into the elevator through the port in air above water (by lowering heavy water level in the head). The bundles are lowered by the elevator onto a conveyor cart which is taken to the reception bay. Fuel bundles are thereafter transferred to the storage bay using manual tools.

In Darlington the transfer of fuel bundles from the fuelling machine head in to a shuttle takes place through a port located below bay water elevation. An air chamber located underwater is used to separate heavy water environment of F/M head from the light water in the bay and the bundles are actually transferred in an environment of compressed air in the chamber. The shuttle containing the bundles is retracted in a ladle and then on to a vertical elevator. The bundles are transferred from the shuttle into the storage modules. Alignment of the modules with the elevator for transfer of fuel bundles is achieved with help of vertical movement of the elevator and horizontal movement of the module conveyor.

In CANDU 9 the system is designed to minimise fuelling machine cycle time. Two small spent fuel storage bays are located just outside the containment wall, one on each side for the respective fuelling machine to directly discharge the spent fuel. Capacity of the bays is to store spent fuel for 6 full power years of operation. The basket modules used to receive spent fuel are designed for direct transfer into a dry storage cask after 5 years of storage. A port magazine located near the outer face of the containment wall but inside the storage bay is used to receive the spent fuel bundles from the fuelling machine head through the irradiated fuel port fixed on an embedded part running through the containment wall.

The heavy water level in the fuelling machine head as well as the port magazine is lowered by pumping compressed air in them and the spent fuel bundles are discharged into the port magazine by fuelling machine rams in the compressed air environment. A provision is made for carrying out dry sniffing while the bundles are passing through a transition piece. The transition piece also provides the flexibility to accommodate uneven settlements or differential movements of reactor and spent fuel storage buildings in a seismic event. The bundles received in the magazine are then lowered in light water within the port magazine. A heat exchanger at the bottom of the magazine is provided to cool the water and the bundles. The bundles are subsequently transferred into vertical basket modules through a tilting mechanism.

Thus major features of CANDU 9 fuel transfer system are commonality of equipment between the new fuel and irradiated fuel mechanism, reuse or adaptation of technology used and proven at other CANDU plants. The design provides adaptability to a wide range of site conditions, low capital cost, short construction schedule, minimum complexity, high availability, low operating and maintenance costs and a generic approach for future developments.

The evolution of the Darlington Fuel Handling Computer Systems are described in Ref. 10. Three areas of the Darlington station fuel handling computer system have been or are being upgraded to current technology levels within the framework of the existing fuel handling control system. These are processor, disk drive and LAN.

The original control system consisted of three independent control systems, connected together, each in turn consisting of three computers. This system was overloaded causing delay in display updates and false stops to fuelling programs. This problem was solved by using efficient softwares and faster processors.

Subsequently, it became necessary to use varying quantities of non-standard long fuel bundles in each channel to control fretting of pressure tubes. This change created additional work of keeping track of these long bundles and hence complicated fuel management needed larger software. It also necessitated an electronic communication between fuel handling system and the fuel and physics department. A comprehensive upgradation program was hence taken up.

PDP 11/70 processor was replaced with more modern and upgraded processor QED 95 D. With these the time of update of CRT displays reduced from 8 seconds to less than 1 second. Average loading of CPU is reduced from 100 % to 70 %, only 5 boards are needed instead of 22 in the original system and the power consumption reduced from 40A to 13.7A.

The disk drive was updated to provide a modern high capacity sub- system to increase the executable and data storage size and improve reliability and maintainability. The new drive has to be compatible with other existing hardware. The disk drive replacement consists of two Seagate 1.0 Gb Winchester type units to be the on-line units and one Syquest 540 Mb removal hard disk cartridge style unit as the off-line unit in each fuel handling computer system. The selection was made on the basis of high reliability statistics, long warranty periods, competitive prices and good trade publication reviews.

A need for interconnection of fuel handling computers to the Information system LAN and to the fuel and physics LAN was strongly felt. The requirements included ability to automatically transfer long fuel bundles logs to fuel accounting system, fuel handling logs to electronic storage media for long term archives and to Information system LAN for technical surveillance and support activities. The existing system does not have the capacity or speed to meet these requirements. Hence, a new product called TCPware has been chosen to interface with ethernet card to be provided to each computer. The fuel handling computers will be then connected with fibre optics cables to a single workstation. The workstation will act as gateway between fuel handling control system and the outside world, fuel and physics LAN and Information System LAN.

This change will be implemented gradually by first making only one way communication from fuel handling to outside world and making other potential uses effective after observing satisfactory performance and building up adequate confidence level.

The application of configurable logic in nuclear fuel handling is described in Ref. 11. The fuel handling control systems at Bruce have been in operation since 1977. Several changes (of the order of 40-50) have been made in the operating and protective logics and more changes are bound to occur as new requirements of safety and applications such as use of fuelling machines for coolant channel inspection gets added. The original hardware has aged and reached obsolescence thus making spare parts for replacement scarce. Reverse engineering to suit each new individual replacement part is very costly and likely to create new problems as thorough testing of each such change would be difficult.

Hence, alternative solution to improve and update the fuel handling control system were evaluated. One option was to use relatively simple PLCs and new latest off the shelf available computers. However, this will involve conversion of software and uncertainties about its behaviour as the new computer systems have more complicated features as compared to the original design and hence the converted software will again need extensive testing and debugging with sophisticated tools.

Another approach is to look at solutions where software element is removed, yet a changeable design is implemented. This is possible with configurable logic devices more generically termed as programmable logic devices (PLDs). Such devices with as many as one hundred thousand gates per device providing the capability of connecting to hundreds of inputs and outputs are now available. PLD, can be configured by a user with inexpensive PC based design tools. So logic design and changes can be implemented by the customer without need of support from the manufacturers. It can be quickly done in the field, without removing the device from its application system. Security can also be ensured by making the devices non-changeable. The flexibility and capability of the PLDs offer many advantages. The speed and logic capability can be used to evolve completely dedicated logic for protective functions. On the other hand hard wired relay logic can also be readily implemented using PLDs.

Protective functions of fuel handling system of Bruce A can be divided into sets/ sub-systems like bridge and carriages for four units, the trolleys, fuelling machines service area drives and auxiliaries. Eight identical modules would accommodate inputs, outputs and logics for eight subsystems. The logic would be unique for each of them. Six identical modules can be used for six control systems, one for each fuelling machine. The tools for creating the configuration data for gate arrays and for simulating the performance of the resulting circuit offer full CAD capability. Logic can be entered in familiar representations, like Boolean logic statements, truth tables, ladder diagrams etc. The tools generate the configuration logic directly. All of the design and system testing can be simulated before committing to actual circuitry.

An interface module has been designed for testing of software changes by simulating the fuelling machine at sufficient speed. This same interface can be used for high speed transmission of logging data to station LAN or to another PC for sorting and analysing to identify system problems and maintenance needs.

In the present system it is not possible to test all combinations of all signals used in the protective logics of an individual drive. With the independent logic of PLD more complete testing can be done after making any change in the logic to ensure that the requirements are met. It is also possible to segregate the subsystems in modules so that a change in one system does not affect the others.

Design of a programmable logic control system for ventilation system was recently done for Bruce A. The logic was defined, configuration gate array was established and simulation tests were run to prove the design in a relatively short time. Implementation cost is expected to be negligible as compared to a software based approach and commissioning effort would be significantly reduced.

The main risk in use of this technology is that custom designed hardware is required. Designing of appropriate interface demands skills and experience. Reliability of interconnection to external devices is a challenge. There could be delays in acceptance of this new technology by the regulatory authorities.

Considering all aspects, field programmable gate arrays provide an inexpensive, yet versatile means for implementing control and protective logic. The reconfigurability will accommodate changes and expansion. The system logic can be well defined and tested due to its deterministic nature. Complete testing will reduce effort in commissioning. Separation of logic in modules will reduce interdependence of functions.

The design of a fuelling machine bridge and carriage meeting seismic qualification requirements is described in Ref. 12. Fuelling machine bridge and carriage are designed to meet requirments of ASME Section III subsection NF and are required to be seismically qualified for on-reactor as well as off-reactor modes. The fuelling machine carriage mounted on wheels used in RAPS and MAPS was found to be unsuitable for withstanding strong seismic disturbances anticipated at certain sites. Hence, a bridge with fixed columns was used for the standardised design of 220 MWe PHWRs and 500 MWe PHWRs to be built in India.

Initially the design was done using static analysis methodology. The columns and tie members were analysed as continuos beams and deflections under load were calculated using moment distribution and super imposition methods. Natural frequencies of vibration for the beams were thus determined. Certain important modifications were made in the design based on this analysis. The radiation shielding which was initially considered as integral part of the bridge was separated and provided as a Roll- on shield rolling on the fuelling machine vault floor. The number of tie members and their size was increased and the columns were fortified with additional bracings. Rating of brakes used for X and Y drive were augmented to withstand maximum seismic forces.

A detailed dynamic analysis of the fuelling machine was carried out using finite element technique . Cases when fuelling machine is standing free and when it is clamped onto an end fitting of a coolant channel were separately considered. Three dimensional beam and stiffness elements were used. A total 435 nodes and 592 elements were identified while making the mathematical model to as closely resemble the real structure as possible. End release code facility of computer code was used to simulate exact end conditions like freedom to slide or rotate etc.

Analysis was carried out for dead load and two levels of earthquake viz. operating basis earthquake (OBE) and safe shutdown earthquake (SSE) using appropriate response spectra generated by enveloping response spectra of different floors. 32 modal frequencies for coupled mode and 16 for uncoupled mode were extracted by the analysis. Stresses and deflections experienced by various members were calculated and compared with allowed values as per ASME code. Though the structures are qualified to meet ASME code requirements, the components subjected to maximum stresses approaching allowable stress value have been identified. They will be strengthened to obtain higher a margin of safety. Deflections are found to be very small and do not cause any concern.

The design of fuel handling system for 500 MWe Indian PHWR [Ref. 13 and 14] has been evolved incorporating the following special and innovative features in design of fuelling machine head and the fuel transfer system

1. The magazine rotor is of bolted design, thus avoiding a difficult welding process and attendant problems like distortion. It is simply supported at both ends instead of cantilever design in earlier designs.

2. A duplicated drive using two pinions, gear trains and motors is provided to give redundancy.

3. The magazine drive shaft seal is designed as an easily replaceable cartridge to facilitate maintenance.
4. Snout assembly is also made modular and hence easy to remove and replace without needing opening of magazine end cover.

5. A single start ACME thread is used for snout clamping mechanism to give assured irreversibility.

6. The most innovative and revolutionary change is made in Ram Assembly by use of co - axial rack and pinion assemblies for B- ram and C-ram. These are driven by two pinions each which are driven by high torque low speed oil hydraulic motors, thus providing a balanced, duplicated drive. The rack and pinion are considerably robust as compared to ballscrews and are expected to give a longer life.

7. A provision is made in the design of both B and C Rams to drive them by heavy water pressure in emergency.

8. C-Ram positioning scheme is fully outside heavy water environment and would give more reliable and trouble-free operation.

9. A transfer magazine will exchange new fuel bundles with spent fuel bundles in fuelling machine head thus reducing operating time for fuelling machine.

10. Drive and position sensing systems of transfer magazine are located in accessible shielded areas to facilitate maintenance and easy access in emergency situation.

11. Provisions are made for ensuring cooling of spent fuel bundles under all abnormal conditions.

12. Shuttle station is provided with a simple wire rope drive mechanism, fuel shutter to prevent accidental dropping of spent fuel and special dampener to absorb impact due to postulated free fall of shuttle carriage.

13. The shuttle is provided with a piston to ensure full face contact when the fuel bundle is discharged out of the shuttle.

14. Heavy water system of fuelling machine can be totally isolated from PHT and run independently using a special tank and associated circuit.

15. Controls are based on programmable logic controller.

16. A dedicated data acquisition system in conjunction with an expert system would be designed to work as the operator support system.

17. A dedicated calibration and maintenance facility (CMF) is provided to test and calibrate fuelling machine heads in a tritium free environment. It consists of three coolant channels, facilities to test fuelling machine heads independently as upstream or down stream machines and simulating all other conditions as in the reactor. Separate facilities for testing individual assemblies (ram assembly, snout assembly, seal plugs etc.) are also provided.

In brief, the fuel handling system for 500 MWe PHWRs would be built using simple, robust and reliable mechanisms of modular construction, facilitating maintenance and ensuring high availability of the fuelling machine to cope up with the work load of regular on-power refuelling. Layout of the fuel handling areas, design of control systems and provision of CMF are designed to aid operation and maintenance of fuel handling system to ensure maximum availability.

Wet storage pools provided in BRUCE station are getting filled up and transfer of fuel bundles to a dry storage will be required before turn of this century. A dry storage container similar to that used in Pickering N.G.S. with certain modifications has been recommended as a solution [Ref. 15]. The evolutionary development has been done to achieve simplicity in design,, construction and use. Main differences in two containers are given below:

1. Fuel is stored in trays instead of modules and trays are directly loaded into the container.

2. Loading into the container will be done in dry state outside the bay, thus eliminating the need of bay floor reinforcement, impact pads etc.

3. The container is 160" high, will weigh 93 tons and hold 600 bundles as compared to 140" high, 62 tons and 384 bundles in Pickering.

4. A double lid of circular shape which is easy to machine and get the required fit is provided against a single square lid made of fabricated steel and filled with concrete in Pickering.

5. Welding is of only 1/2" to 3/4" size and is easy to accomplish by a simplified automatic welding machine as compared to a massive 1-1/2" welding with 11 passes done in Pickering.

6. The secondary lid facilitates leak testing of the primary containment achieved by the first lid and also provides a secondary containment.

7. The simpler construction has achieved 28% reduction in life cycle cost over Pickering.

8. The circular lid permits the lifting provision located at the top face.

9. While the Pickering container is transportable off site, the Bruce container is designed for only on-site handling and storage.

10. The Bruce container can be stored outdoor while the Pickering container needs indoor storage.

Ref. 16 describes the design of the fuel handling equipment which requires detailed structural analysis to be done for major components under different normal operating states and accident conditions to ensure their structural integrity. It is carried out using rules and the methodology specified by the relevant national and international codes. Since this work is of a highly specialised nature and requires knowledge and skills of a different kind, it is generally entrusted to specialist analysts. All the necessary inputs like design drawings, material specifications, code classification, jurisdictional boundaries and loading considered for design, service and test conditions are given to them . A comprehensive picture of the requirements of the relevant codes for the analysis work, reports to be submitted for regulatory clearances, inputs required for the analysis and the results obtained is presented in this paper.

According to the codes different types of reports are required to be generated and furnished for components and supports of different classes. The classification is primarily based on the potential hazard due to the failure of the component. The types of such reports are as follows:

1. Certified Stress reports for design by analysis together with Certified Review of Design reports are required for Design Registration.

2. Certified Design Report Summary of standard supports designed by analysis.

3. Design Reports not requiring certification and registration.

4. Load capacity data sheets and catalogues for supports to be qualified by load rating method.

5. Design calculation are required for components and supports as per original design and for deviations due to fabrication errors, modifications in design, change in operating conditions, change in installation conditions, item replacements or repairs etc.

Classification of pressure retaining systems, components and supports and their jurisdictionary boundaries are given in various sections of the governing code. Design, service and test loadings are defined on the basis of criteria given in the code. Analysis procedures can be either by classical or finite element method or the design made by design rules given in the code. Considerations for structural analysis as per the code include allowable stress limits, stress intensity values, fatigue evaluation, protection against nonductile fracture, evaluation of buckling of vessels under external pressure and of beams under compressive loading, load rating method, experimental stress analysis etc.

Design and analysis of CANDU reactors is done to meet the requirements of Canadian standards (CSA) and ASME Boiler and Pressure Vessel Codes.The fuelling machine head is classified as class 1 pressure retaining component (sub-section NB) and the fuelling machine support frame, carriage, bridge, columns etc. are class 1C of CSA (sub-section NF). In this case, ASME has no rules for mobile supports and CSA permits a wider range of materials.

Earthquake loads are part of level C loading conditions for the fuelling machines, its support structures and the interfacing systems, i.e. process system, reactor structure, coolant channels and feeders. The seismic analysis is done by finite elements method using beam and spring elements. The seismic models representing various configurations such as the fuelling machine attached (at seven different locations) or unattached to the reactor or new or spent fuel ports, midway on the maintenance track etc. are developed. Seismic inputs are taken from the analysis done for the reactor building, in the form of floor response spectra or acceleration time history. The seismic loads generated by the analysis are used in the seismic qualification of the equipment by analysis or by testing.

The methodology adopted in stress analysis of the fuelling machine head is an optimum combination of classical and finite elements methods using computer aided capabilities of modelling and meshing. The analysis of fuelling machine support structure is done using beam (linear), plate and shell type elements and load rating given by suppliers of standard proprietary components.

Following suggestions are made for improvements in the interface between analysis and the other elements of the product cycle.

1. The required inputs should be given timely and adequately.

2. Design specification should have a section on 'Requirements for analysis'

3. The analyst should choose the type of element.

4. The load ratings provided by the manufacturers should be compatible with the code definitions.

5. Close co-ordination between the structural analysis, design, manufacture and maintenance (for repairs/replacements) is vital to find acceptable solutions.

Ref. 17 describes a mini-SLAR delivery system. Ontorio Hydro decided in 1993 that Bruce units 1 and 2 would be run only until they meet fuel channel life limit and they would not be retubed for life extension. As per this decision unit 2 was to be "laid up" in September 1995 and 110 channels in unit 1 should be inspected for contact between the pressure tube and the calandria tube by 1995 to decide about its continuation of operation. Hence, there was an urgent need to develop a system for inspection for location of the spacers and their immediate repositioning to avoid contact between the pressure tubes and calandria tubes leading to blister formation and rupture.

The Spacer Location And Repositioning (SLAR) system was already developed and quite successfully used in units 3 & 4. However, it was not possible to transport this system to unit 1 because of limitations of movement of the trolley on which it was built. Hence, there was a need for an alternative delivery system (mini SLAR) to be made available very fast for unit 1 service. This was accomplished by adopting a very innovative strategy for design, manufacture and commissioning of the system. Instead of following the normal logical and chronological sequence, work was started on as many fronts as possible simultaneously. A multi disciplinary group of persons from design, operation, maintenance, inspection etc. of fuel handling and other reactor systems was put together and the design was evolved as the work progressed.

Since tooling for inspection and retubing was already available and personnel at site had gathered substantial practical experience on their use, a start was made by using the available modules and putting them together with structural members of simple construction which were fabricated at the site workshop sometimes using only simple hand drawn sketches. Proper drawings and analysis for regular documentation were also made in parallel. These were revised a number of times as the system was being constructed, refined, trial operations were done and several on-the-spot additions and improvisations were made.

Initially it was thought that the sealing and shielding plugs could be removed by the fuelling machine. However, in view of the practical difficulties in alternate operation of fuelling machine and SLAR in the same space, a decision was taken to perform this job by the mini SLAR itself. Thus the mini SLAR consisted of a snout in the front and three modules on a shielding frame in the rear, one each for sealing plug handling, shielding plug handling and SLAR tool. Similarly control hardware and software was built in parallel with use of easily obtainable parts and modules. A complete full scale mock up of the reactor vault, the maintenance platform, the fuelling machine bridge, full length pressurised channels, cut away channels for viewing internal operations and a full control console with viewing and communication system was set up. Each mechanism and subassembly was individually tried out as it came and modified to solve problems or add new features which were suggested by the operators during the trials. Integrated testing of the subassemblies and controls was also done similarly. Involvement of operating staff made it readily acceptable for use.

Complete mini-SLAR system was thus built within 14 months, and 109 channels were SLARed in 31 days. As a result, replacement of 27 channels was avoided and life of the reactor unit 1 was extended up to the year 2000 instead of the earlier estimation of 1997.

This mini SLAR has numerous potential future uses. It is being adapted for mounting on the fuelling machine carriage for remote operated homing and clamping on the channel for draining them in unit 2 during lay up. It can be modified to push fuel bundles into a downstream fuelling machine during a shut down condition with depressurised PHT. The fuel bundles can then be reloaded after SLARing. It can be modified by changing modules and used to carry a number of tools for channel inspection, gauging, closure seal face repairs, wet scraping of pressure tubes, channel isolation, and visual inspection.

4. MANUFACTURE

Introduction:

Fuel handling equipment comprises of a number of complex mechanisms, each consisting of several intricately shaped and precision machined components, made of a wide range of prime quality materials. Manufacture of these equipment primarily involves use of state of the art technologies for forming, joining, machining and finishing of metals and modern techniques for inspection and assembly. While basic technologies used in manufacture of other industrial products and machineries are employed to a large extent, certain techniques have been specially evolved for manufacture of fuelling machine components. Since most of the components are custom built to specific shapes and the batch quantities of individual items to be produced at a time are small, it is not economical to use automated mass production technology for their manufacture or inspection. Hence, the manufacture remains a combination of application of modern technology and use of innovative skills.

A general overview of manufacturing processes and other aspects is given in this section. Some information in this section is taken from Ref. 2, 18 and 19.

Evolution of manufacturing processes :

Evolutions of design and manufacture generally go hand in hand. A designer needs to have a clear perception of how the component visualised by him can be manufactured because possibility of manufacture at a reasonable cost is very essential for a design to be converted into a physical artefact. Considerations of the functional requirements to be satisfied by each component as well as various aspects related to its manufacture are viewed together for finalising all decisions in design, from selection of the most optimum concept upto fixing detailed engineering parameters like dimensions and tolerances.

The design of a product would be generally based on the well established capabilities of the prevalent manufacturing technology, with an element of optimism or challenge often added with a view to reach out a little beyond the hitherto known frontiers. Enterprising manufacturers respond positively and work out the innovations and improvisations in their processes to meet the challenge.

Progress made through basic and applied research in laboratories is another major source of new developments in technology. Quite often an invention in one field leads to breakthrough advances in other fields. For example, the advent of computers has revolutionised a wide spectrum of branches of engineering and technology; from stress analysis of complicated structures to precision machining of intricate shapes. A designer can use these new techniques for refinements in design and set further challenging goals for the manufacturer. Thus progress made in design and manufacture motivate each other to expand the frontiers of technology.

Manufacturing aspects of Fuel Handling Equipment :

Engineering of major equipment like fuelling machine head, carriage and fuel transfer equipment had been originally done by the designers of CANDU reactors, based on use of products and processes then available in the North American continent. Guidelines for selection of materials and processes for welding, heat treatment etc. of pressure retaining components and structural supports of different classes are given in ASME codes. These are used to a large extent as prime reference standards and certain special additional stipulations are given in the specifications and drawings to ensure fitness for use in fuel handling system. Materials of best quality and the most modern manufacturing processes are selected for manufacture of intricate internal components. These are thoroughly tested in engineering laboratories as well as in prototype machines and further improvements are made in the selection of materials and processes. Experience obtained in operation and maintenance and especially results of studies made on early failures of certain components is used to bring about further improvements to enhance reliability and life. Finally, after having achieved a high level of performance, a reassessment is now being made to move towards simplification, standardisation and cost reduction.

The scenario of evolution of manufacturing processes vastly differs in different countries. The Indian experience in this area is quite interesting. When the work on manufacture of the first fuelling machine head and carriage for RAPP-2 was taken up, there was a considerable gap between the technology required for their manufacture and the commercially available range. This was because the local industry had been developed till that period primarily for mass manufacture of specific products like particular models of automobiles, consumer goods or machine tools. The technology required for manufacture of fuelling machines was gradually built up from the grass roots level with help of certain enterprises who had the potential as well as the will to take the challenge.

After a thorough study of the design, wherever direct utilisation of the available technology appeared possible, manufacture was proceeded with. Innovations by way of modifying certain machine tools or developing special toolings, jigs and fixtures were attempted for certain difficult parts. Technology for certain processes like nitriding and special hard chrome plating was established by importing the necessary equipment. Manufacture of certain components appeared prima facie impossible with available resources. Such components were imported. A decision was taken to also import all the raw materials and standard proprietary items for the first project and concentrate on efforts to initially develop only the custom built items to be made as per their respective drawings and specifications. After review of the success achieved in manufacturing the items identified in the first phase, the technology was extended to manufacture the remaining components. Design and material of many items were changed to suit the manufacturing routes. Alternative sources for supply of raw materials and proprietary items were also located in the meanwhile by parallel actions. All fuel handling equipment required for the first indigenously built project of MAPS could be successfully manufactured in India by following this multi-pronged strategy.

Detailed design of equipment for the next generation of reactors (NAPS onwards) was done following a ground rule that it must be possible to manufacture every item using already established technology, probably with a little extra effort. Items which were found to be difficult to make in India, such as alloy steel castings of radiographic quality and weldments of martensitic stainless steel materials were altogether eliminated or replaced by fabricated structures and bolted assemblies respectively. However, progress attained by industry in manufacturing technology, like advent of numerically controlled machines and electric discharge machines, was also watched with keen interest and profitably utilised in manufacture of fuel handling components in order to upgrade their quality or reduce cost and time. Use of optical alignment instruments was pioneered in India for the nuclear power plant equipment, including fuel handling. It is now well established and used in other industries as well. Manufacture of fuel handling equipment in a way acted as a catalyst in bringing about a general upgradation of the technological base. Several practices of quality assurance and documentation, which now form the basis of ISO 9000 certification had been already established during manufacture of these equipment.

Fuel handling equipment can be broadly classified under the following categories from the point of view of their manufacture:

1. Stainless steel pressure vessels for fuelling machine head and fuel transfer equipment.

2. Carbon steel fabricated structures such as fuelling machine carriage & bridge.

3. High precision stainless steel components for under water application.(Internals of fuelling machines and fuel transfer system)

4. Heat treated low alloy steel components like wheels, axles, shafts, rails, guides, high tensile fasteners etc.

5. Miscellaneous steel fabrications like base plates, brackets, covers etc.

6. A wide variety of proprietary items.

Details of processes employed for manufacture of equipment under each category are briefly given below:

Stainless Steel Pressure Vessels:

Fuelling machine head pressure housings subjected to high pressure are made by forging processes. Basic raw materials are produced by processes like Vacuum Arc Remelting (VAR) or Electro Slag Refining (ESR) to obtain a high level of cleanliness and freedom from inclusions, segregations etc. Forging and heat treatments are done under controlled conditions and they are subjected to non-destructive examination by ultrasonic technique. Other vessels used for low pressure application are fabricated from plates by following stringent welding qualification requirements stipulated by ASME codes. In addition, special precautions are taken and various techniques are used to control distortion during welding. Welds are inspected by radiography, ultrasonic and liquid penetrant examination techniques. Machining is done either on general purpose or tool room machine tools of adequate accuracy or special purpose machines built to generate particular profiles such as the front end cover.

Carbon steel fabricated structures :

These are also welded and inspected by NDT techniques as per code and are post weld heat treated for stress relief. Internal surfaces of closed box sections are protected from scaling by purging them with nitrogen gas during PWHT. Distortion during welding is again controlled to the maximum possible extent to minimise material removal in the subsequent machining operations. These structures of huge size are required to be machined to a demanding degree of accuracy to achieve precise alignment requirements for the assembly. Certain crucial mating surfaces are finally hand scraped to achieve a full face contact. These equipment are blastcleaned to white metal and painted with special zinc silicate primer and high build epoxy paints. Considering that once installed in the vault it would be very difficult to repaint them frequently, the paints are also thoroughly tested for effects of irradiation on their properties and for decontaminability of the painted surface with use of standard cleaning agents and techniques.

High precision stainless steel components:

Since these mechanisms have to work in water environment, normal low alloy or tool steel materials used in common industrial machinery cannot be used for these applications. They have to be necessarily made out of corrosion resistant high alloy stainless steels. However, common austenitic grades of stainless steels do not provide the necessary strength and resistance to wear, galling, seizing etc. required for certain parts subjected to high stresses or impacts during operation. Hence special materials like precipitation hardenable stainless steels, inconel, hastalloy, aluminium bronze etc are used for manufacture of these components. These materials offer a range of combinations of corrosion resistance and mechanical proprieties after appropriate heat treatments. Certain components are hard chrome plated or nitrided to obtain a very hard and wear resistant surface, while some parts like seal disk are plated with supersoft nickel layer to achieve the high level of leak tightness in service. Each of these processes is unique and requires an extensive amount of research and development effort, production of a large number of samples and their testing and scientific evaluation of the results before all the parameters are optimised and the process is qualified for use. A wide range of finishing techniques like grinding, honing, lapping, broaching etc. are employed to obtain the required dimensional accuracies and surface finish.

Parts which are located outside water can be made of standard materials commonly used for industrial machinery applications and lubricated with oil, grease etc. which also provide protection from corrosion in service. Various rails, guides, rollers, axles, shafts, couplings, lever arms, pins etc. used for drives of different equipment come under this category. Though conventional manufacturing processes can be used for production of these parts, quality assurance requirements of a considerably higher order are instituted because they are also part of remotely operated critical equipment and their reliability is required to be of a higher level as compared to general engineering products. A final surface treatment like electroplating or phosphating is given to these components for their preservation and corrosion protection. Miscellaneous simple minor items such as brackets, spacers, covers etc. are also manufactured by using almost the same level of technology and quality assurance practices as used in case of important structures mentioned above.

Standard Proprietary Components :

The list of standard proprietary components would be very long and growing with more and more new high-tech products entering the market. The most interesting and crucial proprietary part used in fuelling machines is the ballscrew assembly. Considerable efforts have been made in improving this component to obtain a long trouble free and reliable service. Though the pressure housings and other structural parts of the fuelling machine may be considered to have a life longer than the reactor, the ballscrews wear out much faster and need replacements at frequent intervals. Each and every part including the balls, gothic arch profiles, ball return tube and deflector has been closely studied under microscope and several different combinations of materials and minute variations in dimensions in micron levels have been examined in detail. Techniques like preloading of ball nuts as well as unloading of balls by leaving some blank space have been studied. Combinations of loaded balls of harder material spaced by unloaded balls of softer material have also been used with remarkable success. Thus considerable advances have been made in manufacture of fuelling machine ballscrews all over the world, though specific information being of proprietary nature is rarely disclosed. Other drive components have also seen dramatic changes. Ordinary induction motors used in the earliest design were replaced with oil hydraulic motors to get a better control on speed, torque, acceleration, deceleration etc. The oil hydraulic motors are now giving way for introduction of d.c. servomotors, stepper motors etc and their controls based on thyristers and n.c. techniques. These motors give the same advantage of flexibility in control but also are more compatible for integration with computerised control and remove the disadvantage of messy oil spillages and fire hazard associated with oil hydraulics. Technology of manufacture of gears has also made some advances. Irreversible worm gear units have been developed for use in fuelling machine bridge drive in Indian reactors to provide passive safety against the postulated simultaneous failure of the brake and counter balance valves.

Instrumentation and control systems have witnessed a much faster rate of advancement and also obsolescence. Electronic hardware of the present generation has hardly any resemblance with that used in the initial designs. Designers and suppliers of fuel handling control systems have kept in step with the changes occurring on the international scene and used the latest state of the art products to the maximum possible extent without compromising on the requirements of safety and reliability. Similarly latest versions of computers and corresponding softwares are being put to use. Several new tasks like transferring of logged data to a central magnetic media storage, off-line analysis of stored data to study trends and change patterns, real time mimic displays, integration of fuel handling operations data with fuel and core physics data etc. are being identified and explored. These developments have been made possible only due to advancements in manufacture of computers and programmable logic devices having much higher capacity and speed of operation than envisaged before.

Thus manufacture of fuel handling equipment is making continuous progress. Some examples of modern technological processes in use are multiple axis numerically controlled machine tools, co-ordinate measuring machines, laser devices, vacuum bonding, vapour deposition, investment castings, powder metallurgy, wire cutting EDMs etc.

5. TESTING

Considering the complexity of the fuelling machines and their controls, it is necessary to thoroughly test each fuelling machine head in conditions closely simulated to actual operation. It is also wise to perform certain tests after a major overhauling or maintenance before going to a channel. Different facilities designed to meet specific objectives are provided for such testing. They can be broadly categorised as rehearsal facility, on-site test facility and test laboratory.

The rehearsal facility is located close to the reactor within the normal travel of the fuelling machine. It is generally just another reactor channel with all its internal contents and external heavy water connections but provided with additional features and instrumentation. After carrying out maintenance, disassembly, reassembly, recalibration etc. a complete rehearsal of all sequences in a normal refuelling operation is done on this channel to watch and record all important parameters and ensure proper functioning with sufficient safety margins. As the fuelling machine is accessible when it is on the rehearsal facility, any snag can be conveniently rectified without undue exposure.

In some reactors like Indian PHWRs the rehearsal facility can also be used to temporarily store spent fuel bundles in case of an emergency. The fuelling machine or the fuel transfer system can be then attended to and rehabilitated. Redundant and reliable provisions are made to ensure continuous cooling of spent fuel bundles stored in the rehearsal facility in order to qualify it for the intended safety class. However, such storage hampers its primary purpose of testing the fuelling machine and hence it is done in extremely rare emergencies when it is unavoidable. In one recent incident a damaged fuel bundle was successfully handled by first placing it carefully in the rehearsal facility and taking it out directly into a shielded flask to avoid further disintegration during fuel transfer operations.

In multi-unit stations like Bruce and Darlington a service facility is established in a auxiliary building outside the reactor buildings where fuelling machines are regularly serviced and tested turn by turn and kept ready for operation. A pair of fuelling machine heads is mounted on the transport trolley. Three such trolleys run in a common duct under four reactor units to visit either a unit that needs refuelling or the fuel facility auxiliary area to load new fuel or discharge irradiated fuel or to the service facility. All the support systems and controls are available for carrying out testing of fuelling machine heads in this facility.

A Calibration and Maintenance Facility (CMF) similar to this is planned to be provided for Indian 500 MWe reactors. However, the fuelling machine heads are dedicated assemblies for each reactor and they are taken out to CMF only for maintenance and calibration. CMF will be a light water system thus avoiding tritium exposure hazard to the maintenance personnel. It is engineered to give flexibility for testing the fuelling machine heads in pairs in unison or individually and as an upstream or a downstream machine, in hot or cold condition. This facility can also be used to simulate abnormal conditions due to failures for trying out mock up of retrieval operations and special toolings developed for this purpose.

A test facility in an engineering laboratory can be much more elaborate and provided with sophisticated tools for diagnostics and analysis. Such facilities are established in all the countries where fuelling machine heads are manufactured and assembled. Each fuelling machine head is systematically calibrated and debugged part by part, subassembly by subassembly to detect any deficiency in manufacture or assembly. It is then operated step by step, sequence by sequence and finally on full auto mode for complete refuelling cycles. An enormous amount of data is collected during the testing, quickly sifted through, analysed and useful conclusions are drawn. These are fed back to the designers and fed forward to the operators for their information and use. Every fuelling machine head undergoes this testing and is qualified for use in the reactor only after it complete certain number of consecutive cycles of operation without need for any interruption for an adjustment or maintenance. The testing is repeated, if it is interrupted for any reason, until the specified consecutive number of cycles of hot operation simulating actual coolant channel conditions are successfully completed. All modifications in the control scheme, logics, interlocks etc. and changes in the software can be safely tried out at the test facility in an environment free from radioactivity. The facility can also be used for type testing modified channel components, new designs of fuel bundles, seals and gasket materials, development of diagnostic tools, study of flow in the channel and its effects like vibration, fretting etc.

Significant advances are being made in the area of testing . Brief description of these is given in References 20 and 21.

In Ref. 20 the Darlington Fuel Handling Acceptance programme is described. Six Fuelling machine heads are used on Darlington NGS for regular refuelling. Two more heads are delivered as spares. During acceptance testing of the eighth and final head, the charge tube assembly indicated a problem of stalling during movement due to higher than expected torques. Fuelling machine was disassembled and rebuilt using a spare ballscrew and nut assembly. During the second series of testing, the cold runs were successful but stalls reappeared during hot runs.

A thorough investigation was undertaken to find whether the root cause of the problem lied in the ballscrew assembly or fuelling machine head components misalignment. Elaborate instrumentation was done to measure vibration and torques and a series of thirty two tests were completed under different combinations of conditions like on-channel/ off-channel, hot/ cold, pressurised/ depressurised, advance/ retract, with/ without fuel bundles in the carrier tube. Possibility of misalignment and any deficiency in the drive system as a root cause was ruled out by analysing results of these tests. Now a closer look was taken at the ballscrew and nut assembly. The higher torque could be attributed to either pitch/thread depth variations, preload on the ballnut or imperfections in the thread profile. A set of experiments was conducted to narrow down the problem further. The ballscrew and nut assembly was separately tested on a test bench by changing the screw, the nut, bearings, preload, transfer tube and deflectors turn by turn and by adding crud particles in the water. Two parameters, the preload and cleanliness appeared to predominate while others caused marginal variations.

The test rig was hence thoroughly cleaned to remove as much crud as possible, preload was reduced from 600Lbs to 100Lbs and acceptance testing was resumed. However, higher than expected torques were observed again. The ballscrew was inspected more closely. There were indications of double contact of the balls with the flank of the screw thread profile due to certain flaws on the surface and a bend of 13 thou (0.325mm) was also observed. Though a bend of this magnitude is acceptable for operation due to inherent flexibility of the long screw, it was not tolerable for precise regrinding operation. It was hence straightened to within 2 thou (0.05mm) and thread profile was carefully reground in the axial direction without varying the pitch diameter. The screw was reassembled with the fuelling machine and further testing was successfully completed.

CANFLEX is a new 43 element fuel bundle designed for high operating margins. Demonstration of the compatibility of these budnles with a CANDU-6 fuelling machine is described in Ref. 21. CANFLEX has many small diameter pins in the outer rings and large diameter pins in the inner rings to achieve lower linear heat ratings. A demonstration of the CANFLEX bundle in an actual CANDU-6 fuelling machine was necessary to quality it for use in these reactors. The following tests were conducted for this purpose. First the bundles were passed through a `bent tube gauge' to check their ability to pass through the rolled joint section of coolant channel under its own weight. Dimensions such as pitch circle radius, end plate waviness, bundle radii and droop were accurately measured on a co-ordinate measuring machine in free condition and under compression at the mid plane. The bundles were placed in a side-stop fixture to determine the clearance between the end plates and the side stops of a pair of separator assemblies positioned to simulate their positions in the fuelling machine head. The clearances of standard CANDU-6 bundles of 37 elements were also measured for comparison. Accurate measurements of bundle end face were taken to calculate bearing surface area of end caps in contact with side stops. It is desirable to have maximum contact area between pencil end caps and side stops to hold the bundle against the hydraulic drag due to flow, while there should be a definite all around clearance between the side stops and the end plate to ensure that the end plate is not squeezed by the side stops. These requirements should be met for all combinations of dimensions of the fuel bundle and internal bore of fuelling machine within tolerance. Hence, these measurements are taken very accurately.

Another test was conducted to confirm feasibility of grappling of the CANFLEX fuel bundle with the standard CANDU-6 grappling tool used to pull the bundles in case of an emergency requiring defuelling using a single fuelling machine. Two CANFLEX fuel bundles were placed horizontally in a pressure tube of sufficient length with a window provided at the top for observation. Grappling operation with the grappling tool was tried 10 times at random orientations of the tool and repeated with the fuel bundles also rotated to a different orientation. Number of fingers in engagement was recorded for each test. It was observed that sufficient number of fingers could get engaged in all 20 trial operations.

Thus it was concluded that CANFLEX bundles are compatible with CANDU-6 fuelling machines for normal as well as certain abnormal conditions and as such they were qualified for use in reactor from this point of view.

A video presentation on a Fuelling Machine Ram Test-Rig at the International Conference on CANDU fuel handling systems showed the ram test rig developed for use in Gentilly-2 generating station. This being a single unit CANDU-6 type station a pair of dedicated fuelling machines is used for refuelling. Earlier whenever a ram assembly was overhauled it could be tested only after installation in position in the fuelling machine. Finding out and correcting deficiencies in situ was costly and time consuming and increased down time of the fuelling machine. Hence a Ram Test Rig (RTR) has been developed to carry out comprehensive testing in a non-radiation area. It is made of modular construction and mounted on a mobile platform such that it can also be connected to a completely assembled fuelling machine head. In that mode it is called Fuelling Machine Test Rig (FMTR). Thus the ram can be tested either in position in the fuelling machine or when separately mounted on the rear module of RTR. Other components like fuelling machine magazine and end fitting tube are simulated in the assembly of the front module. Control and instrumentation for the operation in either Ram Test Rig (RTR) or Fuelling Machine Test Rig (FMTR) mode are provided by control modules. All electrical, oil and water hydraulic systems are provided in the stationary portion and connected to the ram under testing through hoses and cables.

The reliability of the ram assembly and the fuelling machine has steadily increased and the unavailability due to breakdowns has decreased dramatically since the use of RTR and FMTR. This has resulted in an overall improvement in productivity of fuel handling system and substantial savings in maintenance costs. A similar Test Rig has been recently delivered to Embalse NGS, Argentina.

6. OPERATION & MAINTENANCE

Introduction:

Operating performance statistics for life time capacity factors place various heavy water reactors prominently among top performers in the world. Achievement of high capacity factors was possible only because of consistent performance of fuel handling system feeding the reactors with new fuel by on-power refuelling. Outages caused by failures in fuel handling systems have been negligibly small, contributing to only about one percent of total outage time. On the other hand fuel handling system has been put to work for giving support to other jobs like coolant channel inspection, creep measurements etc. even when the reactors are shut down. Thus the demand for availability of fuel handling system is sometimes greater than the reactor.

During the initial period of operation of a new reactor, from the first on-power refuelling till establishment of an equilibrium core, the effectiveness of fuelling for addition of reactivity is suboptimal because the spent fuel bundles taken out are only partially burnt and the net addition to reactivity by its replacement with new fuel is less than normal. When some of these partially burnt spent fuel bundles are reloaded in another channel for their effective utilisation, net reactivity addition per fuelling is reduced even further. Thus the fuelling machine has to visit more channels for addition of the same amount of reactivity to compensate for depletion due to burning of fuel by fission process for generation of power. This peak in demand comes at a time when the fresh and inexperienced operation and maintenance crews are just getting familiarised to deal with a set of equipment which is still being fine tuned. This puts the operations of fuel handling system under strain. The reactor power may be required to be temporarily lowered on account of low reactivity due to suboptimal performance of fuel handling system during this phase. It is rarely required after the operations are stabilised with an equilibrium core.

Certain abnormal events or accidents have occurred in fuel handling systems due to a variety of root causes. Some of the reasons are undetected errors in the controls, deficiencies in design, manufacture or construction which could creep past the quality assurance checks, insufficient maintenance because of lack of time or due to nonavailability of spares. However, such instances have been few and these have been successfully and competently handled by fuel handling personnel. Even in the case of the worst accidents in fuel handling system the situation could be brought under control and normalised in a matter of a few days and without any significant radiation exposure. Overviews of operational experience in India and Canada are given in Ref. 22 to 25.

The good operating performance and the ability to handle abnormal situations has been possible because of the combination of good design as well as operating practices. Most of the equipment are robust and fault tolerant by their design based on a conservative approach. Redundancy, diversity and adequate operating margins are built in the system to aid the operator to continue with the operation even if some device malfunctions in the middle of a sequence and causes a halt to the normally automatic mode of operation. The alert and proficient operator takes charge of the situation at that moment and promptly brings the system to a safe state by intelligent actions. In most of the cases of single failures, the operation can be completed by using the alternative provisions or manipulation of certain process parameters within their permissible limits. The system can be normalised thereafter.

However, a lot of responsibility rests with the operator-in- charge and he is required to posses a thorough knowledge and expertise to correctly assess the situation, systematically analyse the pros and cons and take the best and safest course of action on the spot. Hence, pursuit of excellence had been an essential ingredient in the orientation and training of the operations personnel for fuel handling. Review of past experience and learning of lessons has been an ongoing process. It is being made more systematic and effective by evolution of special procedures for handling abnormal situations, development of special tools for use in emergency and new techniques made available by the information revolution. With improved man- machine interfaces, on-line diagnostics, ready access to information, proactive approach and meticulous planning of maintenance, the performance in operation is heading towards reaching higher targets of achievements.

As the fuel handling systems matured and gave dependable performance ,more tasks are being added to their portfolio. These are basically related to inspection and maintenance of coolant channels. The fuelling machines are used to support and carry the special equipment to respective channels and are also sometimes called upon to temporarily empty a channel for inspection, hold the fuel bundles for some time and reload them into the channel. Ref. 1 has several papers covering various aspects of operation and maintenance, operator performance, philosophy of training and commissioning. It also include papers presenting details of elastomers and seals specially developed for different application in fuel handling, and papers devoted to the power pulse problem.

Power Pulse Problem

Refuelling against the flow of coolant is adopted in Bruce and Darlington stations. So the downstream end of the channel contains more reactive fresh fuel while fuel bundles at the upstream end have low reactivity due to longer stay and higher burn up. A minimum gap is required to be kept between the upstream shielding plug and the edge of the first fuel bundles in the string to accommodate maximum differential thermal expansion between the coolant tube and the string of fuel bundles, under postulated LOCA conditions. The actual gap increases due to axial creep of pressure tubes and varies from channel to channel. In normal conditions the string of bundles is held against the drag force of coolant flow by a fuel latch at the downstream end. However, in the event of a postulated rupture of the reactor inlet header, the heavy water in the coolant tubes connected to it will rush out through the inlet end. This reversal of flow will drive the string of bundles towards inlet end by as much as six inches (15 cms) till the gaps are closed. In this process more reactive fuel from the downstream end will be moved towards the centre of core causing a surge in the reactivity at a time when the void in the coolant caused by the inlet header rupture has already added positive reactivity to the core. The combined effect would cause a sudden increase in reactor power, termed as `power pulse'.

The safety review done in 1993 indicated that considering this power pulse the shutdown margins in the existing reactor protective systems are insufficient to meet the current safety requirements. It was hence decided to restrict the operation of Bruce-A station at 60% of full power and Bruce-B at 80% in order to have increased safety margins. Since Darlington NGS uses a double loop of primary coolant system, effect of the postulated failure of one inlet header will affect only one quarter of the core. As the reactor has seen less number of years of operation, the elongation of channel due to creep is also considerably less so far. Considering these aspects and the increased safety margins already provided in the protective system, Darlington NGS was allowed to function at its rated power for the time being, though it will have to also address this issue in the long run.

Various solutions to the power pulse problem are described in Ref. 26 to 29.

Once the problem of `power pulse' was identified, a multidisciplinary team was constituted to determine the best alternative design, safety and operational solutions. The solutions fell into two basic categories; to reduce the size of the gap and to eliminate the insertion of positive reactivity due to fuel bundle movements.

Gap Management:

The gap available for axial movement of the fuel bundles can be reduced either by insertion of a creep compensator spacer at the downstream end or modification of the upstream shielding plug by adding an extension piece or by increasing the length of the fuel bundle string itself. All the three alternatives were pursued and new components with appropriate features were engineered and produced for trial installation in the reactors.

Creep Compensation Spacer:

A cylindrical piece of varying lengths is installed between the last fuel bundle and the latch. Its two ends are shaped to match with their respective mating components, the end plate of a fuel bundle on one side and the latch in the end fitting on the other. A flow path is given through the centre and angled elliptical holes. An installation and removal tool will be provided in one of the magazine positions of the fuelling machine. It will be advanced to open the latch, pick up the creep compensator and close the latch again on the fuel bundles. Though this is technically feasible, it adds up 20 minutes to each channel refuelling operation and occupies one magazine position. Additional time will be needed for regular replacement of the creep compensators with incrementally longer ones as the channels creep further. The estimated resultant reduction in refuelling capacity by about 24% was not tolerable for the already very busy fuelling system of Bruce A based on only two trolleys. Consequently, this solution was abandoned for all the stations.

Flow straightening Inlet Shield Plugs (FSISP):

A cylindrical piece of 10.3 cm diameter, 4 cm thick and with 121 axial flow hole is inserted in the skirt region of inlet shield plug and crimped to hold it in position. It reduces the gap by 4 cms and also straightens the flow to reduce flow turbulence and consequently, the fuel bundle fretting problem. Hence, this solution was implemented in Bruce-A to get the multiple benefits. A dedicated trolley with two fuelling machines was used for replacement of 8 shielding plugs by each fuelling machine in one lot. These 16 plugs were taken to the ancillary ports of the central service area where the inserts were shrink fitted and crimped using special remote operated tools to convert them into FSISP for installation in the next round. Shielding Plugs in all 280 inner zone channels were replaced by repeating this process. Reactor power was permitted to be raised to 70% after this modification and further to 75% after refined safety analysis.

Long Fuel Bundles:

In this scheme some fuel bundles are made longer by 1.3 cm than the regular ones. Each fuel channel is loaded with a combination of some long bundles with rest of the regular bundles thus reducing the free space in it. This solution also addresses the problem of interaction of bearing pad of the last fuel bundle with the rolled joint area of the pressure tube causing fretting. It was actually developed earlier for solving this problem and implemented in Darlington. This technique has been accepted in Bruce-B to solve the power pulse problem. A very complicated control scheme and data base management is required to track the movement of fuel bundles of two different sizes in and out of each channel, simulate its effect on reactivity of the core, estimate individual creep rates etc. to ensure that the gap in each channel is within the upper and lower limits, pick up the correct number of new fuel bundles of each size and load them into the channel in a particular order in every refuelling operation.

Fuelling with Flow:

The other alternative is to live with the gap but eliminate its ill effect. if the fuelling scheme is reversed, fresh fuel will be placed at the entrance to the channel from the upstream end and the burnt fuel will be located near the downstream end. This reversal of fuel burn up profile within a channel will result in a reduction of reactivity in the event of the postulated inlet header rupture and will be on the safer side. In view of the problems in gap management, the concept of fuelling with flow was considered to be a superior solution for Bruce `A' because it can remove the restriction on power level completely irrespective of the extent of creep or limitations of safety margins available in the reactor protective system.

Though the normal mode of refuelling in Bruce is against the flow, defuelling with flow was needed at times for certain reasons. Special toolings and control programs were hence developed to hold the latch open and allow the bundles to flow into the fuelling machines in shutdown condition. This technology was further improvised by engineering appropriate changes in the hardware, software and operational sequences to make it work for on-power refuelling.

The major hardware changes are made in the design of the shielding plug at the downstream end and the fuel carrier mechanism. Although the original fuel carrier can also carry out the operations of inserting new fuel at the upstream end it is not equipped to hold the latch open and accept the spent fuel, one pair at a time, by separating them from rest of the fuel bundle string. It is therefore modified by adding a nosepiece, a flat shoulder and inserts. The fuel bundle separation is achieved at the latch by accurately controlled operations of the rams and the carrier tube.

In the fuelling with flow concept the most irradiated bundle would be held against the latch fingers which can contact only the outer peripheral pencil end caps and not the end plate. This bundle is subjected to maximum hydraulic load. While a new fuel bundle is strong enough to be held in this manner, a highly irradiated bundle is likely to crack in the end plate region during the hot shut down condition. This was experimentally verified and cracks due to delayed hydride cracking action were observed in the end plates. Hence, it was necessary to support the bundle on a wider area on its end plate, instead of the pencil end caps to reduce stresses. This is done by adding a nose piece to the shielding plug, which will pass through the latch and contact the fuel bundle end plate. This new shielding plug is called Fuel String Support Shielding Plug or F3SP.

F3SP uses the existing outlet shielding plug design with all of its features like radiation shielding and diversion of flow to liner tubes. Only a nose piece is added at the front end. The nose piece consists of a fuel support plate having rings similar to the end plate of a fuel bundle, the body, an attachment bolt, a thrust bearing and a detente spring. The thrust bearing prevents rotation of the nose piece in relation to the fuel bundle when the rest of the shielding plug is rotated during installation and removal operations. The detente spring ensures that orientation of the nose piece relative to the shielding plug will be correct when the shielding plug is removed and reinstalled.

Sequences for operation of the fuelling machine for removal and installation of the shielding plug, opening and closing the latch and achieving proper separation of fuel bundles at the latch while receiving one pair of fuel bundles in the fuel carrier are different as compared to the original scheme of fuelling against the flow. These were very carefully formulated, and new protective interlocks added. A number of functional tests were conducted at a test channel. Certain postulated abnormal operating motions like stalling due to misorientation were also tried to verify the integrity of F3SP under such conditions. Vibrations of F3SP as well as the latch fingers in unloaded condition were studied to check that they are not detrimental to the fuel channel or the fuel bundles. Fourteen F3SP were loaded into Bruce unit 4 and the most irradiated bundles were placed in contact with them at the downstream end. These were removed after a hot shutdown and examined for freedom from cracks.

Thus after carefully studying all aspects and carrying out extensive testing in test channels as well as in the reactor, Bruce A is now planning to proceed with replacement of all existing shielding plugs at the downstream end with new fuel string support shielding plugs. Handling of the shielding plugs has been elaborately planned. All the shielding plugs will be exchanged in the fuel bay using a special shielding plug handling system installed for this purpose. These will be in modules holding 8 shielding plugs each. New F3SPs will be first placed in the module and exchanged with old plugs. The modules with irradiated old plugs will be lifted and placed in a shielded container with help of a remote operated special crane in the bay. Containers holding 14 modules each will be sent to a radioactive waste operation site for storage in concrete trenches.

All the fuel bundles in all the channels will also be reoriented in the reverse order before installation of new shielding plugs. This will be done in a cascade mode in on-power operations. Initially one seed channel will be loaded with 12 new and depleted bundles. 12 irradiated bundles taken out of it will be loaded in the next channel in the reverse order. This will be repeated for about 13-14 channels, at the end of it the new and depleted bundles will be taken out of the seed channel. Comprehensive simulation studies were conducted by fuel and physics section to predict maximum channel power, bundle power, peaking factors etc. A reorder test was conducted in Bruce-4 by on-power reshuffling of fuel bundles in 12 channels in this manner and all the important parameters were studied. The theoretical predictions were confirmed by these tests and the limits pertaining to reactor regulatory and protective systems and performance of the fuel were also confirmed.

A different approach has been taken in respect of Bruce-B station in view of its different core configuration and gap management by long fuel bundles has been selected as a solution to the power pulse problem in this station. It will be done with a comprehensive monitoring program which defines safe range of gap for each channel. As the channels continue to creep, increasing number of long bundles will be used to maintain the gap within this range.

A special Bundle-13 Verification Tool (BPV tool) has been developed to assist the process by obtaining independent measurements of the gap. This measurement will be compared with the value obtained by the computer program on the basis of knowledge of original fuel channel lengths, creep rates, history of number of regular and long bundles in each channel etc.

The BPV tool consists of a body, a fuel sensor and a spring plunger. The body is made of a modified fuel carrier to facilitate operation with the fuelling machine charge tube. The fuel sensor functions as a depth gauge probe with a sliding contact with the spring plunger bore. Its inboard end has a set of concentric rings to contact the fuel bundle end plate and distribute axial force applied during sensing. The spring plunger is operated by the ram and carries the sensor. The tool is picked up by the fuelling machine at the ancillary port and stored in the magazine. After clamping on to the target channel, the tool is inserted into it for making measurements.

Topics related to O&M are dealt with in References 30-37.

Performance of fuel handling systems are heavily dependent on the operators' skills and expertise. A proactive approach is taken at Darlington NGS to reach and maintain excellent performance [Ref. 30 and 31]. The operation of fuel handling system is fully automated and computers control it in real time by scanning and operating about 5000 I/O points. The operators' interaction with the control system is in three ways. He initiates fuelling runs by giving an appropriate `command', `monitors' the data displayed on CRT screens continuously and `logs' important points about operation and problems, independent of the automatic data acquisition by the computer. Extensive field support is given by the operators in loading of new fuel bundles in the loading trough and handling storage modules in the bay for receipt and storage of irradiated fuel bundles. Training was previously done by use of several levels of Training Manuals and Operating Manuals. However, these manuals often contained errors or lacked in information. Hence, much of the training was done in `oral' tradition based on individual beliefs held by senior operators. This process created certain `legends' independent of the station documentation and led to haphazard interpretation and execution based on implied expectations. As this was a major impediment in the pursuit of excellence, the training process has been streamlined.

The goal is to ensure safe and correct operation and a correct response to fault conditions. Excellence is achieved when this standard is defined and always met. After initial training in classroom as well as in the field and the control room and passing the qualification examination the operator works as a co-pilot under supervision of a qualified panel operator. After gaining experience and passing through an interview he is qualified to operate. Further pursuit of excellence is a continuous process and depends on how best he learns from his experience in a structured manner.

A circle model consisting of experience, expectations and standards, vigilance, technical section inputs, internal peer capability, updates of documents, motivation tools and refresher training course keeps upgrading the operator in each cycle. Each of these is given due importance by the station authorities and concerted efforts are made to continuously upgrade each aspect by learning from experience. Certain points to be noted are as follows:

1. There must be a `buy in' for expectations to be generally adopted and upheld by the operators.

2. Expectations should be expressed in a concrete form meaningful in the work environment.

3. The skill or expertise is hard to cultivate and easy to lose. It needs continuous sharpening.

4. A fuel Handling Incident Report (FHIR) data base is incorporated. Depersonalised reports of incidents that may affect fuel handling (but not classified as Significant Event) are made and circulated to only fuel handling family. Action items are generated from these FHIR and monitored till completion.

5. An internal peer audit compares standards and expectations with day to day experience on an on-going basis.

6. Training and operation documentation is periodically updated to contain current and accurate information.

7. The secret of success of a motivation tool is how it is formed. Proper emphasis should be given to this aspect.

8. Respect is a key tool of motivation. Members must respect each other, the aims of the program, the tools of implementation, the trainers and the standard providers. On their part the trainers and standard providers must respect the operators, their views and provide them with respectable tools and practically achievable standards.

9. The trainers must be responsive to the course feed back.

The commissioning and training philosophy for fuel handling at Cernavoda-1 is described in Ref. 32. Cernavoda-1 NPS in Romania consists of a CANDU-6 reactor built in collaboration with Canada, with certain equipment also supplied from Canada. Commissioning of the fuel handling system and training of Romanian fuel handling personnel was completed under the guidance of Canadian specialists. This was done following a well thought out philosophy and execution as per meticulous planning in order to achieve excellence in performance.

A group of Romanian engineers and technicians were trained at Point Lepreau in Canada and a group of experts went to Cernavoda to supervise construction and commissioning. The Romanian fuel handling staff was placed under their supervision. Initially all the members worked on preparation of procedures and documentation. The field technicians were gradually transferred to form the operations group as work in field construction and commissioning progressed towards completion stage. Since the fuelling machines had been in a state of preservation for about five years they were taken into a clean room and closely inspected for evaluation of their condition. Certain operational checks were made using rudimentary sources of oil and water supply and an electrical test box. They were found to be in good condition, ready for commissioning. However, heavy water supply pumps were in an unsatisfactory state. They were temporarily used for commissioning in the same state and subsequently rebuilt with replacement of parts damaged due to corrosion.

Verification of cable routing and wiring, hydrotest of pressure bearing systems and calibration of all instrumentation devices was done as a part of pre-commissioning. The control console and racks were the first to be commissioned. Fuelling machine bridge, carriage, cateneries and auxiliaries followed in logical order before bringing the fuelling machine heads to their position.

Commissioning of two fuelling machines and their operation at the rehearsal facility proceeded very smoothly. The control software was also installed and tested. The initial operation was in step mode followed by fully automatic run mode. New fuel handling system was also commissioned in parallel and its interface with the fuelling machine was established before draining of light water. The off-power demonstration of refuelling system was done during the hot performance test of heat transport system, for the first time in the history of CANDU reactors. Miscellaneous support systems like grappling were also tested and proven. Commissioning of spent fuel system has to wait a little for filling of the bay with sufficient quantity of demineralised water. Minor problems like pressure instability in oil systems, vibrations in water system, unreliable operation of brakes and panel bulbs and inadequate lubrication of bearings in gear boxes were identified and resolved by appropriate modifications and replacements of deficient parts. Certain other jobs like channel closure plug installation and removal, draining and drying of coolant channels , new fuel loading etc. were performed by the fuel handling unit. A unique innovation was incorporated by fixing two separate platforms, one above and another below, to the bridge to save the time required in shifting them repeatedly for reaching all the channel locations.

The fuel handling operations group consisted of personnel in three categories; personnel trained in Canada who took the lead roles, those who worked under supervision of Canadian experts in Romania took the supporting roles and a set of freshly recruited people. Training programs including qualification examinations tailored to suit these different categories have been arranged in a systematic manner and are progressing without disturbing day to day work. All the initial refuelling operations will be supervised by the expatriate Canadian staff with Romanian specialists as co-pilots. They will take over after obtaining sufficient experience and getting the authorisation to operate. Training of the field personnel has been done with a multi-disciplinary approach to expose them to gain experience in all disciplines. Maintenance personnel have also been thoroughly trained to acquire experience and skills to carry out all the routine maintenance work and also disassembly and reassembly of the complex equipment.

Ref. 33 describes improvements in the fuel handling system performance through improved elastomers and seals. Fluid Sealing Technology Unit of AECL conducts Research and Development work for advancement of technology related to elastomeric components as well as mechanical seals and has helped various CANDU stations in investigation and resolving numerous problems in seals used in various fuel handling equipment. They have now identified specific elastomeric components of superior quality for each class of service after exclusive testing in laboratories. Standardisation on their use will give advantages like longer service life, improved safety and maintenance planning, streamlining environment qualification and simplification of procurement.

There are many types of failures such as extrusion, wearout, tensile cracking, compression set etc. which are influenced by a combination of material characteristics and service conditions. Commonly quoted properties like hardness, elongation etc. do not necessarily relate to the mode of failure in service and hence relevant and comprehensive failure criteria are required to be established for meaningful life prediction for each application.

Infinite variety of formulations are possible for the compounds of the same generic nature (such as Nitrile or Neoprene) because a large number and variety of chemicals are mixed with the base polymer and the compounds are processed differently for mixing and curing . Hence, a compound - specific data base must be generated for all relevant properties. It must also include service specific behaviour, damage parameters and levels of damage considered to constitute a failure for a given application. The number of compounds to be used should be rationalised to fewest to cover the whole range of application so that each of them can be extensively tested to adequately characterise them at a reasonable cost.

The ingredients and process variables for each chosen compound must be closely controlled for consistency in qualification testing and in service. The specification of a seal must specify the particular compound and requirement of traceability of the batch. Inspection method and rejection criteria for surface and internal defects must be clearly specified. A tool employing elastodynamics has been developed for inspection of `O' ring seals for detection of internal defects and for monitoring the stage of ageing for seals in service.

Elastomeric parts must be stored in a relaxed state free from strain and in a cool, dark place free from hazardous contaminants like ozone or vapours. Their shelf life based on rational estimation of deterioration shall be specified. Proper care should be taken during handling each elastomeric product that it is not exposed to incompatible materials or subjected to mechanical damage.

The elastodynamic spot tester tool can be used to monitor degradation due to ageing. If the actual and required values of different properties like hardness, stiffness, damping etc. are measured at the start and the change in their properties due to ageing are measured at time intervals the rate of ageing and margins available before it reaches the limiting values can be determined and the residual life estimated on this basis.

A hydrostatic seal CAN 13 was developed by AECL to replace the earlier seal used in the fuelling machine rams. The earlier seal allowed a leak rate of 2 litres per minute and used to fail due to face rubbing, friction and wear caused by blocking of orifice due to dirt. It was also difficult to manufacture and inspect. The new seal was developed using modern analytical techniques and wear and erosion resistant silicon carbide for the seal face. The new seals have given excellent performance as no failure has taken place out of 64 seals used in eight reactors since 1992.

The Pickering fuel transfer conveyor is described in Ref. 34. Irradiated fuel bundles are transferred from four reactors of Pickering-A NGS to the spent fuel bay on a conveyor cart passing through a water tight square stainless steel tube. The tube is laid at a level approximately 14' (4.2m) below the bay water level and runs horizontally through a long concrete tunnel which connects two elevators located in each reactor building to the bay. This underwater system has been in operation for 25 years without maintenance, because the bay cannot be drained for creating access to the equipment.

As the tube is passing through the containments of the reactor buildings, along with the elevator housing it forms the containment boundary. So if maintenance is to be performed on the lower part of one elevator, all the openings in its upper part must be sealed leaktight and the portions of the conveyor tube connected to that elevator should also be isolated from rest of the tube, so that only that portion of the tube and the elevator to be maintained can be drained without impairing containment integrity of any reactor building.

Various alternatives for achieving this isolation were evaluated. Freezing with liquid nitrogen had been unsuccessfully tried in the past and posed an unacceptable hazard for personnel working in the confined space of the tunnel. Air inflatable bags were also unsafe and unlikely to seal corners of the square tube. Injection of hardenable thermoplastic material had the drawbacks of complexity of the process and the uncertainly about its incomplete removal during normalisation.

The concept of a bung with an expandable elastomeric seal was selected because it was the safest and feasible with use of well established technology of moulding of elastomers known to be compatible with the environment and meeting the requirement of radiation resistance. A two point isolation system consisting of two interconnected bungs having double rubber seals with an actuator and a delivery system has been finally envisaged. Lowering the bungs through the elevator was found to be impracticable. They will be fixed to the conveyor cart in the bay with underwater operation and the cart will be driven to the appropriate location for sealing the elevator to be opened for maintenance. The actuator system is still in the early stage of development.

Development of the seal is successfully completed. The cross section of the tube being square, the seal has a similar cross section with clearance all around in free condition. Wall thickness is varied from a thin section that allows preengagement to a gradual thickening to provide the bulk for expansion. When the seal is engaged with the pyramid shaped plug and rides over it axially, the seal expand radially and is pressed against the inner surface of the square making a leaktight joint. Additional material is provided at the corners for better sealing. A small groove is made at one corner to accommodate the wire rope used for pulling the conveyor cart. Different shapes were produced by making gradual modifications in the mould.

A polyurethane compound was chosen because it is tough, strong, tear resistant, has adequate radiation resistance and is suitable for the temperature range of the application. Hardness is kept low to keep the force required for expansion within reasonable value. Samples were made and tested by placing in a square tube, expanding with a screw mechanism and measuring leakage flow rates at differential pressures of 6.5 and 12.5 psi. A transparent square tube was used to give a visual assessment of how the seal contacts the tube when expanded. The cross section of the seal was gradually modified by study of the observation. After a few iterations the final shape was able to meet the leakage rate requirements.

The operation and maintenance of the SLAR system of Bruce-A is described in Ref. 35. The Spacer Location And Repositioning (SLAR) system is used to find out actual location of the spacers (garter springs) between the calandria tube and the pressure tube and if found displaced from the original position, to reposition them correctly so that a contact between the two tubes is avoided. The contact is detrimental because it can lead to formation of a hydride blister and consequent failure of the pressure tube.

In order to do the SLAR operation faster and without radiation exposure, it is done by specially developed remote controlled equipment. The pressure tube is required to be emptied to facilitate insertion of the sensors for spacer location and the fuel bundles are put back into the channel to avoid unnecessary wastage. Thus a SLAR operation also includes the fuel discharge and loading operations. Hence, a SLAR machine is required to be more complex to perform all the operations of the fuelling machine in addition to its main function of sensing and repositioning of the spacers.

A fuelling machine is modified for building a SLAR Delivery Machine. The snout and magazine portions in the front are retained and an indexing turret is placed in place of a flat head at the rear. The existing charge tube and ram assembly, a fuel tube and a SLAR Module tube are mounted on the turret. Any one of them can be aligned with the snout by rotation of the turret. Two heavy water hydraulic power supply units and seven DC Servo drives are added to cater for additional operations. Since spare cables are not available, multiplexing is done for supplying power to the motors. The SLAR tool consists of a hydraulic tool to raise the pressure tube, eddy current systems for locating the sensors and gap measurements, ultrasonic probes for detection of cracked blisters and linear induction motors to reposition the spacers. A long umbilical cable connected to the tool to carry power, control and inspection signals is wrapped around the cable drum. The operations of SLAR tool as well as the delivery machine are controlled by computers. The SLAR operation is done in the reactor in shut down state and PHT at a low pressure.

A total of 449 SLAR operations have been conducted in Bruce-A units 3 and 4 since 1993. Certain problems encountered in this experience are as follows:

1. On three occasions channel temperature went up from 30° C to 60° C, possibly due to stagnated flow. Flow switches and temperature monitoring will be added to the system.

2. Pressure Regulating Valve settings were getting affected due to crud entering the system from the end fitting. Problem reduced after installation of a filter in the heavy water System.

3. Amplifiers and optocouplers of original servomotors failed on numerous occasions and could not be repaired due to obsolescence. These were replaced with standard off-the-shelf amplifiers with minor modifications in the circuit.

4. The cable drum drive motor has a complicated control to satisfy very difficult requirements and it failed repeatedly. The motors were replaced with higher capacity units having conventional copper wound armatures and a higher current rating to solve the problem.

5. Multiplexing system is used to enable the additional drives to share the existing cables alongwith the existing drives. However, there used to be a mismatch between transfer of the power and motor connections. Timers with delays were introduced to solve this problem.

6. As the SLAR Delivery machine is larger than the normal fuelling machine, more space is required to be created by removal of rails, catwalk and air conditioning units in the vault.

7. The SLAR tool being very complex needed frequent replacements, which was quite a cumbersome operation to be done in an awkward and limited space. A number of problems were faced during the tool changes. The umbilical cable got wrapped around the tether cable and got cut. Poor connections made at small fragile connectors located in a congested space led to erratic signals from eddy current and ultrasonic probes. Stainless steel sleeve and potted cable are very tight fit inside the module tube making their connection very difficult.

8. The sag/droop developed in the Module tube and Fuel tube when left in the extended positions made their operation sticky.

It was not possible to use SLAR in Bruce A unit 1 because of limitations of movement of the trolley in the tunnel. Hence a mini SLAR was developed as described in Ref. 18. Since this mini SLAR did not have fuel handling capability, all 109 channels to be SLARed were defuelled by regular fuelling machines but using flow assisted technique in the reactor in shut down state. SLARing was done by using the mini SLAR machine. Replacement of fuel was accomplished by transferring fuel from unit 2 which was laid up and defuelled. Thus a substantial fuel cost was saved.

A used Fuel Dry Storage Facility (UFDSF) has been recently commissioned at Pickering NGS [Ref. 36]. This facility has a capability to store about 576,000 irradiated fuel bundles placed in 1500 Dry Storage Containers (DSC) by the end of the station operating life. The DSCs are designed to offer interim dry storage for upto 50 years and a means of direct transportation to its final disposal site without repackaging. The facility has been licensed by AECB and safeguarded by IAEA.

New DSCs are carefully inspected for freedom from corrosion and proper fitment of all parts including empty fuel modules. Spent fuel bundles targeted to be loaded are also inspected through a telescope and their identification numbers are recorded. The DSC was filled with demineralised water and lowered on the impact pads in the Auxiliary Irradiated Fuel Bay (AIFB). Four modules containing 366 irradiated fuel bundles about 17 years old were loaded in the DSC under surveillance of IAEA and AECB staff. Lid was placed in position and clamped and the DSC was lifted out of water and washed in the decontamination pit.

Radiation fields were about 0.5 mR/hr and smears showed zero surface contamination, reflecting cleanliness of bay water and effective washing technique. The water inside the DSC was drained by gravity. Tilting in two directions improves draining. It was dried with vacuum system for 14 hours and the internal pressure in the cavity was maintained below atmospheric.

DSC was thereafter transferred to UFDSF workshop. It was again inspected for surface contamination and radiation. Leftover water (about 20 litres) in the DSC was drained out and the cavity was vacuum dried for 4 days.

The DSC cover is welded on its top flange using automatic welding heads. The flange is preheated for 16 hours to get it at a temperature of 105° C. The welding of 1.1/2" groove was completed in 11 passes. Frequent visual inspection was done to ensure a flawless weld. After the weld was cooled for 2 days it was radiographed by X-ray machine. DSC was kept connected to a vacuum system to maintain low internal pressure through out. Final vacuum drying to achieve 1 m bar (a) took less than 2 hours. DSC was backfilled with Helium to 930 m bar (a). Drain and vent ports were closed by welding. The DSC was then placed in a bell jar and Helium leak tested. Thereafter the freshly welded areas were painted and DSC was placed at its designated place with a formal ceremony.

Detailed documentation showing full history of fuel bundles stored as well as all inspections and tests conducted on the DSC will be compiled and filed for preservation.

The radiation shielding provided by the DSC proved to be better than anticipated, as the gamma fields on contact (0.6mR/hr), at 1 metre distance (negligible) and the storage building wall (33 micro rem/hr) were all below the target values. Dose taken by the operation and maintenance crew was also only 20 mR against an estimation of 75 mR. Air emissions from UFDSF are found to be zero and there had been no active liquid discharge.

Initially four DSCs were transferred in two months. Full capacity is expected to be 11 in two months and requirement is 66 per year for all 8 stations operating at 85% capacity factor.

Major difficulties encountered and the corrective actions required to be taken are as follows:

1. Gravity draining was inadequate. Tilting in two directions is required. Lifting beam needs a design change to facilitate it.

2. Welding procedure for plugging of port and drain holes needs modification to speed it up.

3. Welding of the lid to the top flange of DSC Body showed several defects needing rework. It is being investigated.

4. Helium leak detection takes too long time, procedure to be reviewed and modified.

5. Lack of manoeuvring space in the bay causes time delay in all bay operations. These are to be simplified and streamlined.

Different techniques and facilities used for decontamination of fuel handling equipment are described in Ref. 37. The important techniques are as follows:

1. Manual mechanical cleaning with a wire brush or scouring pad

2. Chemical cleaning with strong or weak solvents, detergents etc.

3. High pressure water jets.

4. Vibratory Finishing - combination of mechanical and chemical actions.

5. Ultrasonic cleaning - combination of chemical action with physical action of cavitation formed by tiny collapsed bubbles.

6. Dry ice blasting: Cold cracking of contaminant layer and further loosening due to sublimation and expanding gas under the layer.

A thorough assessment of facilities and technique that can be best for use in Fuel Handling Maintenance shop was done and the following actions were taken. A set of performance criteria was developed and used as a guide line. Procedures for carrying out decontamination work in a systematic and scientific manner to achieve the maximum amount of decontamination with minimum exposure to personnel were made. Water jet was found to be the most effective and safe method for intricate fuelling machine components. A Ram spray booth was custom designed and constructed. It consists of a large tank with ventilation & drain connections, viewing windows, glove boxes etc. An automatic water lancing machine using a two nozzle rotary spray jetting tool moves the spray arms longitudinally and radially to provide spray coverage to all areas. A manual hand held jet is also available. The machine is controlled by PLC and interlocks are provided to avoid injuries. Other facilities like ultrasonic, vibratory finishing, ventilated bench, ventilated sinks are also provided in the shop. Reduction in dose by a factor or five and improvements in productivity, reliability and performance have been achieved.

7. VISION FOR FUTURE

Reactors like CANDU-9 or TAPP-3 and 4 (Indian 500 MWe PHWR) may be constructed in the near future as soon as finance becomes available. Design of fuel handling systems for these reactors has been already substantially completed. Hence, it has been covered under section `Design'. Innovative reactor concepts which still require considerable design and development are considered in this section.

Neutron economy in a PHWR has been one of the prime considerations for PHWR and it can work with the smallest amount of excess reactivity because of on-power refuelling. The PHWR has the basic flexibility and capability to use a wide variety of fuels like natural uranium, slightly enriched uranium, mixed oxide (MOX) and even thorium which is abundantly available in nature.

The future vision for fuel handling systems naturally depends upon the type of reactors and fuels to be used and a number of physics and engineering problems are required to be solved before reaching the commercial stage. However, with the research work in progress in that direction and more thrust expected to be given in the future because of economic motivation, some of these ideas may soon be converted into practical designs.

References 38 and 39 give overviews of improved fuels for PHWRs and CANDU fuel handling systems for such fuel options respectively. This subject can be dealt at two different levels; using advanced fuels in the existing reactors or new reactors to be specially built to use these fuels. Use of plutonium mixed with natural uranium or use of slightly enriched uranium either from an enrichment plant or from reprocessed light water reactor fuel form the first category. Reactors like highly advanced CANDU using a higher level of enrichment or actinide burners form the second category.

Research and development work is actively pursued for the first category. It has been known that if a fuel with a slight enrichment of the order of 0.9 to 1.2% is used in a PHWR much higher exit burn-ups up to 3 times the present levels can be achieved. Thus the uranium mined can be better utilised, fuel cost per unit of energy will be lowered and there would also be a substantial reduction in the volume of radioactive waste to be handled, further adding to the savings. If the heat removal capacity of the station can be augmented, reactor power can also be increased in this concept without increasing the peak bundle and channel powers. Thus the fuel bundles similar in construction to the natural uranium bundles presently used can be fabricated and used in the existing core to get higher power.

Countries having both heavy water and light water reactors can obtain a synergy in their fuel utilisation. Enriched uranium recovered by reprocessing spent fuel form light water reactor can be directly recycled in a heavy water reactor. This scheme has a potential of extracting almost double the amount of energy as compared to reenrichment for reuse in LWR. Plutonium separated from this spent fuel can also be used effectively in HWR in the form of MOX fuel. A dry process (DUPIC) is another attractive option in which the spent fuel from light water reactors is directly converted into fuel for HWR. In this process only the fission products are removed and the fissile materials are not separated. This process will also find much more public acceptability because it is non-proliferative. The above schemes can be implemented without making major changes in the existing reactors. An advanced design of fuel bundle CANFLEX has been developed by AECL for more efficient use of advanced fuels and increasing operating and safety margins of natural uranium fuel.

The options for building new reactors with the advanced fuel cycle offer a wide range of possibilities. Slightly enriched uranium fuel can be used to alter core physics to increase the degree of passive safety. Reactors dedicated to burn transuranium actinides along with usual fissile materials can reduce the volume of radioactive waste having a very long half life by converting them into elements having shorter half lives and also obtain some additional amount of energy from them. This serves a double objective of conservation of fuel as well as reduction of waste.

Different fuel cycles for use of thorium have been envisaged. A `Once Through Thorium' (OTT) cycle utilises thorium as fuel (and conserves uranium to that extent), a `Topped Thorium Cycle' (TTC) can obtain higher burn up and consequently save more fuel and a Self Sufficient Equilibrium Thorium cycle (SSET) is almost like a breeder as it produces as much fissile Uranium 233 as it consumes. However, all thorium cycles need substantial amounts of plutonium or enriched uranium to start with and will take decades of operation and reprocessing to reach the levels of benefits theoretically thought to be obtainable. Reprocessing of irradiated thorium and fabrication of fuel bundles made of a mix of thorium and different isotopes of uranium etc. are still at the laboratory scale and need a lot of development work to reach commercial production levels.

Use of any advanced fuel will affect the fuel handling because of fundamental differences in the properties and behaviour of different fuel materials. New fuel carrying plutonium requires extremely careful handling to prevent any damage to the clad resulting in exposing the pellets to the environment. All advanced fuel containing enriched uranium or plutonium have a potential of forming a critical mass if they are lumped together in certain quantities. Hence, the new fuel handling and storage also needs to be dealt with extreme care and provided with foolproof safety features to prevent unintended criticality. Considerations and concerns regarding physical protection and international safeguards for these fuels are of a much higher order as compared to the natural uranium fuel.

With alternative fuel options the number of bundles to be handled in each refuelling operation will be different than the presently established options. Though the total number of bundles inserted in the core and removed from it per unit energy produced will be less because of higher exit burn up obtainable from advance fuels, they will be required to be reshuffled frequently to obtain the maximum benefit. For example, the highly advanced CANDU (HAC) using about 3.2% enriched uranium will consume only ten bundles per effective full power day, but will require ten single bundle refuelling operations per day. In case of Actinide Burners the actinide bearing fuels have to pass through the reactor very quickly so that new actinides produced by the conventional fissile content of the fuel are not more than the actinides used up by burning. The refuelling rates as high as 20 to 80 visits per full power day are being estimated for these reactors.

Another major difference is that the decay heat produced by spent fuel bundles removed from the core after a high burn up will be more than the present fuel. Hence the cooling requirements for spent fuel bundles while in the fuelling machine and fuel transfer system are required to be re-evaluated and augmented. It is observed that decay heat per bundle as it comes out of the core may not be very significantly higher than at present, but its decay will be relatively slower, requiring higher rates of cooling for longer time periods while in storage.

As the percentage of enrichment is increased the diameter of fuel rod is required to be reduced and more number of rods used in each fuel bundle to limit the maximum temperature at the centre of the fuel rod. Thus the advanced fuel will be having a large number of the order of 50-60 rods. It will shift the pitch circle of the outer most ring outwards leaving smaller space for the separator stops to enter between two adjacent fuel bundles. It will also decrease the bearing area available for the separator stop to hold the bundles against hydraulic drag force.

Various solutions are also suggested for meeting these challenges. New fuel handling requirements for enriched fuels already in force in light water and fast breeder reactors can be easily extended to HWRs. Simplifications of fuelling operations and increasing the capacity of the fuelling machine magazines are considered for increasing their output rates. Ideas such as more than one machine operating simultaneously on the same rector are under consideration. Single ended fuelling scheme conceived for CANDU-3 can be extended to provide multiple fuelling machines independently operating in different zones and from both sides of the reactor at the same time. It will certainly involve very complex control system and reactor core simulation capability. The problem of separator may be addressed by providing a third separator assembly.

The fuel handling community has been preparing to take up new challenges expected to be posed by the advances in the fuel cycles which have been described. Improvements in operation and maintenance, especially in the areas of man-Rem reduction and increasing the availability , simplification, standardisation and cost reduction are the thrust areas in the immediate future.

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