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The fuel changing operation is based on the combined use of two remotely controlled fuelling machines operatir channel. New fuel t machine, are inse

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direction of the loading depends upon the direction of the coolant flow in the fuel channel being fuelled, which alternates from channel to channel. The fuelling machines receive new fuel while connected to the new fuel port and discharge irradiated fuel while connected to the discharge port. The entire operation is directed from the control room through the preprogrammed computerized system. The control system provides a printed log of all operations and permits the manual intervention by the operator. New fuel is received in the new fuel storage room in the service building. This storage room can accommodate up to six months' fuel inventory and can store temporarily all the fuel required for the initial fuel loading. When required, the fuel bundles are transferred to the new fuel transfer room in the reactor building. The fuel bundles are identified and loaded manually into the magazines of the two new fuel ports. Transfer of the new fuel bundles into a

Proceedings

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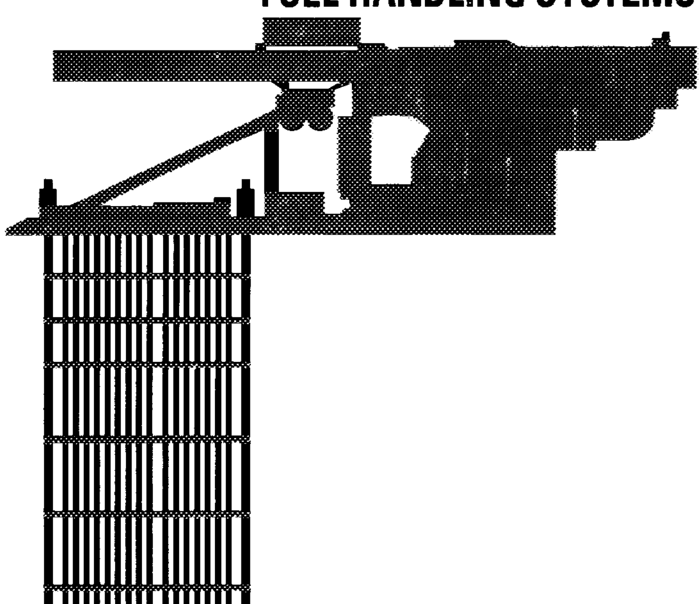
TORONTO, CANADA

INTERNATIONAL CONFERENCE ON CANDU FUEL HANDLING SYSTEMS



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FOREWARD

These proceedings record the information presented at the First CNS International Conference on CANDU Fuel Handling Systems held May 13-14, 1996 in Toronto, Canada.

The papers for these proceedings were prepared according to guidelines supplied by the Canadian Nuclear Society and are generally published as submitted by the authors. Responsibility for the content of each paper rests solely with the author.

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AN INTRODUCTION TO THE TECHNICAL PROGRAM

by

Bill Knowles
Technical Committee Chair

The analogy that fuel handling is the "Heart and Soul of CANDU" is apt in that it is the mechanism that pumps the life blood, fuel, through the body of the reactor to sustain its vitality. Should any of the major functions of the fuel handling system fail then the life of the reactor would soon wane. This conference gives some insight into the health and integrity of fuel handling

Fuel handling for CANDU systems, both in Canada and abroad, has contributed significantly to high reactor capacity factors while having minimal effect on any forced outage rates. Building on this success story are the additional tasks that the fuel handling systems have been asked to play, particularly with respect to reactor inspection and remedial work. We have several papers that address the versatility of the system in this respect.

Papers on the more traditional aspects of fuel handling design discuss instrumentation and control, structural analysis and several future equipment designs. Also in the future, the effects of new fuel cycles and fuel designs on the fuel handling system are considered.

Operational considerations are covered by papers on commissioning, training and in-service repair; while the last part of the in-station fuel cycle is addressed by several papers on dry storage.

The welcome presence of presenters from India and Romania give our conference a truly international flavour. This affords us a rare opportunity to see how the Indian fuel handling development for their PHWR compares to the two basic Canadian designs.

I hope these papers and the poster displays excite your interest and that after the presentations and discussions with your conferees, you will leave the conference with greater knowledge and the seeds of new ideas.

Welcome from the Conference Chair

RONALD A. MANSFIELD
Peng Ceng FIMechE
former AECL Fuel Handling Consultant (Retired)

On behalf of the Canadian Nuclear Society, I have much pleasure in welcoming you to the First CNS International Conference on CANDU* Fuel Handling Systems. The theme of our conference is "Fuel Handling Systems: The Heart and Soul of CANDU - Implementation, Operating Experience and Future Designs."

The high capacity factors achieved by CANDU units is largely dependent upon maintaining the successful performance of the on-power fuelling systems of these reactors. The evolution, over 34 years, of CANDU Fuel Handling Systems has involved all disciplines including design, construction, commissioning, operations and maintenance. This conference brings these disciplines together to share experience and knowledge gained, to improve the performance of operating fuel handling systems, and to advance the future designs.

The four sessions of this conference cover a potpourri of fuel handling topics of interest to all disciplines. It is possible for attendees to listen to all the lectures being presented to gain maximum benefit from the conference. In addition, a poster session and display can be visited during the reception on Monday evening and in conference breaks. You are encouraged to interact with your peers as much as possible at this conference, and learn from their invaluable experience.

I thank you for participating in the First CANDU Fuel Handling Systems conference and wish the team success in keeping the "Heart and Soul of CANDU" in good order for the future.

RONALD A. MANSFIELD

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EVOLUTION OF THE DARLINGTON NGS FUEL HANDLING COMPUTER SYSTEMS

by

CA9700718

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ABSTRACT

The ability to improve the capabilities and reliability of digital control systems in nuclear power stations to meet changing plant and personnel requirements is a formidable challenge. Many of these systems have high quality assurance standards that must be met to ensure adequate nuclear safety. Also many of these systems contain obsolete hardware along with software that is not easily transported to newer technology computer equipment. Combining modern technology upgrades into a system of obsolete hardware components is not an easy task. Lastly, as users become more accustomed to using modern technology computer systems in other areas of the station (eg: information systems), their expectations of the capabilities of the plant systems increase.

This paper will present three areas of the Darlington NGS Fuel Handling computer system that have been or are in the process of being upgraded to current technology components within the framework of an existing Fuel Handling control system.

1. INTRODUCTION

The Darlington Nuclear Generating Station is a four unit station with a fuel handling system consisting of the following primary components:

- Three independent transport trolleys. Each trolley contains two fuelling machines for reactor fuelling and maintenance operations. Each trolley is capable of fuelling any of the four reactors.
- Four Fuelling Facilities Auxiliary Areas (FFAAs) each containing new fuel and irradiated fuel mechanisms to load and unload the fuelling machines.

The F/H control system that controls this equipment consists of three independent control systems connected together in a network to allow electronic communication among the three systems (Figure 1).

Each control system in-turn consists of three computers which work together to provide all the required functionally for fuelling. The south control system contains an additional computer to control a second set of new fuel and irradiated fuel mechanisms.

When the initial control system software design was completed and placed into service, the computers were overloaded causing excessively display update times and false stops to the fuelling programs. This problem was successfully resolved by a combination of software efficiency changes and the installation of faster processors that were capable of running the existing software.

Subsequently, station requirements mandated that the Fuel Handling system be able to handle both standard length and long fuel bundles to control pressure tube fretting. This additional requirement has forced the implementation of a disk drive upgrade to accommodate the larger software.

Lastly, the additional reporting requirements of the long fuel bundle management program has necessitated the installation of an electronic communication link between the Fuel Handling control systems and the Fuel and Physics LAN to provide improved fuel tracking capabilities by the Fuel and Physics department.

2. Processor Upgrade

The justification for the replacement of PDP 11/70 processor started on June 8, 1992. The project was approved on December 31,1992. Evaluation tests started on February, 1993 were conducted in GE Lab System and in DNGS site.

QED95D processor, made by Quickware Engineering & Design Inc. was chosen. Review & specify QA, documentation & hardware tests requirements for QED95D started in June,1993. Review quotation & finalize hardware, cost & performance evaluation was done in July,1993. An order of 21 sets of QED95D processor was placed in February,1994. Unit testing for all 21 sets of QED95D processors in the GE Lab system was done in October, 1994. The installation of the QED95D processors in the F/H systems started in November 2, 1994 and was completed installations in April 5, 1995.

QED95D processor is a DEC Unibus compatible processor that is fully compatible with all PDP-11 Unibus hardware and software. It addresses four megabytes of memory and implements the complete PDP-11 architecture, excluding the Commercial Instruction Set.

The QED95D consists of two processor boards, a memory board, a massbus adapter board and a battery backup & Unibus connector board. These five processor boards replaced 22 boards in the PDP-11/70 CPU chassis, eliminated the PDP-11/70 memory chassis, and three PDP-11/70 battery backup units. Power consumption of the CPU system dropped from 40A to 13.7A. QED95D processor has built-in self-check diagnostics, which are initiated when the CPU is powered up and started.

QED95D is a high-speed CMOS processor with a 32 KB cache memory, it has a major cycle time of 110 nanoseconds (ns), allowing the execution of up to 9 MIPS. By comparison, the PDP-11/70 only runs at 1.36 MIPS.

The new upgraded processors in each trolley system scan and operate 4700 - 5000 I/O points, receive input from operator work stations, drive Ramtek Display Generators. They execute various application programs which operate the process equipment and perform the man-machine interface.

The PDP-11/70's took up to 8 seconds to update CRT displays. With the QED95D which is 4 to 5 times faster, the update time for the CRT displays now is less than 1 second.

The installation of the QED95D was straight forward and took about 2 hours for each computer. The total project cost is $910,000.00.

3. Disk Drive Upgrade

A replacement disk drive is required for the existing DEC PDP-11 RL11/RL02 disk sub-system to provide a modern high capacity sub-system. This will increase the executable and data storage size and give a more reliable and more easily maintained sub-system. This replacement sub-system must be fully compatible with the DEC PDP-11 UNIBUS hardware.

A review of the hardware options for RL02 disk sub-system replacement indicates that a standard UNIBUS / SCSI interface would be the preferred choice so that any standard SCSI storage device could be used now or in the future and give device independence.

Three replacement disk drives are required. Two of the units must be identical fixed and each to be divided into two 300M bytes or more partitions. The third unit must be a removable media unit.

The storage must be non-volatile. The read / write time must be less than or equal to the RL02 disk drive time (55m sec, transfer rate 512K bytes / sec)

The chosen SCSI interface is the CMD 720/M card , made by CMD Technologies due to :
- Low cost.
- Good reliability statistics & warranty.
- Excellent support from CMD Technologies.
- Good operating experience with the CMD SCSI cards over the past couple years.

The final selection of the disk drive replacement is as follows:

- Two Seagate 1.0 Gb winchester type units to be the on-line units in each F/H computer system.

- One Syquest 540Mb removal harddisk cartridge style unit to be the off-line unit in each F/H computer system.

These disk drive selections were made base on:
- High reliability statistics.
- Long warranty periods.
- Competitive pricing.
- Good trade publication reviews.

The total quantities of the disk drive replacement for the F/H computer systems as follows:

- 23 CMD 720/M SCSI host adapters.
- 45 Seagate 1.0 Gb winchester units.
- 23 Syquest 540Mb removal harddisk units.

The preliminary cost estimates are:

- Hardware: 300 K$
- Software: 350 K$

The Darlington Fuel Handling computer hardware configuration is shown in Figure 2.

4. Fuel Handling LAN

A number of user requirements have been identified that require the interconnection of the Fuel Handling control computers to the Darlington Information System LAN and to the Fuel and Physics LAN. The requirements include the ability to:

- automatically transfer long fuel bundle transaction logs from the Fuel Handling control computers to the Fuel and Physics fuel accounting system.
- automatically transfer the fueling activity computer logs from the Fuel Handling computers to electronic storage media for long term archive purposes.
- automatically transfer the fueling activity computer logs from the Fuel Handling computers to the Information System LAN for use in technical surveillance and support activities.
- automatically transfer the job, sequence, and operation data from the Fuel Handling computers to the Information System LAN for use in configuration management and technical support activities.

The current control system interconnects the various Fuel Handling control computers through the use of 2400 baud serial data links. These links are currently running near capacity transferring control data among the various control computers. These links do not have the capacity or speed necessary to meet these new requirements.

An alternative solution was identified by General Electric Canada after searching the Internet for potential solutions. It was identified that a mature product with a large customer base was using the TCP/IP communication protocol. This protocol is commonly used to interconnect various computer types using high speed ethernet data links and to interconnect various computer networks.

The product that has been chosen is called TCPware or RSX by Process Software Corporation. It is designed to run on Digital Equipment Corporation (DEC) PDP-11 computers running the DEC RSX11-M V4.0 operating system software. This product provides FTP file transfer tools, TELNET terminal sessions, and provides facilities to allow PDP-11 application programs to interface to the ethernet.

While the Fuel Handling computers are running a modified version of the DEC RSX11-M V3.2 operating system, the TCPware product has been successfully implemented on that platform at GE Canada in preparation for installation at Darlington in mid-1997. However, RSX11-M V3.2 cannot support the TELNET server software of TCPware.

Each Fuel Handling computer will be equipped with a DEC DELUA ethernet card though which TCPware will interface to the ethernet. The various Fuel Handling computers will then be interconnected via hubs and fiber optics cables to a single Sun Microsystem SPARC20 workstation. The workstation will act as the gateway between the Fuel Handling control system and the outside world (Fuel and Physics LAN and the Information System LAN is shown in Figure 3).

This configuration of the ethernet cabling does not allow for complete independence of the three Fuel Handling control systems. As such, a single failure may impact on more than one control system. This is considered acceptable as the information that will be carried on these data links is not required to operate the Fuel Handling equipment. In addition, this configuration does not connect to the FFAA computers. However, if this network proves highly reliable, then control data may be carried on these links once the FFAA computers have been interconnected and the cabling arrangement has been adjusted to provide functional independence of the three control systems.

The workstation will be equipped with a router between the workstation and each of the Fuel and Physics LAN and the Information Systems LAN. These routers, in combination with the workstation, will provide security for the Fuel Handling system from computer users outside the Fuel Handling system. Initially, the system will be configured to block all requests from computers outside of Fuel Handling. All communication will be one way; from Fuel Handling to the outside world.

The workstation will be set up to automatically transfer various files from the Fuel Handling control computers each day and send them to the Fuel and Physics LAN and the DNGD Information Systems LAN.

To meet the stated requirements the system will function as follows:

- Once each day the workstation will transfer the long bundle fueling log file from each of the control computers and send it to the Fuel and Physics LAN.
- Once each day the workstation will transfer the fueling activity log data file from each control computer. It will store it on large volume media for long term storage and also send a copy to the Information System LAN for user access.
- Periodically transfer the jobs, sequences, and operation data files from the control computers and send them to the Information System LAN for user access.

Once satisfactory performance of this system has been obtained, other potential uses of this link between the Fuel Handling control computers and the other site LANs include:

- Electronically sending the Fuel Change Orders from the Fuel and Physics department to the Fuel Handling control computers for execution.
- Modification of the jobs, sequences, and operations on the Information System LAN and uploading these changes into the Fuel Handling control computers. At present all changes must be entered using the control computer software.
- Transferring plant I/O status data from the Fuel Handling control computers to the Information System LAN for use in technical support and surveillance activities.

FIGURE 1

DARLINGTON FUEL HANDLING CONTROL SYSTEM

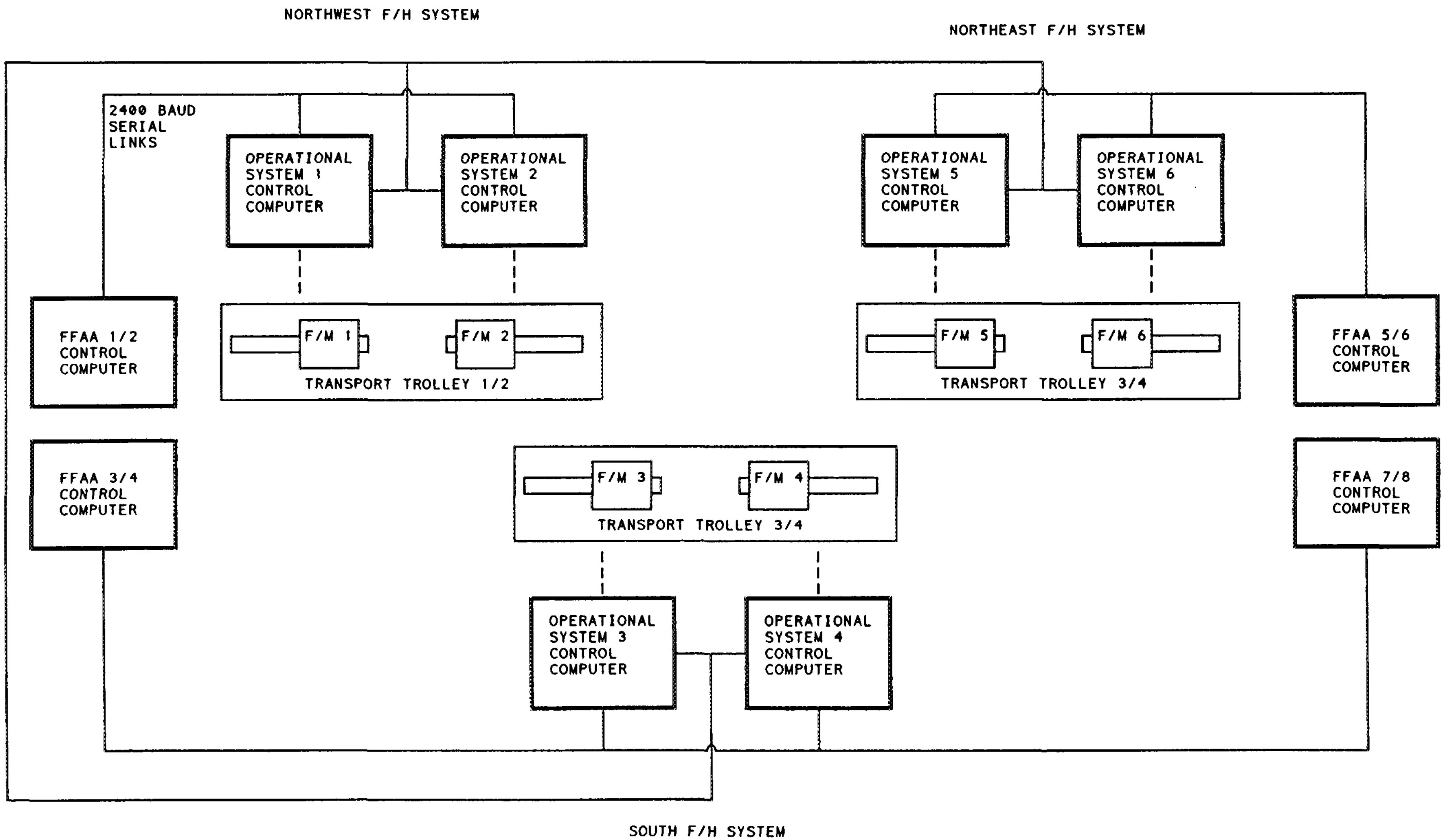


FIGURE 2

DARLINGTON FUEL HANDLING COMPUTER HARDWARE CONFIGURATION

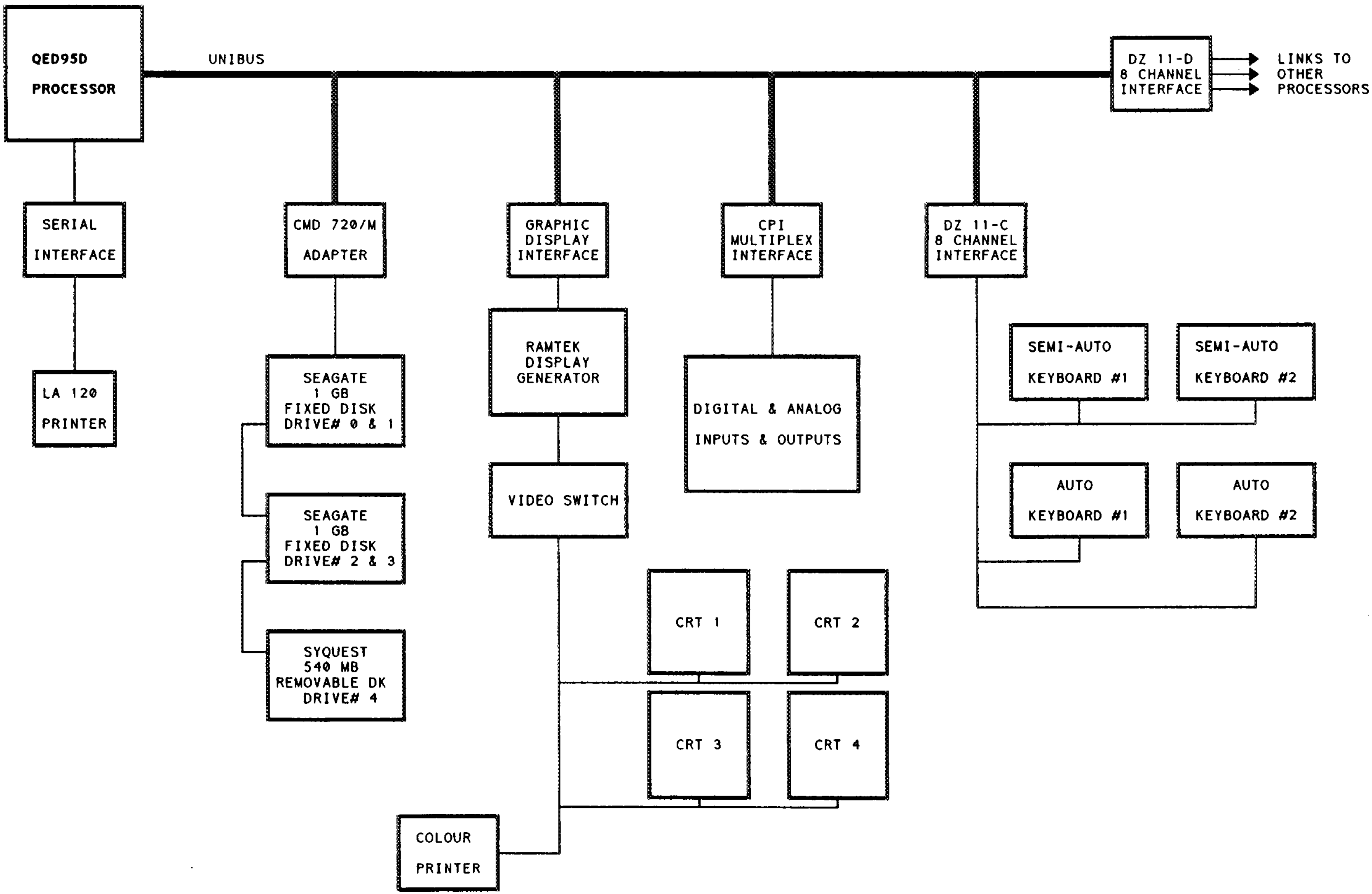
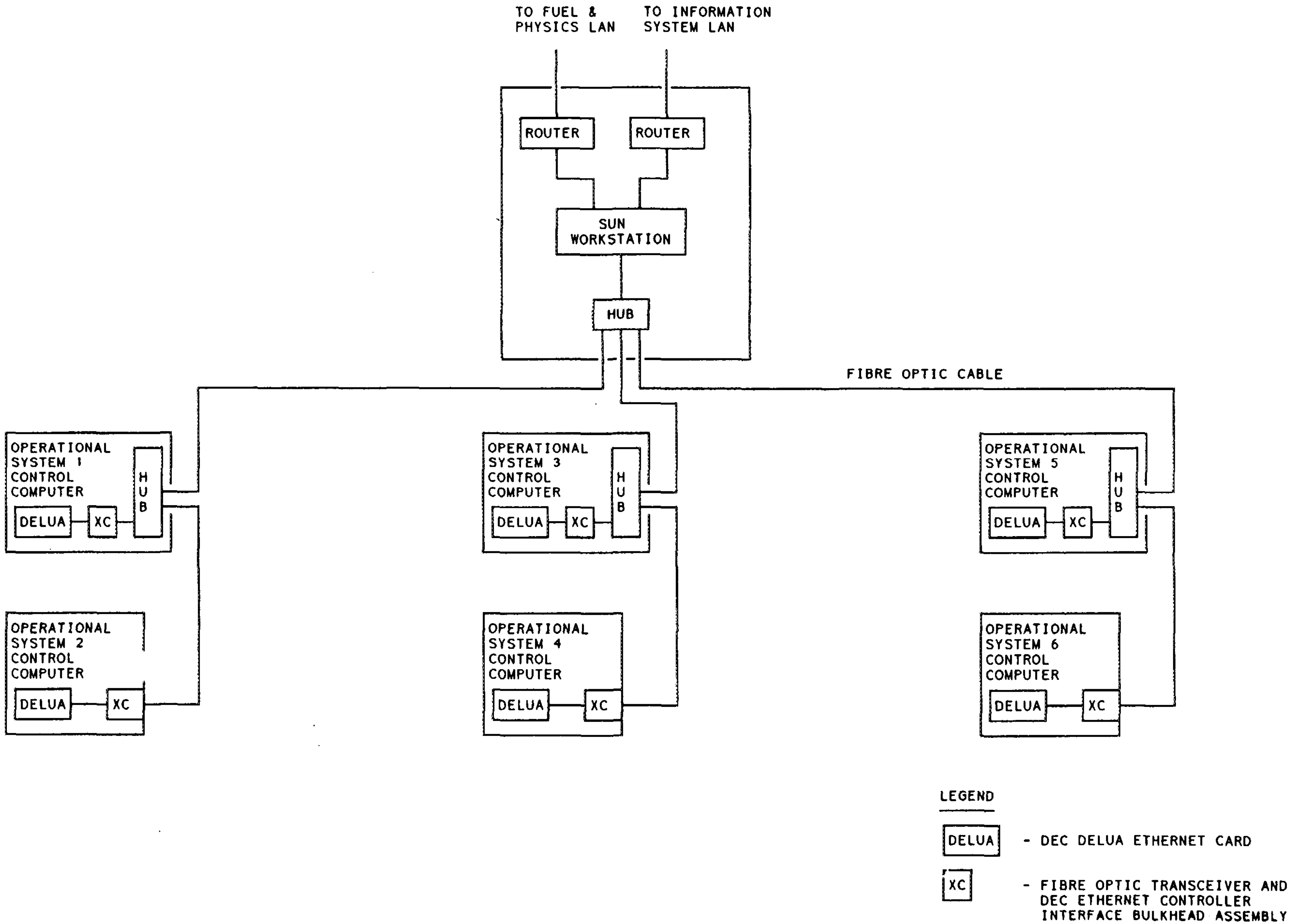


FIGURE 3

DARLINGTON FUEL HANDLING LAN



FIRST CNS INTERNATIONAL CONFERENCE ON CANDU FUEL HANDLING SYSTEMS, MAY 1996

APPLICATION OF CONFIGURABLE LOGIC IN NUCLEAR FUEL HANDLING

by

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CA9700719

ABSTRACT

Control and protective systems operating in the older nuclear power stations are nearing the end of their reliable operating life. These systems are still subject to frequent logic changes. Testing the software logic changes is becoming a significant task with ever greater expense. The software based systems can be replaced with systems using configurable logic. These systems provide new, more reliable technology, offer the capability for change, and provide capability for complete logic simulation and test before installation. There is a base of operating experience with these devices and many potential applications where they can be used to advantage.

DEALING WITH CHANGE

The fuelling systems at the Bruce and Darlington stations are ever-changing. The systems at Bruce A have been in operation since 1977, carrying out routine automatic on-line fuelling. The design had evolved up to that time from the initial concepts carried through the design and manufacturing stages to actual operation of the systems in lab testing at GE. The transition from concept to smooth operation required many changes to the operating and protective logic. Many changes were made during the site commissioning phase, as system parameters were fine tuned to cope with actual site parameters. During station operation, new modes of operation and the addition of maintenance and inspection functions have resulted in numerous changes to the logic. The Bruce A protective logic is currently running at software version 35, and the count started some years after the commissioning was completed. The Bruce B protective system, theoretically a duplication of the A system, is running version 50. The Darlington protective system, on line at the station since 1990, is running version 23. The certainty in these systems is that changes, input and output circuits and the logic to process them, will be required and must be accommodated.

The other certainty that we are faced with is that system changes become more expensive to implement as time passes. The requirements for a more thorough software design process have become more stringent in recent years. Design changes are implemented by personnel who are not as knowledgeable about the system as those who carried out the initial design. More extensive design reviews and independent testing have been added to the process to compensate. Shoehorning code into the older computers with limited memory capacity requires reorganizing program and data storage locations; this changes the program flow. This creates more uncertainty in adequacy of test coverage when modifications are made.

The original system hardware is becoming difficult to maintain. Is it now time to revisit the control and protective systems because of the equipment obsolescence? Is it time to overcome some of their shortcomings, including reducing costs for hardware maintenance and for implementing changes? Would a different approach offer the opportunity to more fully test design changes and reduce the site commissioning effort when changes are made? From the point of view of the technology available, the answers are 'Yes'.

For example: A hardware failure of the protective system at Bruce A recently caused a breach of station containment. The protective equipment is over 25 years old, possibly approaching the limit of its reliable operating life. Replacement parts are no longer available. Other protective features are designed as backup to prevent incidents, but reliability of operation becomes a concern.

Another example: The Bruce A I/O system has gone through periods of high unreliability, also at the end of its reliable operational life. Equipment outages put greater demands on the other systems. We have had to reverse engineer and build replacement parts in some instances to keep the systems operational.

ALTERNATIVES

Overcoming the problems presented by these situations can be done in many ways. One of these is to use relatively simple programmable logic controllers to implement the protective logic. PLCs offer ease of programming and program change. For the control computers, purchase the latest off-the-shelf model of equipment similar in capability to the original controllers, make the appropriate software conversions, and install the replacement equipment. These are practical approaches, offering another 25 or so years of more reliable operation. The hardware capacity can be expanded, providing capability of accommodating future system logic changes. One challenge is to rein in the costs of the software conversion, while doing a sufficient level of quality assurance on the changed software.

However, another uncertainty is introduced with a new computer system. This relates to the design of recent computer systems. There are more complicated features used in the design of these systems as compared to the design of the currently operating 1970 vintage systems. Greater use of cache memory, pipelining of data, parallel fetching of program instructions, and a myriad of other features provide greater speed of operation at the cost of the greater internal complexity. More complex real-time software operating systems are required to harness these features. Software conversion would be done using higher level languages. There is uncertainty whether the new system will behave like the older systems. There are many examples of emulation systems being used to reduce this risk. More sophisticated tools are required to assist in system debugging. Implementation challenges are not insignificant.

OTHER SOLUTIONS

Another approach is to look at other solutions to meet the requirements, whereby the software element might be removed, yet a changeable design implemented. The system could be designed so that testing could be far more thorough and more easily carried out than using a software based approach. A design offering these advantages could be implemented using configurable logic devices. These are more generically termed Programmable Logic Devices, or PLDs.

DEVICE HISTORY

Programmable logic devices have been around and evolving since the 1970s. Some of the earlier forms were application specific integrated circuits, ASICs, which were fabricated by IC suppliers based on a customer's logic design. Later came programmable array logic, or PALs, a more generic device that could be configured by the user. These early devices were limited in I/O and logic capacity. They use various logic configuration approaches such as fusible links, where a device was "burned" for a particular application. Later still came programmable gate arrays; these are configured using configuration data that interconnects the gates within the chip to implement a required logic design. These devices are being manufactured with as many as one hundred thousand gates per device, offering capabilities for highly complex logic, or many logic paths, to be implemented in a single device. The pinouts on these devices offer up to several hundred configurable connections to the outside world, providing the capability of connecting to hundreds of inputs and outputs, and of implementing complex logic functions.

PARALLEL LOGIC PATHS, FUNCTION INDEPENDENCE

An important feature of the gate arrays is that the interconnected gates of the logic provide independent parallel paths for the various logic circuits that are implemented. The transfer function connecting an output to any number of determining inputs operates independently of all other circuits within the chip. Logic timing for each path is a function of the number of gates and interconnections used by the particular set of logic, with pin to pin transition times in the order of tens of nanoseconds; the circuit can be modeled to determine precisely the timing characteristics. Furthermore, predefined time delay characteristics can be designed in.

CHANGEABILITY

PLDs can be configured by a user who has the relatively inexpensive PC based design tools. Logic designs and changes can be implemented by the "customer" without requiring manufacturing support. Device logic can be reconfigured in the field, with a very short outage, without the need to remove the device from its application. Although logic devices can be changed in situ, the logic can be fixed into a storage device so that it does not change during system operation or outage. Non changeable devices could be selected if the required level of system security requires that feature. However, for a system that undergoes frequent changes, system unreliability can be easily introduced by having to remove modules or chips to make changes. Field configurability can be accommodated by providing the on-board connections to accept the logic from a PC and "permanently" retain it in a storage device for transfer to the logic device on power up.

ADVANTAGES

The flexibility and capability of PLDs offer many significant advantages for applications within fuel handling, as well as other nuclear applications. The speed and logic capability offer the possibility of completely dedicated logic for functions such as the protective interlocks. Implementation of the protective logic in the older software based systems has worked well. The cycle times of most processors, including the minicomputer systems of the early 1970s, have performed the job adequately. However, there are many logic paths that are processed sequentially in the protective software cycle. There is timing dependence, as well as system function interdependence, of all logic paths on all others. It is not practical to design tests that would test all situational combinations. There is not enough time in the next millennium to run such a suite of tests. At the other end of the technology spectrum, hard-wired relay logic can be readily implemented using PLDs. Rewiring of relay contacts to make logic changes would be eliminated. The logic replacement offers "programmability" for any future changes.

CASE STUDY

The Bruce A protective function can be readily handled by separating the protected drive subsystems into a number of sets. Each set of drives could then be protected by a single logic device. The fuelling system drives fall naturally into sets: There are the four unit bridge and carriage systems, the service area drives, the trolley, the fuelling machine, and the system auxiliaries. The drives controlled by the outputs for each subsystem are independent. Input signals from the various subsystems are required for the interlock logic of each of the subsystems. Eight identical modules would accommodate the inputs, outputs and logic for the eight subsystems. The logic for the programmable gate arrays would be unique for each of the eight modules. This set of modules would be identical for each of the six control systems that make up the station complement of fuelling systems, one for each of the two fuelling machines associated with the three trolleys. These modules could be used for other general purpose logic functions as well.

DESIGN TOOLS

The tools for creating the configuration data for gate arrays and for simulating the performance of the resulting "circuit" offer full computer aided design capability. Schematic capture tools can be used to input the logic in various familiar representations. Logic can be entered as Boolean logic statements, truth tables, logic symbols or as ladder diagrams, for example. The tools provide the capability of designing a higher level user design entry language that is best understood by the designer, and can then present the logic in other formats, as preferred by the customer. After the logic design is complete, the tools generate the configuration logic directly; there is no "hands on" manipulation of the "code". Then, a logic "fit" can be done, based on a target gate array device. The gate utilization factor can be determined, that is, defining the spare capacity in the device. This factor defines the growth capacity for logic changes as the design or system matures. The system signals can be simulated to determine the time response of the various circuits implemented in the device. All of the design and system testing can be simulated before committing the design to actual circuitry.

OTHER APPLICATIONS

PLDs are of interest at this time because of the numbers of gates available on a single device, and especially because of the increase in the pinout of the devices. The speed and logic capabilities have led to their use in other areas within nuclear applications. These devices form the logic elements on many modules making up modern computers and especially emulators of older technology machines. Recently the PDP-11/70 computers and memories of the fuel handling controllers in the Darlington station were replaced with a single module, functionally equivalent, but much faster than the original.

The need for testing of the software changes for the Bruce stations has been met by designing an interface module around one of these devices. This interface makes a dual port memory resident in a PC appear as a simulated fuelling system. The PC has sufficient speed to simulate the actions of the fuelling system. The control or protective computer software can be tested without intrusive changes that would otherwise be needed to simulate system operation.

This same interface module has been adapted for other functions. By changing the configuration logic, and with minor software changes in the control computer, logging data can be passed at high speed transmission rates from that computer to a PC for transfer to a station local area network, providing up-to-date log data in the technical office. Having this log data on-line on a PC offers the capability of easier sorting and analyzing log to determine system problems and maintenance needs.

The problem of the unreliable I/O equipment at Bruce A can also be resolved by making use of the interface module mentioned above. This device can be reconfigured to provide an interface between the current control computer and a recent model I/O system. This approach permits the selection of the best available I/O hardware, providing an interface whereby the controller software remains unchanged and continues to interact with the system in the same way as before the change. System qualification and testing can be done in the lab, reducing the commissioning effort after installation of the replacement system.

LOGIC SIMULATION AND TESTING

One of the greatest benefits of using this technology is the ability to more completely test a logic implementation than is the case with a software based design. In the case of the protective system noted above, there are four drives associated with the bridges and carriages for each of the reactor units. There are numerous input signals that define the protective logic for each of these drives, many of the signals common to all four drives. With the current software based system, these four drives are included in the same protective logic system as the other fuelling system drives. It is practically impossible to test all combinations of all signals occurring in the common processor in testing the logic for an individual drive. Limitations have to be placed on the test cases to achieve reasonable assurance that the logic is correct. With the alternative independent logic of a PLD, more complete testing can be done to provide assurance that requirements are met. Since it is practical to segregate the logic for the subsystems into separate modules, a change in one subsystem has no effect on the others. Of course, multiple software systems could also be used. In this case there would still be more significant timing variations resulting from the variable paths through the software logic. Software quality assurance objectives can be met, but there is still a greater compromise between exhaustive and practical testing levels.

DESIGN CYCLE TIMES

Recently there was an opportunity to propose a design for a ventilation system control for the Bruce A station. This control uses channelized data from 24 temperature monitors at critical locations within the plant to activate the ventilation system. The logic requirement was that if any of the signals was determined by two of the three channels to be over the temperature limit, ventilation would turn on. The logic for this circuit was defined, the configuration logic for a particular gate array was established, and simulation tests were run to prove the design concept within relatively short design time. The design implementation cost for this control would be negligible compared with a software based approach. The front-end testing that could be done to prove the design would significantly reduce the commissioning effort after installation of the equipment.

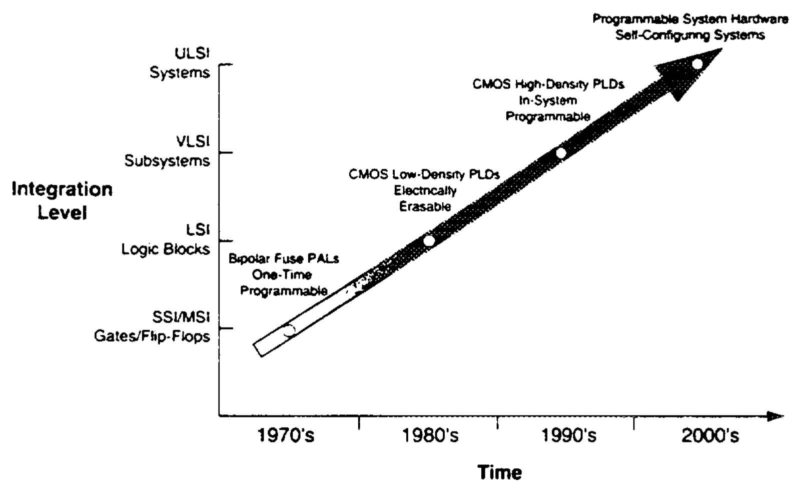
RISKS

Any new design approach is not without risks. The main risk in use of the technology is the fact that "custom-designed" hardware is required for its implementation. There are no off-the-shelf general purpose modules available offering a customer base of experience. The problems of designing the appropriate interface for the logic devices to withstand the environment in which they will be operating depends on the skills and experience of the design group. The upside of this is that the design can be tailored to fully meet the requirements of the particular application without the need for costly modifications of a "standard" product line. Reliability of interconnection to external devices provides a design challenge. From another perspective, there is the potential for delays in acceptance of changes to use this newer technology by the regulatory authorities.

SUMMARY

Field programmable gate arrays provide an inexpensive, yet versatile means for implementing control and protective logic within a nuclear power station. The reconfigurability of the logic devices will accommodate the changes and expansion found necessary in logic functions. The system logic and its timing can be well defined and tested due to its deterministic nature. The capabilities for completeness of testing will reduce the level of effort required during the system commissioning phase. Separating the logic into individual modules greatly reduces the interdependence of functions within the "logic processor". The completeness of definition and testing of the logic functions should ease the burden of getting regulatory approval for design changes.

Programmable Technology Evolution



REFERENCES

Choosing FPGAs, ASICs, or Cores for DSP Based System Design, B. Tuck, Computer Design, vol 35 no 2, Feb 1996

Technology Trends, Electronic Engineering Times, March 22, 1993

In-System Programmability Manual 1994, Lattice Semiconductor Corp.

Cypress Programmable Logic Data Book, 1996, Cypress Semiconductor Corporation.

Altera FLEX 8000 Handbook, 1994, Altera Corporation.

PURSUIT OF EXCELLENCE IN F/H OPERATOR PERFORMANCE

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Abstract:

Performance of any F/H system is heavily dependent on the performance of the system operators. Sadly, this topic often receives little attention until incidents occur.

Darlington has taken a proactive approach designed to reach and maintain excellent operator performance.

Expectations and Standards are generated and published. A detailed, goal oriented, auditable training program which includes hands on training modules ensures operators reach the standards before qualifications are granted. Refresher courses maintain current skills. A depersonalized, computerized reporting system ensures training and standards are updated to match current situations.

F/H System at DNGD

The Fuel Handling system at DNGD can be viewed as a special purpose robotic manipulator encased in a pressure vessel. The pressure vessel forms an extension of a reactor Primary Heat Transport System during fueling operations. Fuel Handling routinely passes fuel through containment boundaries, and handles irradiated fuel bundles, each one of which can contain active elements several orders of magnitudes greater than allowable releases to the public.

The fuel handling system at Darlington has three trolley systems and two Fueling Facility Auxiliary Areas (FFAA's). The three trolley systems run in a common duct under the four units to visit either a unit that needs fueling, or to an FFAA to load new fuel or discharge irradiated fuel.

A trolley consists of two half trolleys, and its associated FFAA. Each trolley half has a fueling machine, and a set of auxiliaries that provide D2O and air at appropriate pressures and temperatures as is required during the fueling process.

Control System

The operational computers scan and operate about 5000 I/O points and operate 4 work stations per trolley half. The main workstations have a CRT and two keyboards for operator interaction.

The computer controls the system in real time. Operators insert high level commands called 'jobs' which define for example, the number of bundles to be pushed in a reactor channel.

Operator Interaction with F/H

The man machine interface puts the operator in the loop in three ways:

1) Command - The control system requires operators to initiate fueling runs. Following initiation, operators monitor the system to have a clear view of the status and to be able to act appropriately in an event.

2) Monitor - Sensor display data tell the operator the status of the system, and allows him to take effective action to maintain control during an event.

3) Log - The system logs all auto and manual operation. Operators manually log fueling operations and problems. This provides an opportunity to review operator and machine performance at a later date if required.

Field Work

The Fuel Handling system requires extensive field support. New Fuel Bundles must be loaded into the New Fuel (NF) mechanism, and their serial numbers must be entered into the bundle tracking system. When fuel comes out of the reactor, storage modules must be loaded onto the Irradiated Fuel Discharge Mechanism (IFDM), and must be moved for final storage when full.

Any special work requires the loading of special equipment or components. Discrepancies between the expected and actual components in the FM can have drastic consequences at the unit.

Previous Training Systems

Operators were trained by using several levels of Training Manuals (TM's) to operated the machine via operating manuals. The TM's explained and justified the principles that operators must follow in the Operating Manuals (OM's), and deviate only with appropriate authorization. Although good in theory, often there was missing or conflicting information. Often the TM's did not reflect the OM information, and were not kept up to date. Many situations were not covered in either document.

The training manuals were poorly written, making information transfer even more difficult.

As a result, much of the training was done via 'oral tradition' or individual beliefs held by senior operators as to status of systems. Often based on faulty information or misunderstanding, these legends developed a life of their own independent of the station documentation. This process was a major impediment in the pursuit of operator excellence.

Many standards and expectations were implied. Not formally documented, individual interpretations led to haphazard interpretation and execution.

Requirements for Operators

The goal of any operator performance program can be stated simply:

Ensure the system is operated safely and correctly, and correct responses are made to fault conditions.

Excellence in operator performance is achieved when this standard is defined, and always met.

Taking these very general statements and converting them to usable daily directions is where the circle of excellence comes in. The circle generates standards and expectations which cover both the philosophy and direct operation of the job, as no set of written instructions can cover all issues.

Circle of Excellence

The unified approach to operator excellence is best described in the model described in Figure 1.
The circle model conveys several important concepts:

The movement is continuous and self renewing. As individuals enter into the circle, their actions and experiences are a building block for the next step.

The circle is an iterative process, adaptable to the real world, where 'perfection' is a moving and changing definition.

The circle has mechanisms for defining a direction, determining how far off from that direction you are going, and it has tools to get you back on track. Since the redefinition phase repeats, the circle analogy allows for changes in direction as the work environment changes.

Initial Selection:

Initial selection of candidates is based on corporate hiring practices, which provides assurance of a reasonable level of capability. Movement into Fuel Handling is mostly on a volunteer basis, which is the first step in having a positive motivation towards job excellence.

Initial Training

Initial training (level IV) at DNGS starts with a common stream for all operators. Courses cover the full range of station systems and both radiological and conventional safety.

Fuel Handling Specific Training

Following completion of generalist operator training, candidates may wish to enter into the F/H department. This set of training is performed in the following order:

Field Training: Operators are trained in all the field equipment, with a combination of classroom and on the job training. Expectations are that they will obtain a clear understanding of all F/H field equipment by the time the are fully qualified.

Main Panel Operator (MPO):The MPO level of training begins where the field studies stop. Candidates are introduced to the control panels and control systems. Training concentrates on explaining the 'why' as much as the 'what' of control system operation. Candidates are examined to ensure they have an in-depth knowledge of fuel handling from a panel perspective. Mastery of this package allows candidates to be co-pilots.

Copiloting

Copiloting is the transition phase where individuals combine knowledge based training with actual operational experience. They are allowed to perform OM procedures under the direct supervision of a qualified panel operator. This allows them to gain experience in a limited risk situation..

After the specified period of co-piloting, a final interview concentrates on Operating Principals and Procedures (OP&P's), and the Recommended Operating Practices (ROP's). These emphasize the operating philosophy as much as the logic. Operators at this level have the additional responsibility of dealing more with interfacing work groups, i.e. maintenance, Authorized Nuclear Operators (ANO's) who operate the reactor units.

Experience

Experience is the gist of the mill in the pursuit of excellence. Operators, like most of us, learn a great deal through properly structured experience, in both the copilot and operational phases. Experience generates the problems which measure the successes and failures of the process. Properly recorded, experience helps point the direction to excellence.

Standards and Expectations:

Standards and Expectations are key to the pursuit of operator excellence. Individuals will not strive for excellence if they do not know what excellence is.

Standards are measurable goals against which performance of specific actions are measured. Expectations are the requirement that certain actions be performed or attitudes be upheld.

The standards and expectations documents are the base of the pursuit of excellence.

- Standards and expectations define the current definitions of excellence, or the manner in which it may be approached.

-By comparing standards and expectations against a new goal, or challenge, it measures the amount of change required to move to the new direction.

- Standards and expectations define the requirements of tools to get to the level of required excellence.

Expectations come from the stakeholders in the performance process, customers, and the group themselves. They includes technical and maintenance groups, the operations coordinator, and the operators.

Generation of Expectations

A major thrust at DNGD in the last year has been to formalize standards and expectations.

DNGD has taken the tack that there must be 'buy in' for expectations to be adopted generally and upheld. The first stage is for management to provide the expectation that the F/H system will be operated with a high degree of professionalism. 'Performance Objectives and Criteria' provide a starting point. The next step is to have operators accept that having published expectations is a valuable tool to achieving the first, agreed upon expectation.

The fact that operators provide many of their own performance standards is a measure of the buy-in they already have. Operators (as with other work families) recognize any tool that helps them achieve excellence is in their own self interest. No one wants to initiate or worsen a caper.

The generation of 2nd level expectations was done in a cooperative environment to maximize buy-in.

Expectations were fleshed out by field staff, and given concrete form that was meaningful in the work environment. As an example, A housekeeping standard by itself is open to interpretation, as two people may have two different views of acceptability. A concrete interpretation of that general standard came from a field operator: "all fuel pallets must be stacked in the north end of the fuel area".

Vigilance

The lessons learned by experience are only retrieved through vigilance. The skill, or expectation is hard to cultivate, and easy to loose. Issues such as a positive attitude for incident reporting, perceived positive effects of previous reports, must be addressed to have an effect vigilance program. A vigilance program is often made more effective by the use of a reporting system:

Fuel Handling Incident Report (FHIR) and Technical Section Input

To aid in the process of vigilance, Darlington NGD has incorporated a FHIR data base, with the Significant Event Report (SER) database. FHIRs have a format similar to SER's, but with circulation limited to the F/H family. F/H personnel enter depersonalized reports of incidents that may affect F/H, but are not classified as SER's. FHIR's are reviewed by operating maintenance and technical section and action items are generated. Regularly, the action items are reviewed for completion and are closed out when complete.

FHIRs perform several functions: 1) they may indicate short coming in the OM's or TM's, 2) they may indicate a good response to an incident that should be communicated to the operator group; 3) They may illustrate a new hazard that is not clearly understood; 4) they may document an ongoing problem that needs attention from another work group.

Followup actions of FHIR's often involve the Technical Support Unit, and/or design. FHIRs help those groups focus on issues important to the operator group.

Internal Peer Capability:

Another important player in directing the pursuit of excellence is an Internal Peer capability. On an ongoing basis, this process asks people, "Are we going in the right direction and are we near where we want to be?". Examples of internal peer tools are questionnaires and specially trained people who compare standards and expectations documents with day to day operating experience.

Training and Operating Documentation Updates

The results of FHIR's, and Internal Peer Audits must be used to ensure training and operating documents reflect the pursuit. Documentation that is out of date brings the whole system into disrepute, and reduces the by-in for the rest of the process. Updating of the documentation demonstrates to the operator family that their needs are important, and that the job is important, besides the obvious attributes of having correct procedures and information.

Motivational Tools

There are various tools available to shape the performance of the operator family. Many have been tried and found wanting. The secret of successful use is a combination of how the tool itself is formed, along with its use.

Tools for the pursuit of excellence are often straight forward. If a standard requires that a certain action be performed safely, a tool will be an authorized procedure to perform that function. Operating manuals, training manuals are tools to reach a standard of system knowledge.

Job expectation guides both inform and measure against the current definition of excellence, as embodied in the document.

The model of the pursuit of excellence in itself is a tool, used to visualize the process.

A second key tool of motivation is respect. Members of the F/H family must respect each other, and the contributions they bring to the table. Each job group must realize the important contribution the others make to the overall goal of F/H.
Respect (buy-in)

Respect must infiltrate every element of the model for it to be effective:

Operators must respect the aims of the program, the tools that are provided, and the trainers, and standard providers.

The standard givers and trainers must provide tools the operators can respect, and must respect the operators ideas and opinions. Respect and disrespect are shown in many different ways. One of the most important aspects in this model, is appreciating and considering other peoples points of view, and not discarding them simply based on the source. It is important to recognize that each group in this circle brings different talents to the table, and there is no one group that has monopoly on them all.

Job Performance Measures

Job Performance Measures (JPM's) provide an important tool to measure quality of the physical actions of an operator. The JPM instructs the operator to perform a certain sequence of operations, using the real equipment. This allows the operator to form a kinetic (body image) memory of the action, increasing the chance of correct operation in the time of need.

Refresher Training

Refresher training provides a means of the operators meeting expectations on an ongoing basis. Refresher training reinforces components of the job expectations that are not normally performed as a part of day to day duties.

Key aspects of refresher training are:

Selection of Training Topics

The FHIR data base helps in the selection of training topics. New hazards, challenges, or changes covered since the last session are easily detected and covered. Topics will be relevant as they have already happened.

Refresh rate

The refresh rate is affected by the complexity of the covered material and the rate of use in the field, and the projected use of the information.

As an example, refresh training on maintenance equipment used during outages should be provided, and provided just before use during the outage.

Procedures seldom used, i.e. power transfer procedures, should have a higher refresh rate than a procedure that is used every fuelling push.

Fitting in Shift Cycle

Since the customers for the training are usually on shift, multiple editions of the training should be considered.

Problem Reporting and Feedback

Trainers must be responsive to course feedback. While it will not be possible to please everyone, feedback can provide important considerations for course design.

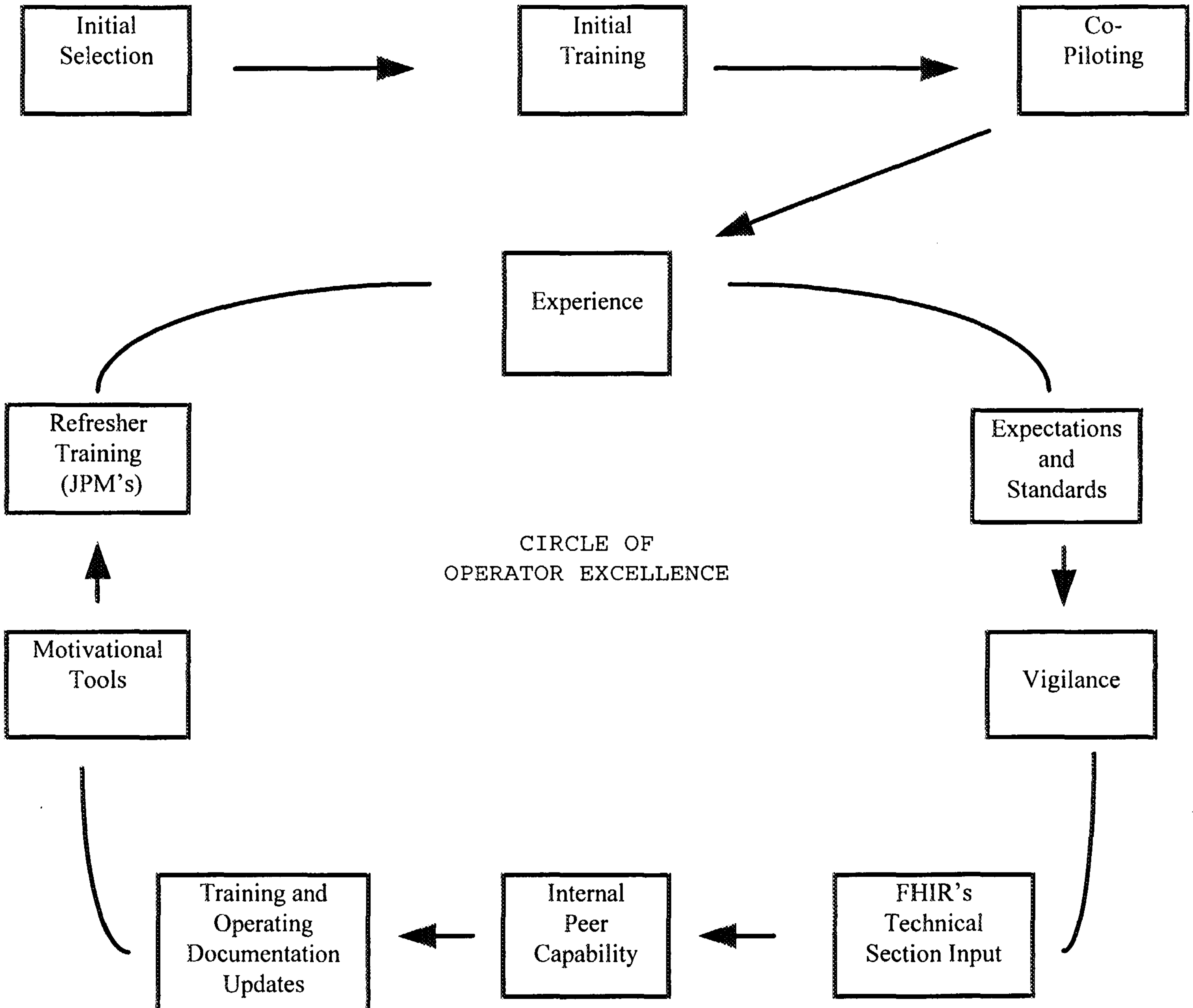
Initial Selection
Initial Training
Co-Piloting
Experience
Expectations and Standards
Vigilance
FHIR's Technical Section Input
Internal Peer Capability
Training and Operating Documentation Updates
Motivational Tools
Refresher Training (JPM's)
CIRCLE OF OPERATOR EXCELLENCE

FIGURE 1

BETTER FUEL HANDLING SYSTEM PERFORMANCE THROUGH IMPROVED ELASTOMERS AND SEALS

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1. ABSTRACT

In the area of elastomers, tests have identified specific compounds that perform well in each class of CANDU® service. They offer gains in service life, sometimes by factors of ten or more. Moreover, the aging characteristics of these specific compounds are being thoroughly investigated, whereas many elastomers used previously were either non-specific or their aging was unknown. In this paper the benefits of elastomer upgrading, as well as the deficiencies of current station elastomer practices, are discussed in the context of fuel handling equipment. Guidelines for procurement storage, handling and condition monitoring of elastomer seals are outlined.

In the area of rotary seals (non-elastomeric) there have also been significant advances. In the fuelling machines, low friction is needed for accurate positioning. The development of an improved seal is described, which since 1992 has saved at least M$1 per year at Pickering alone.

2. INTRODUCTION

Over the past twenty-five years the Fluid Sealing Technology Unit (AECL, Chalk River) has helped solve a variety of sealing problems in CANDU® fuel handling equipment, primarily in response to specific station requests for help. Following are some of the component problem areas that have been investigated and resolved:

- D_2O hose failures (Darlington '95, Lepreau '91, NPD and Douglas Point)
- Marotta fill, vent and drain valve seal retrofit (Darlington '94)
- Snout plug and squeezer mechanism seals (Pickering '91)
- Darlington and Bruce flathead O-rings and quick-disconnect seals (~'93)
- NPD snout seals
- Ram ballscrew face seals

We are now at a point in our knowledge of elastomers, where we have identified specific elastomer compounds that have "superior" performance in each class of CANDU® service. A wholesale upgrade of seals on fuel handling equipment to the "superior compounds" is recommended. Furthermore, adoption of improved elastomer quality assurance, procurement and handling practices as well as improved in-service monitoring are recommended.

3. ELASTOMERS

Historically, equipment in nuclear plants has contained whatever elastomer each component supplier traditionally used for non-nuclear service. Most plant operators do not realize the resulting proliferation of elastomer compounds, many of which are far from optimal for the service conditions, has multiplied the costs to provide station reliability, maintainability and safety. By standardizing on a handful of "superior" elastomer compounds, service life can be extended, maintenance planning and safety can be improved, environmental qualification can be streamlined, and procurement and handling of replacement parts can be simplified. Extensive tests have identified high performing compounds for each class of CANDU® service. These designated "superior" compounds offer gains in service life sometimes by factors of ten or more over others being used. Moreover, the aging characteristics of the "superior" compounds are becoming much better known than those of other compounds, many of which are only known by generic type.

4. CRITERIA FOR ASSESSING ELASTOMERS

To establish the life of any elastomeric component, the type and magnitude of deterioration that causes a failure in service must be known, as well as the deterioration rate. There are many possible types of failure (e.g., extrusion, chemical attack, wear-out, tensile cracking, load relaxation and compression set). They are all influenced by a combination of material characteristics and service conditions. Commonly reported material properties alone cannot quantify extrusion resistance, etc. Some properties are closely related to common failure modes (e.g., tensile strength to tensile cracking failure), but other commonly quoted properties such as hardness, ultimate elongation, permeability and thermal expansion are only tenuously associated. A primary challenge in assessing an elastomeric component for a particular service is to determine when and how it might fail in service. Relevant and comprehensive failure criteria based on functional properties are necessary for meaningful life prediction. These need to be based on a fundamental understanding of the performance needs for the particular application.

5. IMPORTANCE OF COMPOUND- AND SERVICE-SPECIFIC DATA

A database of relevant properties and behaviour is key to elastomer selection and life prediction. This data must be compound-specific because within a given elastomer class (e.g., nitrile), the base polymer is compounded with varying amounts of fillers, vulcanizing agents, anti-oxidants, anti-ozonants, processing aids, plasticizers and accelerators from any number of suppliers. These variables, and the method and degree of mixing and curing, all profoundly affect functional properties of the final elastomer product.

Properties alone, however, are not enough for the database to be usable for severe service. It must also include *service*-specific behavior. Most of the published data on elastomers is misleading because: (1) the effects of the fluid are neglected (e.g., air versus water versus other fluids), and (2) the measured damage parameters that life predictions are based on often have little bearing on how a part actually fails in service. Parameters are more often chosen for testing convenience than for functional relevance.

Besides choosing the most appropriate damage parameter(s), the level of damage considered to constitute a failure must also be chosen judiciously. Compression set may be correctly identified as the most likely failure mode for a particular application, but its magnitude for failure may be poorly estimated. For example, a compression set failure criterion appropriate for a piston seal will severely underestimate life for a bolted flanged joint. A high compression set criterion is appropriate for a flange seal since the seal is in a static, highly squeezed face seal arrangement, with no extrusion gap and no changes of squeeze. In contrast, a low compression set criterion is appropriate for a piston seal that is dynamic, lightly squeezed for low friction and wear, and has parts with tolerance stack-ups that create significant eccentricity between the piston and bore.

Naturally, the amount of testing must always be a compromise with the value of the results. Consider that accelerated thermal aging requires data at four temperatures for reasonable extrapolation, and at each temperature the duration of the test must be iterated to obtain the desired level of damage (see Figure 1). Consequently, developing a compound-specific database can become very expensive. This is another reason to rationalize the number of compounds used in the field to the fewest that adequately cover the required range of applications.

6. "BATCH" APPROACH TO QUALITY ASSURANCE

To ensure relevance of elastomer test data, the ingredients and processing variables for each chosen compound must be closely controlled for consistency, both in original qualification testing and in subsequent service. To ensure that the correct specific compound is received, purchasing specifications must not open the door to other compounds in the same class of elastomer. Otherwise, performance in service may be unacceptable (i.e., low safety margins, unreliability and frequent replacement). If alternative compounds are needed as back-up, each must be separately qualified. Purchase specifications, as a minimum, must require that each elastomer seal be of a particular compound, traceable to the particular "batch" of ingredients, mixed and processed together to form the unvulcanized stock from which the part was made. A certificate of conformance should be supplied specifying the compound, its batch number and date of cure, along with the hardness, specific gravity and tensile strength of samples from that batch, as compared with the manufacturer's expected values. In this manner the elastomer compound can effectively be characterized.

7. DEFECTS IN ELASTOMERS

Inspection methods and rejection criteria for defects are often neglected in quality assurance programs for elastomeric parts. This can compromise their integrity and reliability. Surface defects in most parts can be detected by the unaided eye. In the case of elastomeric parts this is most effective when the elastomer is strained appropriately (i.e., by bending, pulling, pressurizing, etc.), since many defects such as cuts and tears are difficult to see in unstretched parts. Size can be compared to reference standards that correspond to acceptance limits, and (if necessary) measured using optical and mechanical aids (e.g., calibrated magnifier, depth-measuring microscope, stylus profilometer).

For detection of internal defects (inclusions, voids, discontinuities in reinforcing materials, etc.), as well as for monitoring state of aging (on the shelf or in service) a non-destructive technique called elastodynamics has been found most useful. A tool has been developed, employing this technique for inspection of O-ring seals. Basically the tool measures reaction force on two pinch rollers while the seal is driven and squeezed between them. Localized defects are signalled by spikes in reaction force. Any generally low or high force, or variation around the seal, signifies abnormal properties when compared with a known baseline. A version of this tool, developed for monitoring the condition of elastomeric parts of all types, is described in Section 9.

8. STORAGE AND HANDLING

Elastomers are subject to deterioration with time, temperature, and other environmental influences. Ideal storage conditions are cool, dark and free from contaminants (such as ozone, solvent vapor, etc.). Elastomer parts should be stored in a relaxed state, free from strain (i.e., not folded, twisted, or hanging on a rack). Their shelf life (expressed as expiry date) should be stated and be rationally based (e.g., if 90% of "as-new" lifetime for the particular service is deemed acceptable, and proper storage at the maximum allowable temperature is known to cause 1% loss per year, then shelf life is 10 years). Many elastomers are very stable under store-room conditions. Measurement of critical functional properties (e.g., compression set, extrusion resistance, hardness) of elastomers of certain ethylene-propylene and nitrile compounds stored under proper conditions has shown them to be essentially unchanged after more than twenty years (for example, see Figure 2). Unfortunately, not all elastomer compounds are this stable.

Many elastomers are incompatible with common solvents. Ethylene-propylene elastomers are notable for their lack of resistance to petroleum-based products. Some elastomers have very poor resistance to cutting and tear propagation. To cover the many facets of proper use and handling of elastomeric parts, a one-day on-site training course for mechanical maintainers has been developed. This has been very well received by station personnel, and has been presented at eleven venues.

9. MONITORING IN-SERVICE DEGRADATION

Nuclear plants contain many examples of vaguely defined elastomer compounds that make accurate prediction of service lifetime impossible (yet many of these parts are accessible for interim inspection). Even in applications where well-defined compounds are used, the service conditions may not be well known. There are also cases where station personnel would like to assess the state-of-aging of elastomeric components held in inventory for long times.

In applications such as these, a recently developed tool called an elastodynamic spot tester promises to be very helpful. Sufficient data is needed to show how a particular elastomeric compound's properties change qualitatively with time in the service environment. Also, the starting quantities and minimum required quantities for these properties must be known. Then, making an interim measurement of these quantities using the elastodynamic spot tester gives a non-destructive method to pinpoint the current "effective age" of the part in terms of the percentage of service lifetime expended.

For example, most elastomers age-harden. If hardening is the failure mode and functionality of the part requires hardness less than say 80 durometer*, with new parts being 70 durometer, then an interim measurement of hardness of 75 durometer suggests that half the service life remains (50% effective age), assuming the degradation rate is linear with time. However, aging data may also show that hardening accelerates and that the part's age is therefore already say 90%.

The advantage of elastodynamic spot testing is that it measures much more than hardness. It also measures stress relaxation and recovery, and can measure these as a function of mechanical stress level in order to discern the onset of permanent

* "Durometer" is a measure of rubber hardness, as determined by indentation depth of a stylus under a given load after a given period of time.

damage. Essentially, the stiffness, damping and strength are quantified to effectively "fingerprint" the material. It then only remains to relate these to age, in the same way that measurements of weight, running speed and subsequent pulse rate, for example, might pinpoint the effective age of an individual in a population of genetically and environmentally similar humans.

10. FUELLING MACHINE RAM SEALS

For fuelling machines in Pickering and the 600 MW CANDUs, the severe requirement is for seal torque to be no more than 0.6 N.m—low enough to achieve accurate positioning of the rams. Seal gland conditions are:

- 25 to 55°C
- 0, 3 or 10 MPa (gauge)
- 24, 36 or 246 rpm, reversible

Reliability is essential. Many seal types were tested during early development. The most acceptable was a six-pocket-with-orifice hydrostatic design, similar to those used in the first CANDU Main Coolant Pumps at NPD (Nuclear Power Demonstration), then at Pickering. Its face materials were bronze vs. carbide-coated stainless steel. Leakage was about 2 L/min at full pressure, and its main failure mode was dirt or erosion affecting the orifice and de-stabilizing the seal, thus causing face rubbing, friction and wear. It was also difficult and expensive to manufacture, with quality assurance of the orifices, face coating, and the bellows for axial movement being particular problems.

Performance was tolerable throughout the 1970s and early 1980s because each plant kept a spare machine available, but the days of generous spares came to an end in the mid-1980s—higher reliability was demanded.

A different type of hydrostatic seal (Figure 3: CAN13 seal) was developed using analysis techniques and materials not available to the early CANDU designers. The bellows was replaced by an O-ring of material with demonstrated capability for the service. Silicon carbide was chosen for the new seal faces based on extensive hydrostatic seal testing in the 1970s, when it was found to be both resistant to erosion and tolerant of moderate rubbing contact. The need and means to hold it securely in compression (by shrink-fitting) to prevent breakage had also been understood.

Requirements for low torque and low speed necessitated a hydrostatic seal; the extreme range of pressure required a seal face that would neither erode while on pressurized standby, nor wear when rubbing at pressures as low as atmospheric.

The flat-conical seal face configuration was developed so that the flat region of the face would rub lightly and not wear out-of-flat while running at atmospheric pressure and 246 rpm. The conical region was optimized to meet the torque requirement while having minimum leakage at the higher pressures. There were three reasons for this:

(1) to minimize the erosive flow,
(2) to minimize the amount of dirt consequently filtered out between the seal faces,
(3) to reduce the 2 L/min leakage per seal suffered with the previous seal design.

CAN13 seals were first installed in Pickering in 1988 after extensive testing in laboratory testers and fuelling machines. Full commitment to this design was made in 1989 and completed by 1992. There have since been no failures of the 64 seals running in these eight reactors, despite dirty operating conditions in Pickering-B that have caused many pump seals to fail. The 600 MW CANDUs have similarly changed to the new seals.

Savings due to installation of CAN13 seals at Pickering are estimated to be at least M$1 per year, with about a third coming from maintenance savings and two thirds from extra kW-h of production. Extrapolating over the expected lifetimes of the existing plants now using this seal, the savings are of the order of M$20. The economic viability of future plants has also been much enhanced—all this coming from an original 1985 investment of $53,230 for design, development, rig-testing and supply of the first two seals.

11. ACKNOWLEDGEMENTS

The contributions of many colleagues and funding from the CANDU Owners Group towards the work presented here is gratefully acknowledged.

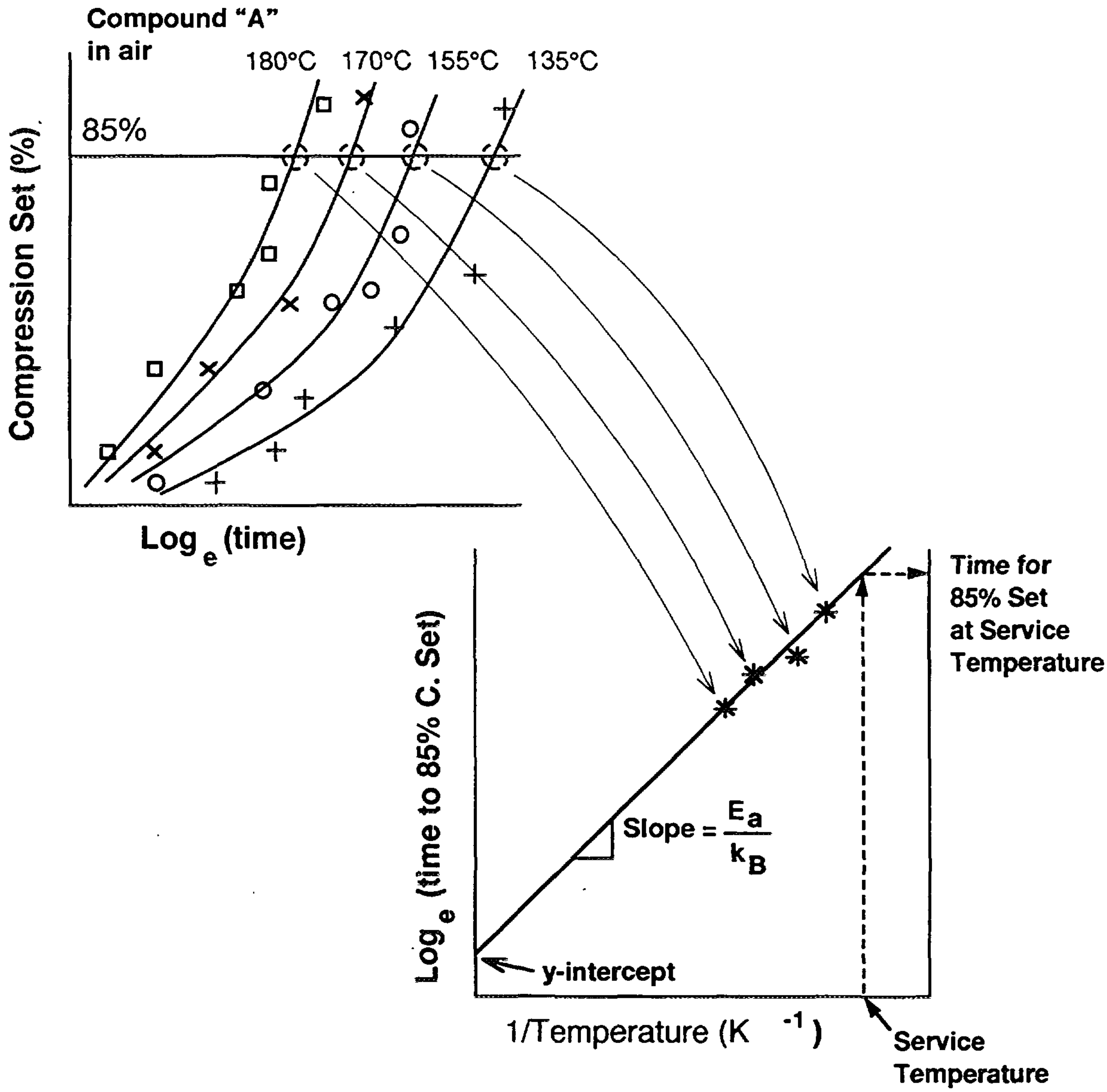

Figure 1: Arrhenius Thermal Aging Tests. This figure illustrates the number of data points required for a reasonable extrapolation.

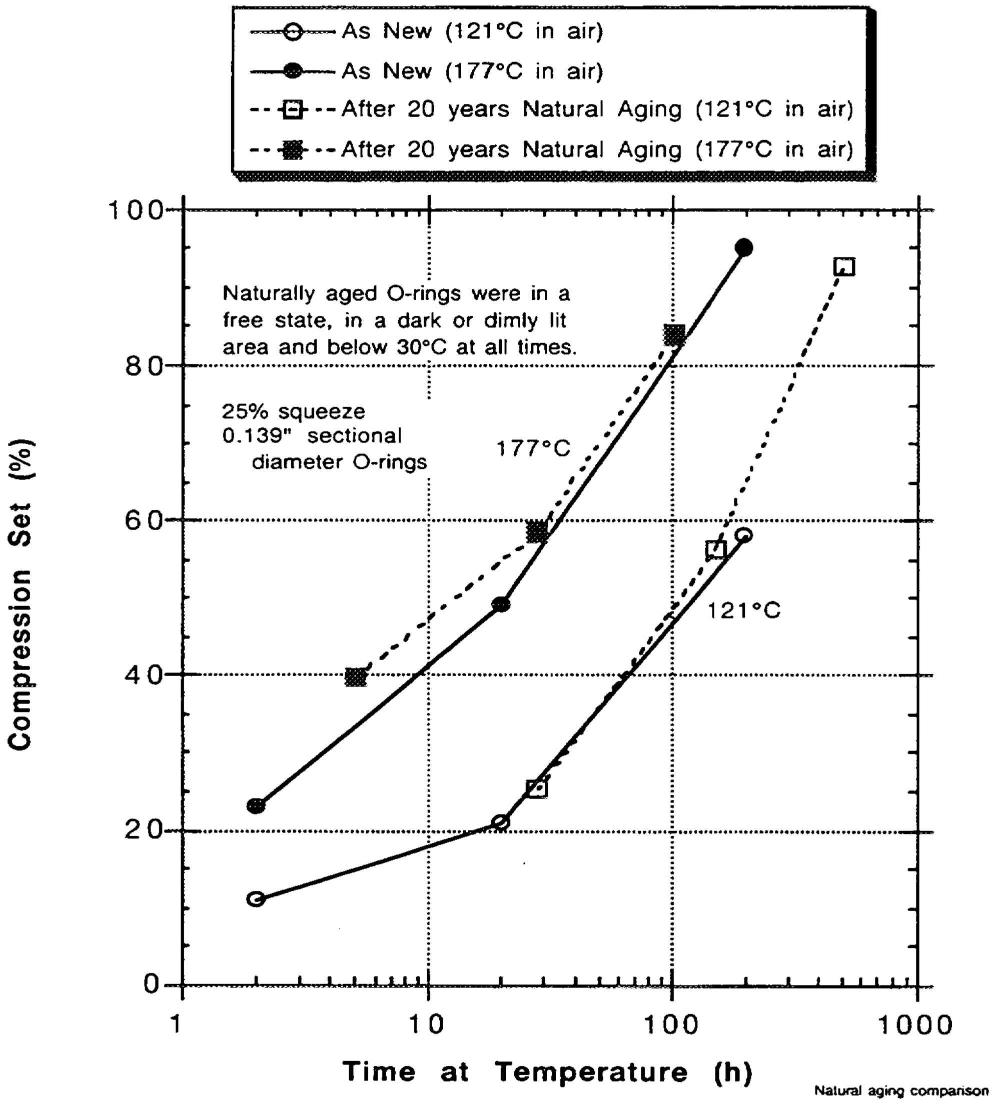

Figure 2: Effect of Twenty Years of Natural Shelf Aging on a Specific Nitrile O-Ring Compound.

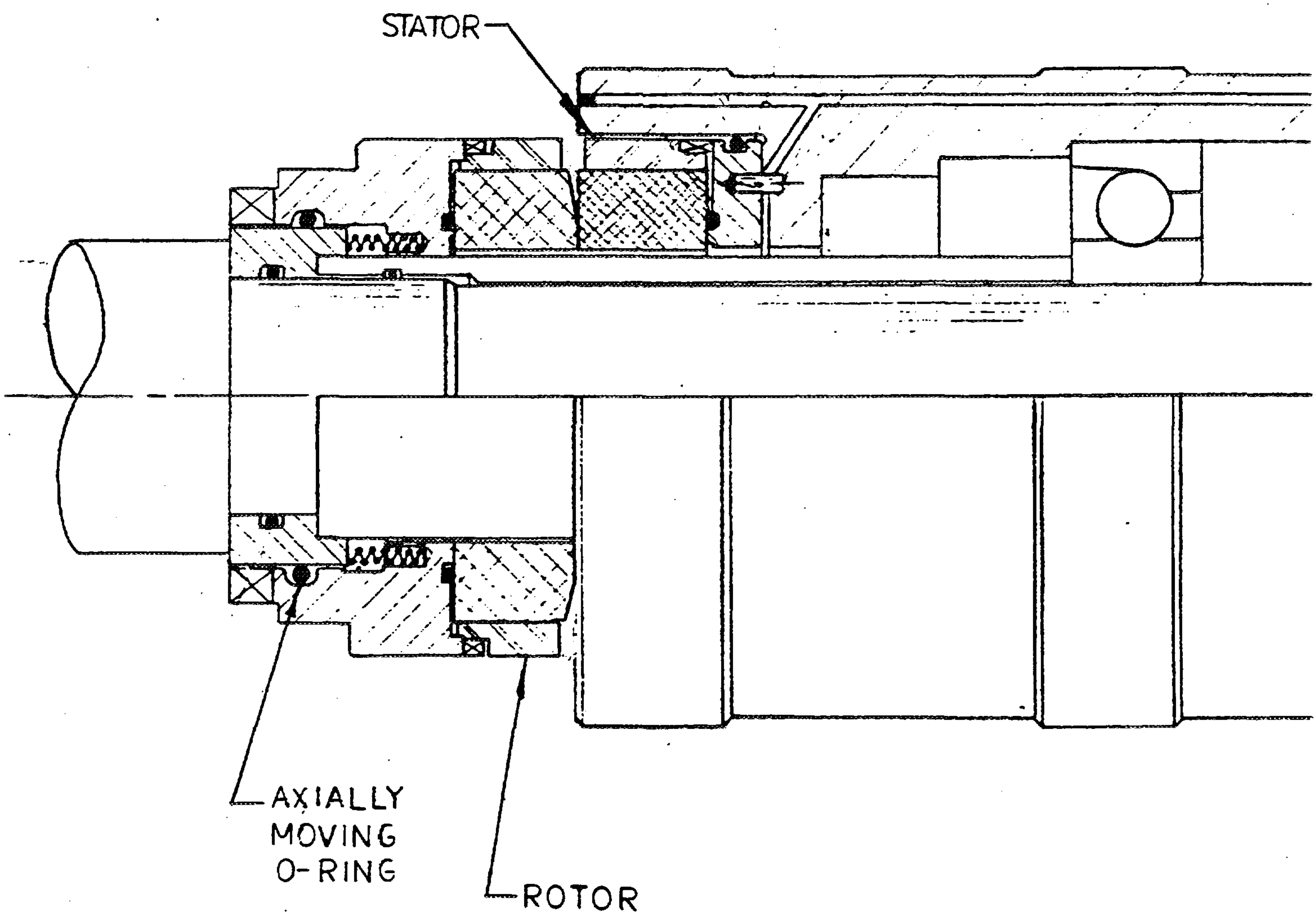

Figure 3: CAN13 Seal for AECL Fuelling Machine Ball Screw Rams. The rotating seal face is flat-conical to give hydrostatic lubrication when pressurized.

PICKERING IRRADIATED FUEL TRANSFER CONVEYOR ISOLATION

CA9700722

by

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ABSTRACT

Pickering A NGS has been in operation for 25 years and is one of the longest in service CANDU[1] stations. Some underwater fuel handling equipment, notably the conveyor stops, have been without maintenance throughout that time.

This paper describes the concept of a conveyor isolation system that permits draining of a single or multiple elevator columns and also the early stages of a development program for the elastomeric sealing element. The prototype seal element has been proven in lab tests to be capable of limiting leakage to 0.5 IGPM at the design pressure of 6.5 psi.

The design of a sealing element is particularly interesting because the conveyor tube is a square cross-section which contains an additional obstruction , a conveyor drive cable.

A seal delivery, actuating and positioning system has been conceptually laid out and the design is proceeding with projected implementation in 1998.

1. INTRODUCTION

The Pickering A irradiated fuel transfer system has been in service for more than 25 years and some components have been almost completely inaccessible for maintenance. Some of these items are the conveyor stops and possibly the conveyor tubes. To perform this maintenance, part of the conveyor tube must be drained and a temporary maintenance bung put in place to maintain containment for the reactor buildings connected to that conveyor system. The conveyor system requires confined space work rules while performing any maintenance in the tunnels below the elevation of the bay water surface.

The Pickering A irradiated fuel transfer system includes a long conveyor system that transports irradiated fuel bundles from the reactor buildings to the irradiated fuel bay. The conveyor system and elevators are filled with bay water which provides part of the reactor building containment system.

The conveyor system comprises a horizontal, watertight, square stainless steel tube; a conveyor cart, which carries the fuel, and the drive system for the cart. The conveyor tube is approximately 14 feet below the surface of the bay water and runs though a long concrete tunnel 4 feet wide and 4 feet high which connects

[1] CANDU® - Canada Deuterium Uranium Registered trademark of AECL

the elevators of both RB[2]s to the IFB[3]. The details are shown in Figures 1,2 and 3.

To drain the conveyor tube for maintenance, a means of isolating the receiving bay water is required. The bay can not be drained because of the presence of irradiated fuel particles which are currently shielded by the water.

If only one isolation point is provided it would be impossible to isolate the Unit 2 West elevator without draining all the other elevators in Units 1 and 2.

2. REQUIREMENTS

The conveyor tube sealing/isolation fixture must permit isolation of any single elevator bottom housing. The sealing system must have built in redundancy. The system must by mechanically locked into place when isolating the water column during personnel access.

Although not required to be absolutely watertight the seal must limit leakage to 0.5 IGPM[4]. The seal must be able to tolerate occasional exposure to gamma fields of up to 10 Rads per hour. The sealing element must have a service life of 30 days.

The sealing system must not require assembly or set up inside the small conveyor tunnel.

The sealing/isolation fixture shall be designed to conform with Ontario Hydro's ergonomic design guidelines.

Compressed air shall not be discharged under water from any pneumatic equipment used in the fixture design.

3. CONTAINMENT

Pickering A System

The reactor building is a part of the containment system designed to prevent the escape of activity in the event of any postulated accident to the reactor. To keep the size of the RBs to a minimum, reactor auxiliary systems which do not contain highly radioactive fluids and which, therefore, do not require the containment and shielding provided by the RB, are located in the Reactor Auxiliary Bay, which includes the IFB.

One of the functions of the spent fuel elevator in normal operation is to provide RB containment by means of a hydraulic seal between the RB and the IFB. However, when an isolated elevator is drained of its water and an opening is created in the lower housing or in the conveyor tube the containment boundary is broken via the openings in the top housing assembly. The openings in the top housing must, be sealed before draining the elevator. The major opening can be conveniently sealed by clamping the fuel transfer mechanism to the elevator. The two access holes at the top of the housing could be sealed using commercially available expandable bungs. This leaves the openings for the mechanisms (such as fuel stop and return ram assembly, its drive motor and gear box, etc) which are fastened to the top housing. It is unlikely that these connections are air-tight. If not, then this equipment must be removed and the openings temporarily sealed with blanking plates.

The sealing/isolation fixture provides the containment function for the other unit that is not opened up and may be at power. This is the reason that the bung must withstand the accident condition of 12.5 psig which is 6 psi greater than the design pressure.

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[2] RB - Reactor Building

[3] IFB - Irradiated Fuel Bay

[4] IGPM - Imperial Gallons Per Minute

Pickering B System

Pickering A and B spent fuel elevators are similar in design dimensions and in concepts with respect to reactor building containment. The B system has individual conveyors for each reactor building, therefore the draining of one elevator does not affect the other building. The concept for containment maintenance applied in A is also applied in B. That is to make the elevator top housing and upper column the containment boundary when the water is drained. This is accomplished by plugging all the leakage paths into the top housing. The top housing and column are designed to withstand the +/- 6 psi differential pressure The elevator lower column embedded part is already containment boundary.

The major leakage paths are the elevator port and the access holes. There are many minor miscellaneous leakage paths where various actuators, shafts and cylinders enter the top housing.

Depending on the concept for closing the port there are some differences between A&B. The A port is sealed by the transfer mechanism. In B there is a gap between the transfer mechanism and the port. This gap was intentionally provided to allow the operation of a "sniffer" for defected fuel. The B port would have to be sealed with a specially designed bung conforming to the code requirements.

Both A and B top housings have two access holes on top which have to be sealed so the concept would be similar, differing in details

The miscellaneous leakage paths in the top housing of A and B may differ in detail but the concepts could be common,

Therefore the containment concept proposed for A is also applicable for B

4. SEALING SYSTEMS ALTERNATIVES INVESTIGATED

Several alternatives were considered and evaluated before setting out on the design and development of the reference design. Freezing was considered as a common method of isolation used in CANDU plants. However freezing had been unsuccessfully tried on the conveyor tube in the past and the large amount of liquid nitrogen required in a confined space presented a hazard unacceptable to Station operating personnel.

Air-inflatable bags such as used in other industries for isolating conventional piping were considered. This method has several disadvantages. The bag is unlikely to seal the inside corners of the square tube. Worker safety would not be sufficiently guaranteed by the bag which would be subject to sudden loss of air pressure or puncture.

A concept was considered of injecting a thermoplastic material which would harden in place forming a plug. It could later be removed by melting. The drawbacks for this type of system are the complex equipment for installation and removal and the possibility of incomplete removal.

The concept selected was the bung based on an expandable elastomeric seal. This established method of fluid sealing offers a number of advantages in this application. Custom moulded elastomer seal elements can be made to conform to the shape of the tube. Most elastomers can survive radiation doses up to 10^6 rad without losing significant physical properties. Many grades of commercially available material could be safely used in this ambient temperature application. Mouldable elastomeric material can offer more freedom in design than the other concepts considered.

A two point isolation system was devised that is capable of isolating one single elevator. The system consists of two bungs with a linkage and a delivery system. The bungs have double rubber seals with remote actuators. See Figure 4

5. POSITIONING AND ACTUATING SYSTEM

When the design concept for the isolation fixture was established the consideration was given to the method of its delivery to the correct location in the conveyor tube. The minimum dimension of the fixture would not allow its insertion through the openings in the elevator top housing. The fixture could be lowered through the access hole in the floor plate in spent fuel transfer room to the bottom of the elevator lower housing. Once in the lower housing, however, it would be impractical to drag the isolation fixture inside the conveyor tube.

The most convenient means of installing the isolation fixture would be through the bay water, into the conveyor unloader. Cutting a portion of the conveyor tube extension from the bay wall to the unloader was also considered but was deemed unnecessary.

A system was conceived that would permit loading of the isolation fixture from the IFB. The conveyor cart would be driven into the bay where an adapter would be lowered onto the cart . The adapter locks into the slots of the cart. The first bung assembly which has wheels to permit it to travel down the conveyor tube, is attached to the adapter. See Figure 4.

The cart would then move forward to permit the installation of a connector piece and the second bung assembly. Then the complete isolation fixture can be moved down the conveyor tube to the appropriate elevator bottom housing. The connecting piece is of sufficient length that the two bung assemblies can effect a seal on both sides of the elevator housing including the conveyor stop.

The bungs are accurately positioned at the elevator by the irradiated fuel conveyor cart. A solid metal backup system locks the bungs into position to prevent slippage The isolation fixture position feedback is provided by the conveyor cart positioning instrumentation. The conveyor stop is not relied upon because it may be out of service.

The actuating system is still in the early stage of development and concepts such as hydraulic, electric and ballscrew actuators are being discussed.

6. DEVELOPMENT AND TEST PROGRAM

The development of an expanding elastomeric seal to form a watertight seal inside a square tube is unique. The cable lying on the bottom of the tube presents another problem. A small stabilised leak rate is desired.

During the concept study several alternatives for sealing the conveyor tube were looked into. Of the methods considered, the approach based on expanding a solid elastomer seal was selected for further development. The seal configuration was designed in conjunction with a simple expansion mechanism consisting of a movable truncated pyramid plug. By moving the plug axially the seal shape can be modified in radial and axial directions.

During the design of the seal it was recognised that the large radial gap of ½" required a substantial cross-section and consequentially a large force to expand the seal. Additional expansion by a factor of 1.4 will be required at the sharp corners. To avoid expanding a large cross-sectional volume all at once the seal was given a square tubular shape. In the axial direction the wall-thickness was varied from a thin section that allows pre-engagement with the expansion plug to a gradual thickening matching the slope of the plug to provide the bulk for expansion. The thick straight section following behind was tapered on the outside and inside to a flat section suitable for mounting of the seal. The thick section therefore could expand by riding on the slope of the plug but also flex outwards providing bulk for compression. The clamping bars served to prevent the seal material from flowing axially. The seal shape thus was modified by expansion in the radial direction and compression in the axial direction. Figure 5 shows a sketch of the seal cross-section before and after engagement. The seal is designed to be self-locking in the installed condition.

The additional material required at the corners of the seal can be provided in several ways. The most simple manner is by modifying the inner mould shape to give more material on the inside corner of the seal where it rides on the slope. The slopes of the truncated pyramid can be sculptured or the outside dimensions of the corners could be increased. However, both these methods require more complex machining and will be difficult to modify afterwards. Accordingly, for the development work the preferable way of changing the seal shape as required was by modifying the mould. The mould design was modular and accommodated easy modifications by allowing re-machining of parts only. The mould was made from sections of aluminum angle fixed to a base plate in the form of a square annulus. Strips of aluminum bar were machined and attached to outside of the inner square and the inside of the outer square to produce the required cross section

The material selected for the seal was polyurethane. It is a very tough and strong material with a high tear strength. It is one of the better materials for radiation resistance and the temperature range of application is quite acceptable. To keep the expansion loads reasonable the hardness of the seal material was kept in the low (40 to 55) Shore-A durometer range. For the development seals two urethane materials were used; Adiprene L-83 and Conathane TU-4010. Both materials can be cast relatively easily using the same mould. The Adiprene L-83 was polyol cured to lower the hardness to around 50 Shore A. This material also required multi-component mixing and a curing temperature of 100 °C. The Conathane TU-4010 is a two component material which can be mixed, poured and cured at room temperature. The Shore A hardness of 40 can be lowered further with the addition of a modifier.

To test the seals a section of the conveyor tube was constructed and a seal expansion mechanism was designed and built. Figure 6 shows the complete test arrangement. The leakage flow rate was measured at the design differential pressure of 6.5 psi and the accident condition differential pressure of 12.5 psi.

The details of the seal expansion mechanism can be seen in Figure 7. The drive rod and expansion plug are moved axially by means of a double acting nut on a threaded portion of the rod end. This nut is held captive on the end of the support tube. All leakage paths at the plug, drive rod, seal flange and blind flange are sealed with O-rings. The travel of the expansion plug is 2¼".

A number of seals were manufactured from Adiprene L-83 or Conathane TU 4010 materials to Shore A Hardness 55 and 40 respectively. The first seal was moulded with a constant cross-section without allowance for extra material at the corners and without a chamfer or groove to accommodate the cable. This seal provided an initial assessment of the magnitude of the expansion force, required stroke length and flow of seal material with corresponding change in shape.

The second L-83 seal was dimensionally changed to fit better on the mounting flange and the shape was changed by tapering the inside bars towards the corners giving more bulk at that location. The expanded seal did fit tight inside the test tube except for gaps of about 1/16" x ½" at three corners.

To improve material flow and sealing at the corners the next seal was made from the softer Conathane TU - 4010 material. The shape was also modified to accommodate the cable and to give even more material towards the inner corners. This seal was tested at a differential pressure of 6.5 psi and yielded a leak rate of 3.4 IGPM.

To provide visual assessment of the seal shape, a transparent Lexan conveyor test tube was constructed. A 1 inch square grid pattern was drawn on one face of the seal and when expanded the seal first made contact with the centre of the face of the conveyor tube and then the contact area spread out towards the corners. The early clamping at mid-plane caused the seal material to be held there by friction and be stretched towards the corners. This helped to visualise that premature clamping at mid plane prevented full sealing at the corners and it illustrated the need to make first contact at the corners. The grid pattern on the seal face became barrel shaped and showed that seal material was being extruded forward of the expansion plug. The expansion plug, seal and walls of the Lexan test jig were lubricated with a silicone mould release agent to reduce and delay the friction factor. The seal material did indeed flow much earlier towards the corners and the contact area with the conveyor tube was increased especially in the corners. The mould was therefore modified to move the corner material forward in the next seal and at the same time to reduce material at mid-plane. Figure 8 shows the shape at the inside corner of the seal.

The result of pressure testing the final seal made from Conathane material was a leak rate of 0.9 IGPM, all of it at the bottom corner. When the test was repeated with a ¼" solid rod instead of the cable, the leakage rate was reduced to 0.6 IGPM. The seal was leak tight except for a small leak at the upper corner.

The seal was installed again with every effort taken to position it concentric. Tested with the cable in place the seal produced leak rates of 0.4 IGPM in one direction and 0.45 IGPM in the opposite direction. The leak rate at 12.5 psi differential pressure was measured at 1 IGPM while the seal maintained its integrity. This was the test requirement for accident conditions of overpressure. In this condition a higher leak rate is permitted temporarily.

The torque required to drive the expansion plug over the full stroke length was measured at 29 foot-pounds.

7. ACKNOWLEDGEMENTS

The authors would like to acknowledge the contributions by Rathin Chakraborty to the project concept and design phases. Barrie Keal of the Sheridan Park Engineering Laboratory also contributed to the development of the working prototype seal. The overall project was funded by Ontario Hydro's Pickering ND.

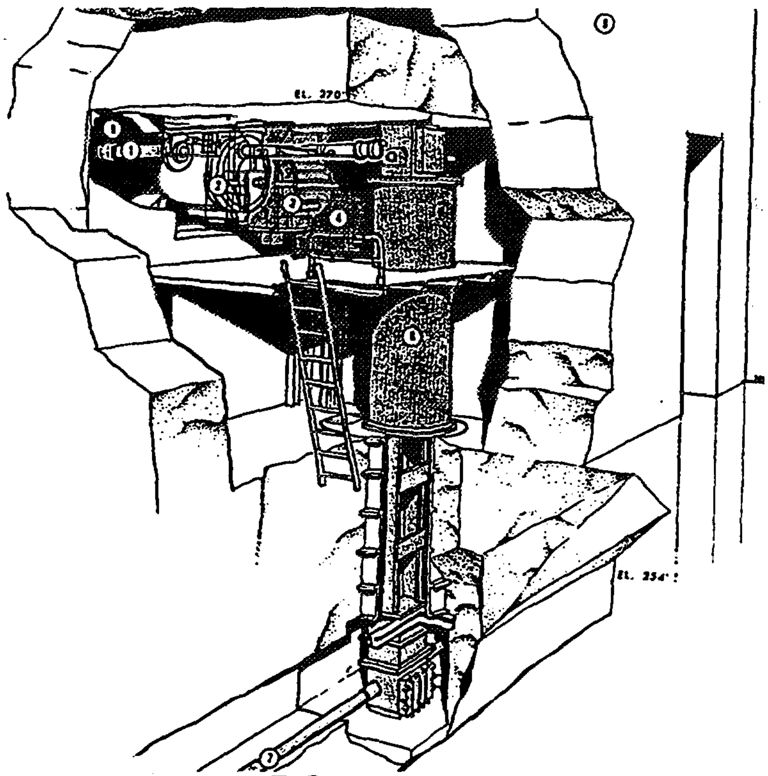

FIGURE 1
FUEL TRANSFER ELEVATOR AND CONVEYOR

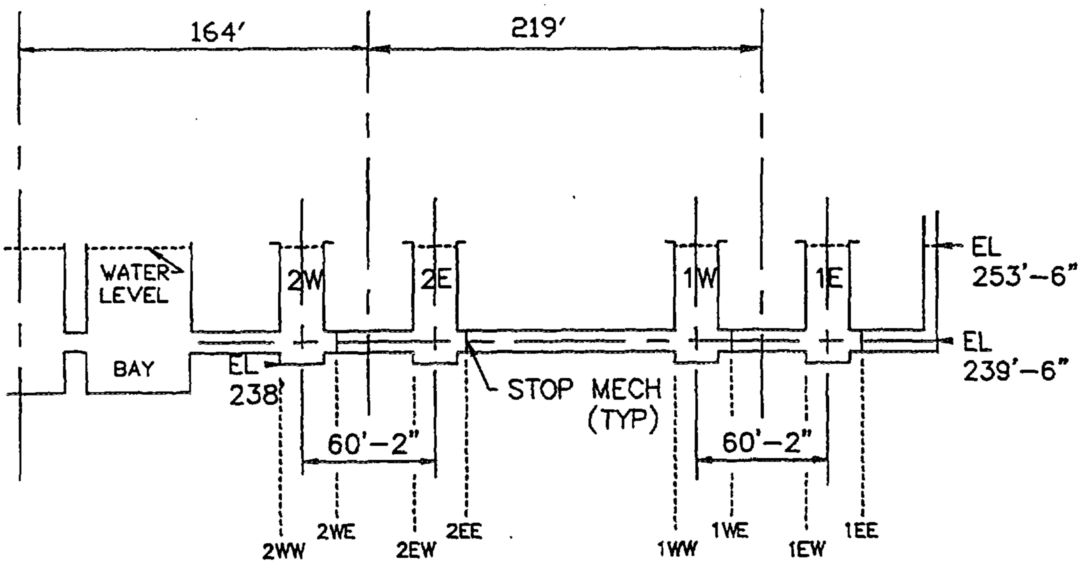

FIGURE 2
CONVEYOR ELEVATION

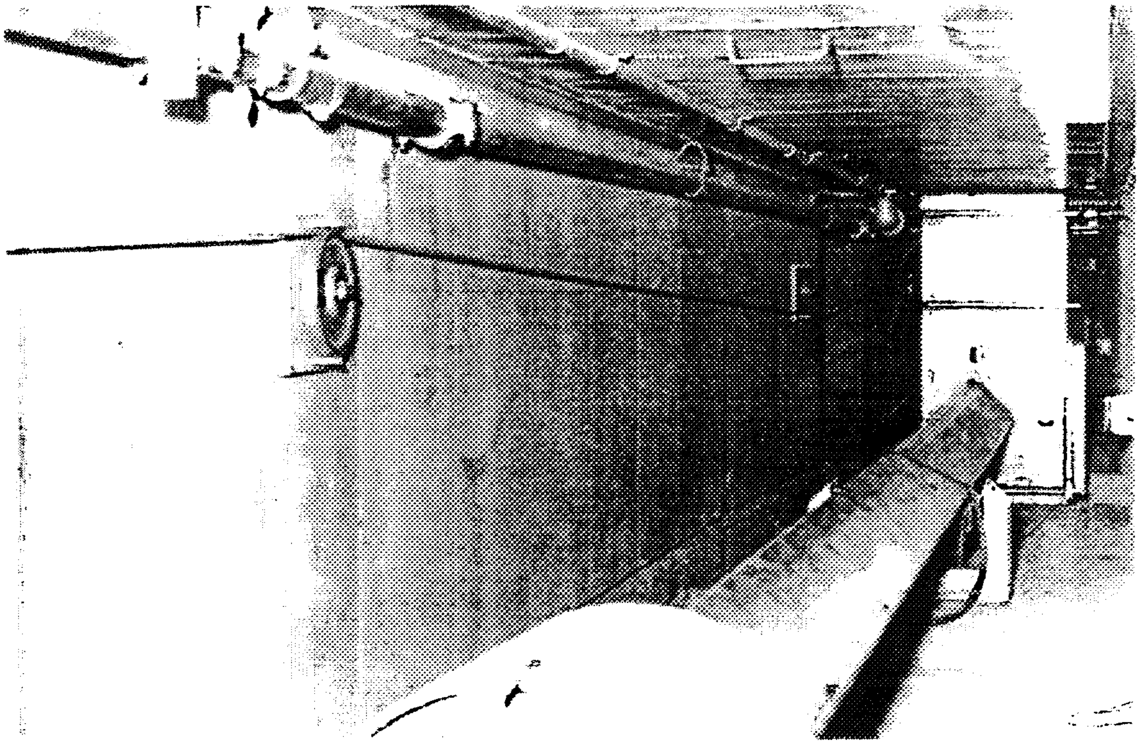

FIGURE 3
PICKERING A CONVEYOR

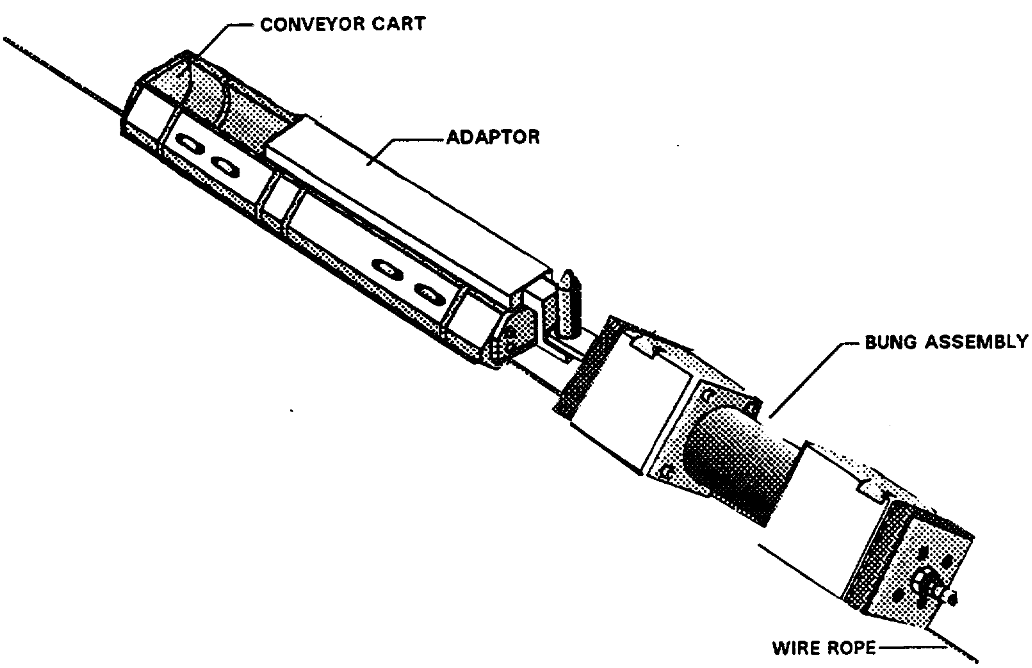

FIGURE 4
CONVEYOR CART WITH BUNG ASSEMBLY

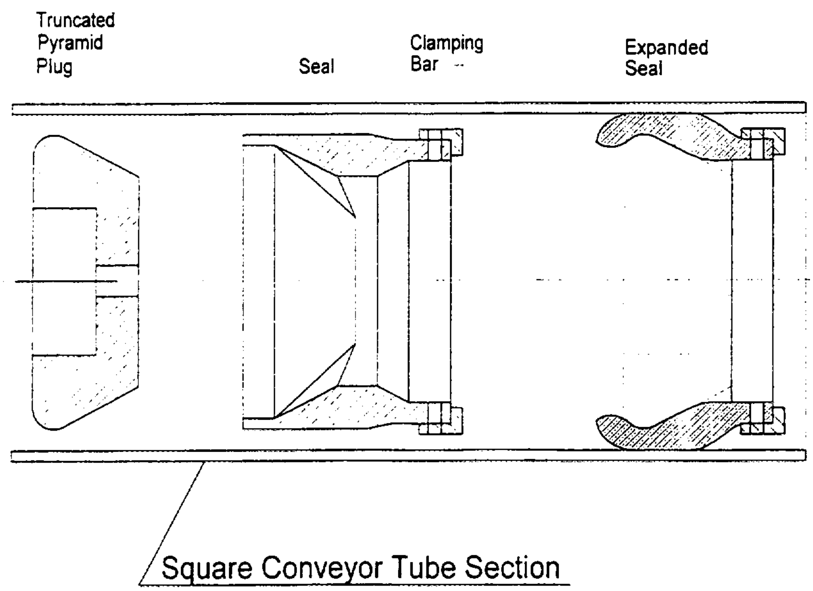

FIGURE 5
CONVEYOR SEAL SCHEMATIC

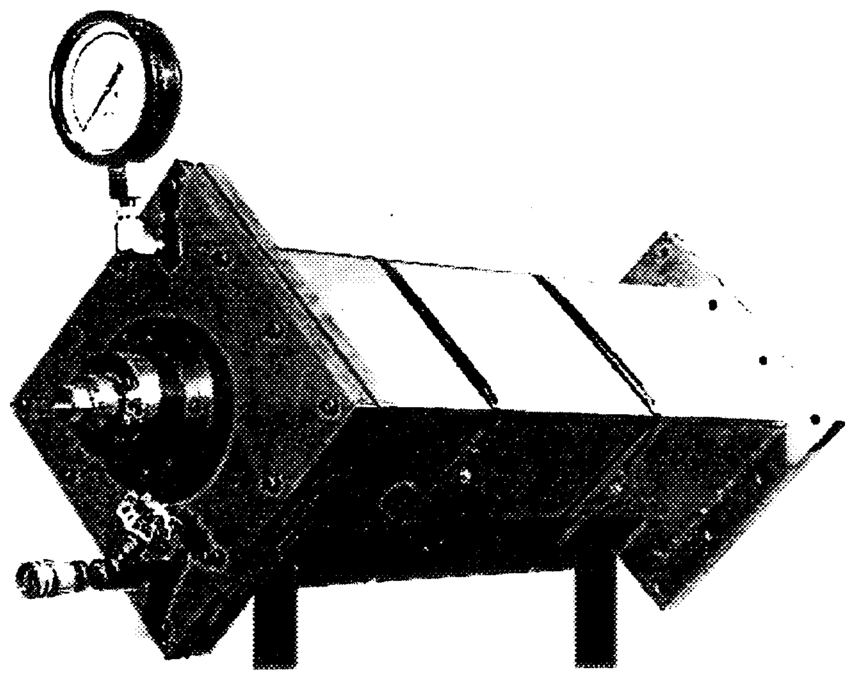

FIGURE 6
CONVEYOR SEAL TEST RIG

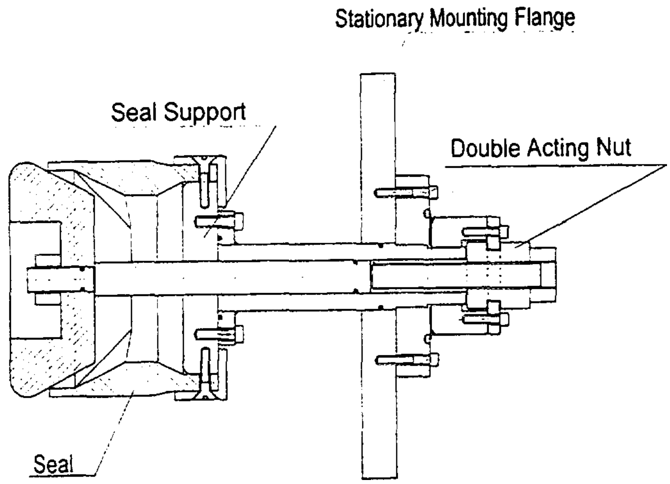

FIGURE 7
CONVEYOR SEAL EXPANSION MECHANISM

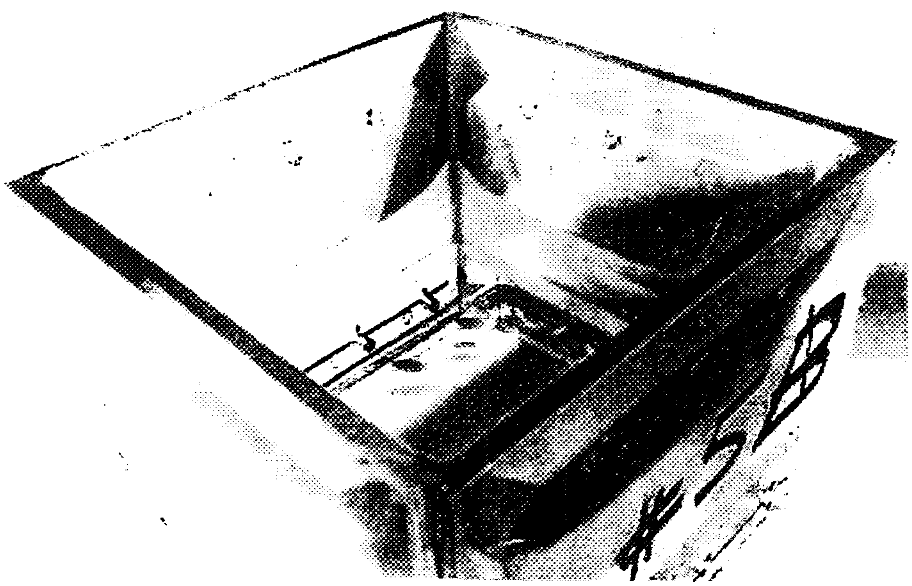

FIGURE 8
URETHANE SEAL ELEMENT

FUEL HANDLING AT CERNAVODA 1 N.P.S.- COMMISSIONING AND TRAINING PHILOSOPHY

CA9700723

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ABSTRACT

Efficient operation of a Candu nuclear power plant depends greatly on the reliable and safe operation of the fuel handling system. Successful commissioning of the system is obviously a key aspect of the reliability of the system and this coupled with a rigorous training programme for the fuel handling staff will ensure the system's safe operation.

This paper describes the philosophy used at Cernavoda 1 N.P.S. for the commissioning of the fuel handling systems and for the training of staff for operation and maintenance of these systems. The paper also reviews the commissioning programme, describing the milestones achieved and discussing some of the more interesting technical aspects which includes some unique Romanian input.

In conclusion the paper looks at the organization of the mature fuel handling department from the operations, maintenance and technical support points of view and the long term plans for the future.

1. INTRODUCTION

In August 1992 a group of 14 Romanian engineers and technologists arrived at the Point Lepreau N.P.S. in New Brunswick, Canada to start an extensive training programme into the operation, maintenance and technical support of the Candu 6 fuel handling system. This programme, which was 11 months long for technical personnel and fifteen months for field personnel, included hands-on operations and maintenance training as well as considerable systems and skills training in the classroom.

At about the same time a group of expatriate Canadians from operating CANDU plants in Canada were assembling in Cernavoda, Romania with other Romanian staff to form the nucleus of the fuel handling commissioning group. Some of the expatriate members of this group were initially assigned to the construction department to provide technical assistance. Working under the direction of the expatriate fuel handling superintendent this group developed the commissioning philosophy, prepared the overall commissioning plan and started on the preparation of detailed procedures. In November 1993 the Romanian trainees returned to Cernavoda to take up their assigned positions within the fuel handling commissioning group.

As well as preparing for system commissioning the fuel handling group were required to support the following associated construction activities:

1) The verification of the Romanian built channel closure installation equipment.

2) The installation of the closure plugs at each end of the fuel channel.

3) The removal of these same plugs for draining

and drying of the fuel channels following the hot performance tests.
4) The manual installation of the first fuel load together with the installation of the shield plugs and once again the closure plugs.

2. COMMISSIONING PHILOSOPHY

In order that the Romanian fuel handling staff acquired as much operating and maintenance experience as possible it was vital that they be part of the commissioning process. To this end 4 composite teams of Romanian technical and operational staff were formed, each supervised by an expatriat. Each team would be responsible for commissioning a number of fuel handling systems together with the preparation of the associated procedures and other documentation. In the early stages all team members worked on the production of documentation, but as construction progressed and turnovers were imminent the field technicians were released from the technical groups and formed into their own operations group to carry out the hands on commissioning work in the field. Initially the operations group was supervised by expatriate with a Romanian deputy who would later become the supervisor of the group.
Once the field work was underway the remaining team members formed the technical unit with each engineer retaining the same systems responsibility to support the commissioning work in the field. As well as leading each team it was the responsibility of each expatriate to train the senior Romanian staff to perform the system engineering function so that as time progressed the Romanian staff could take over the system responsibility thus ensuring a smooth transition as the expatriate staff leave the project.
To gain even more hands-on experience it was agreed with the construction department that the pre-commissioning activities would be under the direction of one of the fuel handling expatriates using the Romanian fuel handling staff to perform the field work. Through the experience gained while pre-commissioning, preparing documents, commissioning and operating the hardware and the training given by the team leaders, this philosophy has provided a solid foundation for the Romanian staff in the design, operation and maintenance of the CANDU 6 fuel handling system.

3. COMMISSIONING ACTIVITIES

3.1 PRESERVATION CHECKS

One of the first tasks to be done was to evaluate the condition of each of the three fuelling machines as they had been under a state of preservation for over five years. This was much longer than the two years storage time expected when the machines were shipped from Canada.
For preservation the fuelling machines were filled with demineralised water and pressurised to 10PSI with nitrogen. The complete assembly was then encapsulated in a foil wrapper which was filled with a nitrogen atmosphere.
A clean room with temporary lighting and a pressurised ventilation system, was established in the warehouse where the fuelling machines were stored. The cold test facility was moved into the clean room and its oil and water systems commissioned to provide adequate, if rudimentary pressure sources to operate the fuelling machine systems.
After removal of the foil encapsulation an inspection was made of the external surfaces which were found to be in good condition. Some decomposing of the pipe sealant used on the oil system pipe threads was observed but this had no effect on the capability of the sealant to perform its function. No sign of any oil or water leaks could be observed.
A simple 110VAC electrical test box was assembled enabling the operation of each oil system solenoid valve to be verified. Samples of the hydraulic fluid and demineralised water were obtained for analysis before operation of the fluid systems. These analyses showed that the hydraulic oil was in good condition but the demineralised water was in poor condition and required changing.
The cold test facility was connected to the fuelling machine oil and water systems and in spite of a serious shortage of tubing and fittings operation of all aspects of the machine were verified. At the same time an inspection was made of the accessible surfaces of the fuelling machine and the components, as they passed into or through the snout. After the inspection the snout plug was replaced and the machine refilled with water to enable a pressure test to be

completed.
The result of this inspection showed a little staining on some of the components otherwise there was no sign of deterioration on the inside of the fuelling machines. The operational checks also verified that all systems were still operating correctly and hence the three fuelling machines were considered ready for commissioning without any rehabilitation. The fuelling machines were returned to the proper storage environment until required for installation.
The D_2O supply pumps for the fuelling machines also required inspecting after being in a state of preservation for about ten years. The preservation for this equipment was not satisfactory as the pumps had been shipped and stored with the plungers installed. Although these plungers had been coated with preservation oil, significant corrosion had damaged the surfaces which had an adverse effect on the life of the pump packing. Although commissioning of the D2O supply system continued with the damaged plungers the frequent seal failures caused frustration which was only solved when the fluid end of the pumps were rebuilt with new seals, plungers and bushings.

3.2 PRE-COMMISSIONING PROGRAMME

Before each system was turned over the pre-commissioning team had identified all system control components, thus identifying deficiencies, completed a wire-by-wire verification of the wiring and cable runs in the system and set up the MCC for each electrical load. All instrumentation devices were calibrated by the pre-commissioning group as part of this programme.
From a mechanical point of view all pressure systems were hydrostatically tested before turnover in a cooperative programme between the construction group and the responsible fuel handling commissioning team. This was also a successful programme as it familiarised the fuel handling team with the layout of each system prior to the start of commissioning and also helped in identifying further deficiencies.

3.3 COMMISSIONING PROGRAMME

A simplified diagram showing the commissioning programme is detailed in figure 1 with the completion date of major activities shown above the activity. The programme spans just over three years from the assembly of the team to the off-power refuelling demonstration. Some of the commissioning work was still outstanding at this time, particularly in the irradiated fuel discharge system where testing was held up due to delays in filling the discharge and storage bays.
Following the turnover the commissioning team completed equipment tagging, flow sheet checks and load verification before energising the system. Similarly in the field the mechanical checks and control valve set-ups were completed, also the electric motors were bumped, when permissable, to verify the direction of rotation.
Pre-operational checks of the control logic and instrumentation were being completed at the same time in preparation for initial operation of the system. During the initial operation the system operating parameters were established and alarms verified. The system was then ready for the final operational tests in which the system performance was verified under all conditions and fully automatic operation established under computer control.
The control consoles and termination racks were the first systems to be turned over and this was in May 1994. It was necessary to have these systems first to ensure that control power was available for all the remaining fuel handling systems. Commissioning of the fuelling machine bridges, carriages and catenary systems together with the fuelling machine auxiliary systems including the F/M oil, the F/M air and the F/M D_2O systems followed in logical order. Commissioning on all these systems proceeded relatively smoothly so that by October 1994 they were ready for the fuelling machines to be mounted on their carriages.
Commissioning of the two fuelling machines also proceeded very smoothly and by March 1995 they were ready for initial operation at the rehearsal facility. This facility is a full scale fuel channel complete with end fittings and shield and closure plugs. The complete assembly is mounted between the two fuelling machine maintenance locks. The channel, which can be pressurised to reactor pressure, is supplied with D_2O from the fuelling machine systems and thus with the exception of high coolant flow, the

rehearsal facility is an accessible fuel channel which is readily available for testing and training purposes. The rehearsal facility contains twelve dummy fuel bundles that are identical to the Candu 6 fuel bundles and these are used for all refuelling operations at this facility. For fuelling machine testing or operator training eight more dummy fuel bundles are placed in one of the fuelling machines with the other machine left empty to receive the fuel bundles from the channel. To compensate for the lack of coolant flow to move the fuel string towards the downstream fuelling machine it is necessary to use the computer refuelling jobs designated for reactor shutdown operation.

It was necessary to have the fuel handling software in a fully operational state at an early stage to support the commissioning of the other systems. To gain experience in the organisations and operation of this software the fuel handling staff assisted the software group in this commissioning process. Some changes to the software were required during the commissioning of the individual systems when some small operational defects were found.

For the initial operation of the fuelling machines the refuelling was completed under computer control in step by step, mode to allow the operators to carefully monitor the performance of each fuelling machine. As the initial operation was very successful a second test was immediately made, but this time the system was operated fully automatically in run mode. The success of this test proved that the fuelling machines and auxiliary systems were capable of fulfilling their design function.

The new fuel system was being commissioned at the same time as the fuelling machines and was ready for operation by April 1995. This allowed a complete test of the refuelling operations to be witnessed by the regulators, except for the spent fuel system which was not yet ready.

The heat transport system was being drained of light water at this time, thus it was also a convenient point to drain and dry the fuelling machines and auxiliary D_2O systems which so far had been operating with light water. This was a tedious process especially for the fuelling machines and the D_2O valve stations with their multiple circuits, each with many smaller tubes. However the operation was successful and the subsequent D_2O fill followed with a minimum of downgrading in the fuel handling system.

Due to the success of the automatic refuelling operations a decision was made to complete the off-power refuelling demonstration during the heat transport system hot performance tests. This was a CANDU first as no other CANDU station has had their fuel handling system in an operational state at this stage of station commissioning. This is a tribute to the Romanian commissioning staff as well as to the AECL designers.

Following the off-power refuelling demonstration the miscellaneous supporting systems such as grappling and special tooling were commissioned. The grappling system was tested at the rehearsal facility and during these tests the operation of the fuel grapples were proven as well as the pressure tube seal and latch mechanism. A further test of the grappling system proved that a channel could be completely defuelled in the shutdown mode using both fuelling machines and grapple extensions to push the fuel into the downstream machine.

in August 1995 the commissioning of the spent fuel system was sufficiently far advanced to allow the fuel to be discharged from the off power refuelling demonstration, but the final test of this system did not take place until December 1995. Following this test, which was also witnessed by the regulators, the fuel handling system was declared ready for the on-power refuelling demonstration scheduled to take place in October 1996.

3.4 COMMISSIONING PROBLEMS

As in any complex system there were some problems to overcome during the commissioning process. Most of these were of a minor nature and were easily solved with a simple mechanical adjustment or wiring change. One of the more complex problems was a pressure instability in the two Romanian designed oil hydraulic power units. The specification called for the use of pressure compensated piston pumps in these units but in this case the improper valve type was used to operate the compensation system. After many tests and discussions with the power unit manufacturer the correct type of valve was finally

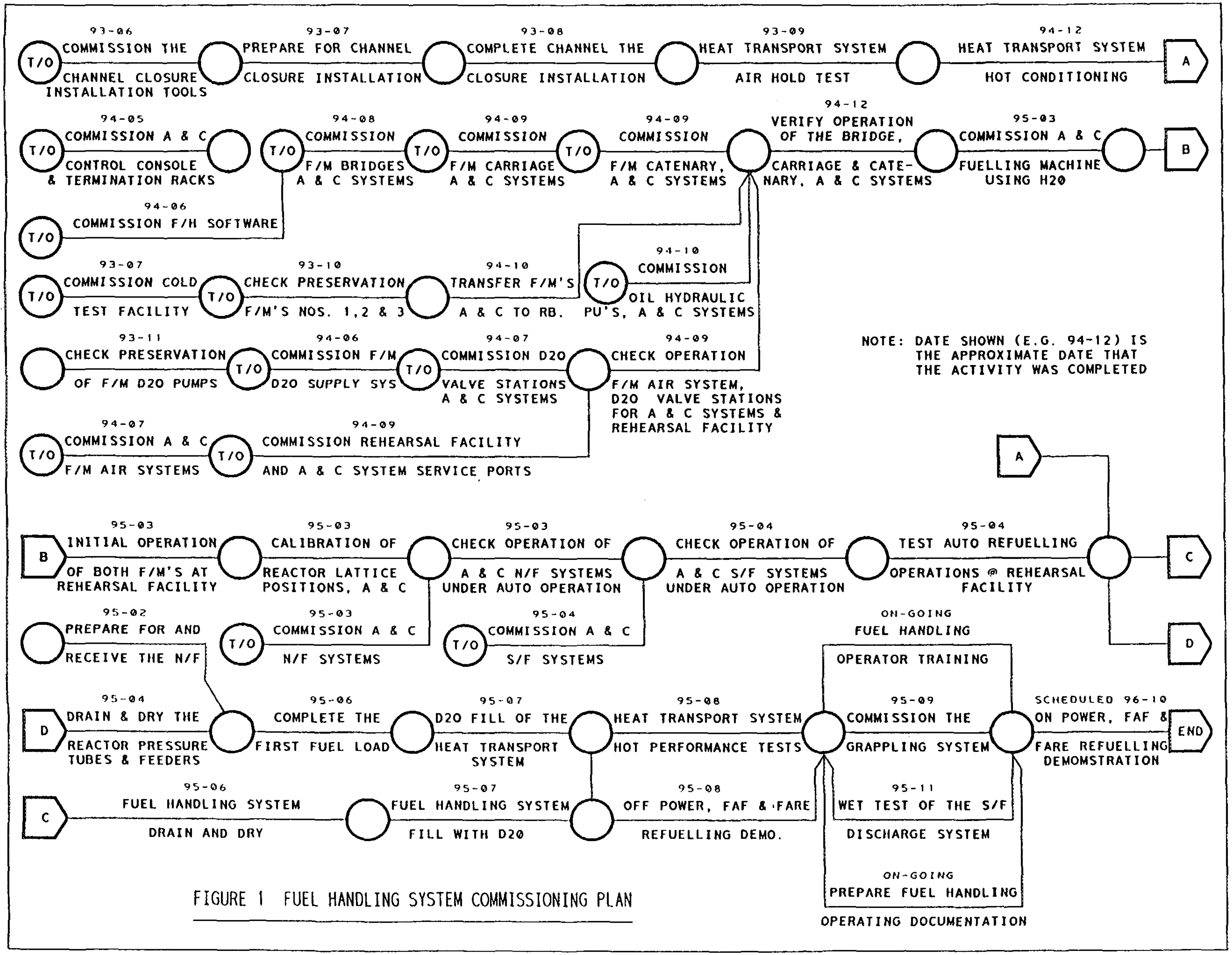

FIGURE 1 FUEL HANDLING SYSTEM COMMISSIONING PLAN

installed and the problem resolved.
A more serious instability problem occurred in the F/M D_2O supply system when a loose connection in the pressure control system resulted in severe vibrations in the piping to the high pressure filters. The resultant pipe movement broke a tube weld at the filter input. While investigating the failure it was realised that the tube was too constrained and a coiled length of tube was used to repair the weld to allow freedom of movement of the filter and pipe without straining the tube. The tubing connections on the other filters in the system were similarly modified.
Although both fuelling machine bridges worked well, some problems were experienced with the brakes as they occasionally failed to release. If this failure went undetected by the operator the brake was very quickly burnt-out and required rebuilding. To overcome this problem each brake was fitted with a limit switch to close on brake release and wired as a permissive in the bridge drive motor control circuit.
The Romanian manufactured bridge ballscrew jacks were supplied with inadequate lubrication to the upper bearings. The jacks operated well in the short term, but because this equipment is not accessible for maintenance with the reactor at full power it was necessary to request the supplier to redesign the gearbox to provide ample lubrication to all bearings . The eight replacement gearboxes have since been received and installed.
On the electrical side the Romanian panel light bulbs on the control panels were so unreliable that it was often difficult to know if a light was on or off, but lit by the warming buss. The bulbs were eventually replaced from Canadian supply. A number of the Romanian designed power supplies were also unreliable and were not suitable for use in the specified application. In some cases a small modification was all that was required to correct the problem, in other cases replacement units were ordered from Canada.

3.5 SUPPORT ACTIVITIES

The fuel handling group enjoyed a good relationship with the construction department and other commissioning groups throughout the project and lent support in many areas as the need arose, of particular interest are the activities associated with the heat transport system and the fuel channels.
The first of these activities was the installation of the channel closure plugs prior to the heat transport system air hold test. Special pneumatically actuated tooling had been designed by AECL for this purpose and manufactured in Romania. The operation of the tooling duplicated the action and forces of the fuelling machine rams while installing or removing the channel closure plugs. Commissioning of this tooling was the first operation and this took place in June 1993. During commissioning of these tools some problems were experienced with the manufacturing tolerances causing difficulties in achieving the correct operating parameters but these were overcome. The tools performed well for this programme but required extensive rebuilding and seal replacement before they could be used later for the removal of the closure plugs. This was partly due to the use of an incorrect lubricant that also damaged the seals.
In order to complete this programme efficiently when most of the fuel handling staff were in Canada, it was necessary to use personnel from the construction, operations and maintenance departments to form the two teams required to do the work. A training programme was set-up to train the teams on the operation of the tooling, the careful handling of the closure plugs, the cleaning and inspection of the end fittings and the verification of the plugs before and after their installation. To make the training more effective a spare reactor end fitting was mounted in the training area and a special test closure plug installed into and removed from this end fitting by the team members. This was repeated until they were completely familiar with each operation.
A platform was mounted on each fuelling machine bridge to provide access for the tooling and operators to the reactor end fittings. Due to the design of the bridge structure it was only possible to access the upper sixty percent of the end fittings from this platform and as in previous CANDU projects, facilities were provided to hang this same platform below the bridge to reach the remaining end fittings. Moving the platforms was a time consuming operation and interrupted the flow of the work part way through the programme. To overcome this problem one of the Romanian senior fuel handling staff requested

that the platform structure be redesigned to have two platforms mounted on each bridge. This design called for one platform to be on top of the bridge and the other platform suspended at a suitable distance below the bridge so that the entire reactor face can be accessed without transferring the platforms. Due to fabrication delays the lower platform was not available for the closure plug installation programme but this innovation was to prove very effective when the same platforms were used for later programmes, when many platform position changes were eliminated, with the resultant time saving for the project.

The channel closure installation programme was carried out in August 1993. It took thirteen days of twelve hour shifts to complete the work including the time lost for transferring the platforms from the top to the underside of the bridges. During the heat transport system air hold test that followed immediately, six closure plugs were found to be passing slightly. In all but two cases the plugs were reseated using manual tooling. The two plugs that could not be reseated were removed and replaced with spares using the pneumatic tooling.

The second support activity was the draining and drying of the reactor fuel channels and installation of the downstream shield and closure plugs prior to the first fuel load. This programme utilised the same pneumatically operated tooling, but this time the channel draining system built into the tool was also required.

By April 1995 when this work was carried out, the trainees had returned from Canada, but it was still necessary to form a similar composite team as this task was part of the first fuel load programme and required considerably more manpower than the fuel handling group could provide. In order to provide extended twenty four hour coverage in two, twelve hour shifts it was necessary to assemble three separate teams.

A dummy fuel channel assembly was set up for training each of the teams for both the drain and dry and the closure and shield plug installation operations. For convenience the manual fuel loading equipment was set up in the same area and training for this operation completed at the same time.

While testing the draining operation it was found that the there was too much of a flow restriction within the channel closure installation tool which caused the draining time for each channel to be much too long. As it was not possible to modify the tool to correct this deficiency, a high capacity, self emptying wet vacuum unit was connected to the tool to increase the flow rate and thus reduce the channel draining time.

The reactor fuel channel draining programme was successfully carried out in ten days. Each team worked in two groups, one at each reactor face and thus, with a set of equipment at each face they were able to drain adjacent channels simultaneously. When all the channels in one row were drained the two groups would work together to pass the drying swabs through each channel after which each end fitting would be drained using the wet vacuum with a special wand attached. To ensure that all the water was removed from the end fitting liner tubes, the water was trapped in a calibrated flask so that the volume retained could be verified. Once the end fitting liner tubes were dried the closure plug region and the external surfaces of the end fitting were thoroughly dried and cleaned, and then the downstream shield and closure plugs installed.

The third support activity was the first fuel load programme which followed shortly after the drain and dry programme. This programme also included the organization of the arrival of the first fuel shipment. Most of the fuel came from Canada although some Romanian made fuel was included in the first fuel load to verify the reliability of this fuel.

During the commissioning of the two Romanian built, pneumatically actuated fuel loading mechanisms, problems were experienced with the speed control of the actuating cylinder during fuel insertion. Some redesign of the control system and control valve modification was required to obtain the correct operating parameters.

The training for the first fuel loading concentrated on the handling and inspection of the fuel, the operation of the manual fuel loading mechanism and the record keeping. The dummy fuel channel assembly was used in the training sessions and each team operated the fuel loading mechanism many times to practise loading twelve dummy fuel bundles into the channel.

The programme started in late May 1995 and was completed in early June. The actual fuel loading process took just under 5 days to complete which was another CANDU first. Once all the fuel was loaded the fuel loading equipment was removed from each face and replaced with the channel closure and shield plug installation tooling. The upstream shield plugs and channel closure plugs were then installed in the end fittings and the equipment and platforms removed from the bridges. Again the success of this programme was largely due to the fact that two working platforms were mounted on each bridge. For this programme each team was divided into three groups, one at each reactor face and a smaller group to provide support to the groups on each face for materials handling and radiation control. To minimise congestion on the platforms the fuel was unpacked and inspected at floor level before being lifted to the platform.

4.0 TRAINING PHILOSOPHY

In any group of people there will always be a range of age, experience, technical capability and dexterity even though all the people would have been selected to perform specific functions with a common interest. This fuel handling group is no exception and thus it is the job of the fuel handling training officer to arrange training for individuals at different levels so that each person can progress to the maximum of their ability.

From a training point of view the fuel handling operations group falls into three categories. The first category is for the six operations staff that were trained in Canada (three others had been promoted) and hence able to take lead roles in commissioning activities. The second category is for the eight people that remained in Romania while their colleagues were in Canada, and thus only qualified to take supporting roles in the commissioning activities. The third category is for the five people that were more recently hired and had minimal or no prior exposure to fuel handling operations. The training programme was set up to meet the needs of all three groups while also having to meet the departments objective of having sufficient fully trained staff ready to safely operate the system by the time the first refuelling is required. At the same time the department still has to complete the day to day operation and maintenance of the fuel handling system.

The training required for the first category was one of refreshment of systems training in the classroom with extensive fuelling machine operations training at the control panel. For the second category more formal systems training was required both in the classroom and in the field, to be followed at a later date with training at the control panel. For the third category in depth systems training will be given in the field, together with skills training, to be followed later with formal classroom training. When sufficient knowledge and experience has been gained the control panel training will commence. It should be noted that in each category the examinations have an 80% pass mark Once the control panel training is complete a final very comprehensive examination has to be passed before the trainee can be given interim authorization to refuel the reactor and then only as a co-pilot for twenty refuelling operations. At this point a practical examination with oral questions is given at the control panel by the fuel handling superintendent and the operations supervisor. If this examination is passed then the trainee is classified as a fuelling specialist, and fully authorised to refuel the reactor. All fuelling machine operations training takes place with the machines on the rehearsal facility, usually under expatriate supervision and once reactor refuelling commences the co-piloting requirement for the category one senior specialists will be supervised by the expatriate staff. Once fully authorised and experienced the senior fuelling specialists will supervise the co-piloting of the fuelling specialists when they are qualified.

The formal training programme started in October 1995 after the intensive commissioning period was over. To avoid a minimum of disruption to the day to day operations the operating group was divided into two teams, each with its own supervisor and leaders. One team was scheduled to be the operations crew for the week looking after the routine operation and maintenance of the system. They were also performing refuelling operations at the rehearsal facility and completing field check outs. At the same time the other team spent the week in the training centre taking the extensive fuel handling systems training. The following week the two teams exchanged roles. For the team at the training centre the first two days of the week were spent attending lectures on a specific system given by the responsible system engineer for that system. On the third morning a comprehensive examination was written after which a second course would be given, with another examination written on the following Friday afternoon. The second team would take the same two courses the following week. The examination was only written by the category one and two students, the category three students attended the lectures as an introduction to the systems, they will study them in depth later. This phase of the training was completed by mid-March 1996 (at the time this paper is being written). For the category one trainees the next phase of training will be concentrated on panel operations. For the category two and three trainees the emphasis will be on field operations training with oral field check outs for each system given by the operations supervisor. Coincident with this training some classroom training will occasionally be given to category one and two trainees on special subjects such usage of the operating manuals.

At the end of April the category two trainees will write a comprehensive examination covering all fuel handling systems. The successful candidates will be classified as fuel handling field specialists and authorised to carry out all the field operations associated with fuel handling. The more experienced field specialists start their control panel training programme as time permits. At this same time the category three trainees will be classified as fuel handling specialists-in-training and commence skills and formal systems training. In mid-June the category one trainees will write a comprehensive examination on the operation of the fuel handling system from the control panels. After this a practical examination with oral questions will be given at the control panel by the fuel handling superintendent and the operations supervisor. The successful candidates will then be classified as fuelling specialists authorised for on-reactor fuelling machine operations under expatriate supervision. Once reactor refuelling commences and their co-piloting is complete this group will be authorised for on reactor refuelling and the more suitable will be selected as senior fuelling specialist

It should be mentioned that throughout the commissioning process the field personnel were gaining experience in all disciplines, is was deliberately organised this way as it is necessary for fuel handling personnel to be cross trained in all disciplines. In the fuel handling group all personnel are expected to be able to fulfil the mechanical, control, electrical and operations functions at any time although some technicians will take lead roles in certain disciplines based on their backgrounds and experience.

As well as the maintenance training received in Canada, training for maintenance activities has been an on-going process throughout the commissioning phase. Usually this training has been given by the expatriate staff and to date this has been very satisfactory and most maintenance activities can now be competently carried out with the Romanian field crews working with the maintenance procedures that are now in place. In some cases the Romanian systems engineers provide additional support to the maintenance activities on their own systems. Certain special procedures such as changing the fuelling machine ram assembly, rebuilding the F/M snout assembly and rebuilding a ram assembly do still require special training which is scheduled to take place soon.

By the time the reactor will be ready for first refuelling this programme will have trained six fuelling specialists ready for control panel operation and eight fuel handling field specialists to give support to the fuelling activities. The field specialists will also be supported by the specialists-in-training and this team will be well able to operate and maintain the fuel handling system at Cernavoda 1 NPS

5.0 THE FUTURE

The immediate future for the fuel handling group is basically to continue the training programme as well as maintaining the system in readiness to support any on reactor operations that might be required during phase B and C commissioning and to prepare for the commencement of reactor refuelling. As the time for reactor refuelling approaches the operations group will be organised into two crews each working eight hour shifts to cover sixteen hours operation for five days per week. Each shift will have two senior fuelling specialists, two fuelling specialists, two fuel handling field specialists and one or more fuelling specialists-in-training. The fuel handling operations supervisor will not be on shift and will not be directly involved with the day to day operations of the crew, this will be the responsibility of the two senior specialists. The fuel handling control panel will be manned by a senior fuelling specialist assisted by a fuelling specialist. From an operations point of view these two men are working under the direction of the shift supervisor.

Work protection will be provided through the work permit system and in the case of maintenance work on fuel handling systems the work protection and work permits will be issued by the senior fuelling specialist at the control panel, after approval by the main control room operator. All electrical and mechanical isolations in the field will be carried out by the field fuelling specialists.

The rest of the crew will be working in the field under the direction of the second senior fuelling specialist. Their task will be to support the fuelling operations and to complete the maintenance activities.

The crews are scheduled to work the day shift one week and the evening shift the following week. After this two week cycle the crews rotate the job functions so that those working on the control panel assume field responsibilities while the senior and one fuelling specialist from the field take over the control panel responsibilities. It can be seen that over a few shift cycles all qualified personnel will spend time on maintenance or field operations and on control panel operations thus forming a highly experienced and specialised team to operate and maintain the fuel handling system. This team approach will add greatly to the system reliability as all qualified members of the crews have a greater understanding of the working of the systems. Thus they can quickly respond to failures, also they can more easily diagnose problems and often anticipate problems before they become more serious, which will ultimately be reflected in the overall efficiency of the station.

In the long term the group will be striving to improve their skills in all areas and to improve the maintenance operations through the development of special tools and fixtures. It is hoped that the inadequate fuel handling maintenance area will be replaced by a purpose built facility that will have sufficient space to rebuild F/M rams and fuelling machines at the same time, as well as providing space for special test facilities.

The fuel handling technical unit as well as supporting the normal system operations will also be looking to the future to see how they can improve the efficiency and reliability of their systems. Replacement of obsolete equipment is one of their priorities, this particularly applies to the oil hydraulic valves most of which are already out of production with spare parts no longer available. They also have to prepare for outage work and to support reactor maintenance programmes such as CIGAR, scrape sampling and single fuel channel replacement. Looking further into the future they must be giving thought to expanding the irradiated fuel storage facility including the possibility of dry fuel storage.

Hopefully the philosophies presented here, when successfully applied, will make the fuel handling group at Cernavoda 1 NPS rank amongst the best in the CANDU group of nuclear power stations.

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FUEL HANDLING SOLUTIONS TO POWER PULSE AT BRUCE NGS A

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ABSTRACT

In response to the discovery of the power pulse problem in March of 1993, Bruce A has installed flow straightening shield plugs in the inner zone channels of all units to partially reduce the gap and gain an increase in reactor power to 75% FP. After review and evaluation of solutions to manage the gap, including creep compensators and long fuel bundles, efforts have focused on a different solution involving reordering the fuel bundles to reverse the burnup profile. This configuration is maintained by fuelling with the flow and providing better support to the highly irradiated downstream fuel bundles by changing the design of the outlet shield plug. Engineering changes to the fuel handling control system and outlet shield plug are planned to be implemented starting in June 1996 thereby eliminating the power pulse problem and restrictions on reactor operating power.

1. INTRODUCTION

1.1 Fuel String Relocation Effect

Fuel channels in CANDU 850 MW reactors each have a 13 bundle fuel string which is supported against the flow at the downstream or channel outlet by a latch. The latch consists of four spring fingers which contact the ends of several peripheral fuel bundle elements comprising the fuel bundle. On-power fuelling is achieved by pushing in fresh fuel bundles against the flow through this latch into the fuel channel, displacing spent fuel out into the upstream fuelling machine.

Initially, the fuel bundle at the inlet end of the fuel channel is partially supported within the skirt of the inlet shield plug. In time, as the pressure tube creeps and elongates, the inlet fuel bundle eventually rests fully in the pressure tube, widening the gap between it and the inlet shield plug. This gap can be as large a 15 cm (6 in).

One of the design basis safety analysis accidents which sets the most demanding requirements on speed and reactivity of the shutdown systems is the large break loss of coolant accident (LOCA). A LOCA can lead to a rapid power increase due to positive reactivity arising from draining of coolant from the fuel channels. This is followed by a rapid power reduction once the trip initiates and the shutoff rods and/or poison injection are activated.

A postulated failure of a reactor inlet header causes a reversal in the channel pressure drop, simultaneously driving the fuel bundles in a large number of channels against the inlet shield plug all in a fraction of a second. This relocation of the fuel bundles to the inlet end introduces positive reactivity as fresher more reactive fuel replaces more irradiated fuel in the central high flux region. The reactivity is proportional to the size of the gap or the relocation distance. Therefore, in addition to the positive reactivity introduced by coolant voiding, the relocation affect causes the "power pulse" to increase in magnitude. Without further confirmation and modeling, the effectiveness of the shutdown systems could only be guaranteed at lower operating power levels.

1.2 Discovery of Power Pulse

Methods of preventing pressure tube fretting at the inlet rolled joint burnish mark was the subject of investigation in the late 70's and early 80's. Studies were initiated to investigate the consequences of removal of the 13th fuel bundle. Consideration was given to the higher reverse impact velocity effect on the integrity of fuel bundles, fuel channels and the calandria structure and limited reactor physics implications. Reverse flow bundle acceleration tests done in 1989, confirmed the potential for rapid fuel string movement. A proposal was made in 1992-1993 to remove the 13th inlet fuel bundle at Darlington to address abnormal fuel support fretting. In response to an AECB question regarding the reactivity effect, analysis of a large break LOCA caused by a postulated inlet header failure for this configuration revealed the significance of the power pulse for a 12 bundle fuel channel. Subsequently, the analysis was extended to address current gaps and found to be a concern for the operating reactors. Until further design review and analysis was completed, Ontario Hydro took a conservative approach where all CANDU 850 MW units, including Bruce A were derated to 60%FP to re-establish acceptable shutdown system margins. Differences in reactor core physics in Bruce B, allowed an uprating back to 80%FP at that station. At Darlington, the reactivity increment was accommodated within available safety margins and the derating was subsequently removed. One contributing factor for Darlington is the dual figure of eight loop primary heat transport system configuration which limits the number of channels affected by a postulated inlet header failure to 120 channels as opposed to 240 channels at Bruce A and B. Another factor of significance is the small gap in Darlington fuel channels due to the short P/T creep life to date.

2. SOLUTIONS

2.1 Background

To eliminate the postulated "power pulse" concern and return the derated reactors back to their high power levels of operation, a multidisciplinary team was setup to define the best alternate design, safety and operational solutions.

Solutions fell into two basic categories; those that served to reduce and/or manage the size of the gap and those that eliminated the insertion of positive reactivity. After evaluation of a number of potential solutions, several options were pursued in parallel. Gap management has been pursued at all three stations, however because of its uniqueness, Bruce A has adopted a separate long term approach.

3. GAP MANAGEMENT

To quickly reduce the gap and allow a commensurate increase in power, two gap management solutions, creep compensators (CC) and flow straightening inlet shield plugs (FSISPs) were considered.

3.1 Creep Compensators

3.1.1 Design

Installation of a cylindrical spacer in the downstream latch serves to relocate the fuel string upstream and reduce the size of the gap at the inlet. The spacer or "creep compensator" varies in length by 1.3 cm (0.5 in) increments, reducing the inlet gap from 3.8 cm (1.5 in) to as much as 15.2 cm (6 in) depending on the amount of creep in the particular fuel channel. The increasing gap size caused by pressure tube creep could be compensated for by installing an incrementally larger size of creep compensator. The cylindrical CC is held in the downstream latch fingers by a shoulder on the outer diameter. The upstream end is shaped to match the outer two rings of a fuel bundle end plate and reacted the 6675 N (1500 lbf) hydraulic load of the fuel string through the downstream end plate of the 1st fuel bundle. The coolant flow goes down through the center or through 10 angled elliptical holes in the wall of the cylinder near the upstream end. The pressure drop across this end of the fuel string actually reduces slightly with the installation of a creep compensator because the flow is effectively redirected away from the latch fingers.

3.1.2 Installation

The creep compensator installation tool occupies a magazine position and functions to install and remove the CC from the downstream fuel channel latch. It consists of a fuel carrier, which is picked up by the fuelling machine ram and charge tube assembly, and is capable of opening the channel latch. Within the carrier is a spring loaded plunger assembly which interfaces with the ram. The front of the plunger is fitted with spring loaded fingers which engage with the CC. During removal of the CC, the ram and plunger are advanced, engaging the fingers with the CC and pushing the fuel string upstream off the latch. The charge tube advances the carrier, opening the latch. The ram is then retracted to a given position. The carrier is then retracted, closing the latch onto a taper on the upstream end of the CC. Retracting the ram then allows the latch to close and support the fuel string. Installing the CC, simply involves advancing the ram and plunger with the CC attached, until the latch fingers close into the groove on the outside of the CC. The ram is then retracted, whereby the spring loaded fingers engaged with the CC are pulled free from the CC by the larger spring force of the plunger. The removal and installation activities requires approximately 10 minutes each to complete.

3.1.3 Outcome

The viability of this concept was jeopardized by the additional 20 minutes of fuel handling time required to remove and reinstall the CC for each channel fuelled and the loss of one magazine position. Fuel handling time was also required to replace CC's with longer ones to account for P/T creep. This 24% reduction in capacity could not be tolerated by the two trolley fuelling system at Bruce A when operating at high reactor powers. The third, or south extension trolley has been removed from fuelling service and dedicated to the fuel channel spacer location and repositioning (SLAR) rehabilitation program. Consequently, this solution was abandoned at Bruce A and at other stations.

3.2 Flow Straightening Inlet Shield Plugs

3.2.1 Design

The existing fuel channel inlet shield plug (MKIIIA) was modified to a MKIIID flow straightening inlet shield plug by crimping a flow straightener in the shield plug skirt. The primary function of the shield plug is to provide radiation attenuation. It is removed during fuelling operations and otherwise remains engaged with the liner locking lug in the fuel channel. The flow straightener is a solid disc made of low cobalt modular ductile iron, identical to the shield plug material, 10.3 cm (4 in) in diameter and 4 cm (1.6 in) thick, with 121 flow holes, 6.3 mm (0.25 in) in diameter (figure 1). This insert provides a flow straightening function which has been proven to reduce flow turbulence and fuel bundle inlet fretting[1]. It also provides a one time reduction of 4 cm (1.6 in) in the size of the gap.

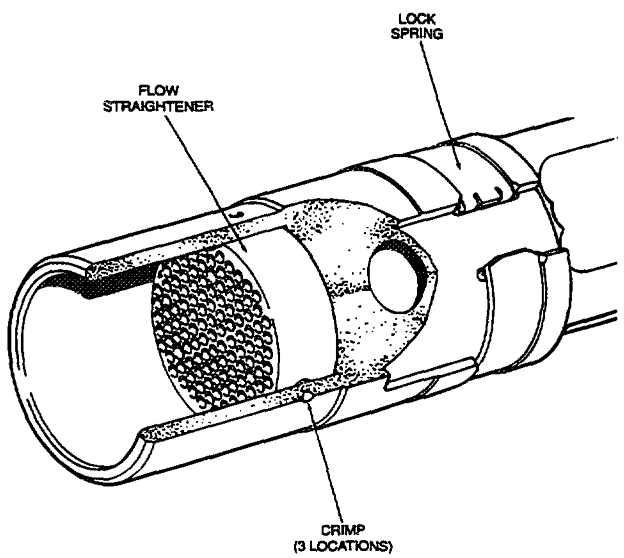

Figure 1: Flow straightening Inlet shield plug.

3.2.2 Installation and Commissioning Operations

Crimping machines used to install and crimp the inserts into the shield plugs were installed and commissioned on the fuel bay gantry.

The FSISP crimping campaign was initiated by taking a dedicated trolley to the target unit, where each fuelling machine (F/M) head installed 8 new inlet shield plugs complete with crimped insert and removed 8 regular inlet shield plugs, taking approximately 1 hour per channel visit. These 16 new FSISPs were removed at the end of the campaign and stored for use on the following unit and finally sent for disposal. One control room fuel handling operator operated the dedicated trolley. Two F/M heads were locked onto the ends of the channel for each inlet shield plug removed or installed in order to perform a pressure drop measurement ensuring no flow blockage. Channel outlet temperatures were also monitored for the same purpose. The trolley was then taken to the central service area and the F/M heads locked onto the east and west ancillary ports. On the bay side, one crimping machine was aligned with one of the ports and prepared to receive the shield plugs. On average, 2 field operators worked in the fuel bay area, operating the crimping machines. The fuelling machine ram and charge tube assembly picked up a shield plug and pushed it through the port to the crimping machine. An flow straightener insert, cooled in liquid nitrogen, was then remotely installed into the skirt and 3 crimps were applied to the outside of the skirt. The flow straightener was therefore secured by the combination of the shrink fit and the crimps. The plug was then retrieved back into the fuelling machine head and the process was repeated for the next and remaining shield plugs in that head.

When all 8 plugs were modified, operations were completed for the shield plugs in the opposite F/M head. The trolley then returned to the unit with each of the 2 F/M heads carrying 8 modified shield plugs which were then swapped for regular shield plugs. The process was repeated until all 280 inner zone fuel channel inlet shield plugs were modified.

3.2.3 Outcome

Early in the installation campaign on the first targeted unit, after approximately 1 week of operations, the crimp depth on test coupons (1 test coupon every 8 shield plugs) began to wander out of specification. A delay of 2 months was incurred resolving this problem.

Typically, the process of installing 16 FSISPs and removing 16 regular shield plugs took 16 hours. The fuel bay crimping operations for the 16 shield plugs took approximately 4 hours. On average, accounting for shift change and breaks, the rate of processing was 80 shield plugs per week. The total effort (engineering and operations) associated with conversion of 4 units required approximately 12,000 man-hours with a dose expenditure of 1.97 rem.

Flow Straightening Inlet Shield Plugs installed in the 280 inner zone channels of all 4 units at Bruce A permitted an increase in reactor power from 60% FP to 70% FP. Subsequent additional safety analysis permitted a further increase to 75%FP. Table 1 shows the dates for completion of installation of FSISPs and subsequent increases in reactor power.

3.3 Long Fuel Bundles

To eliminate the "power pulse" concern, two long term solutions; long fuel bundles and fuelling with the flow (FWF) were devised. Bruce B opted for the long fuel bundle solution, thereby controlling the gap, while Bruce A chose to eliminate the problem by adopting FWF. Lower channel flow rates made the FWF option undesirable at Darlington.

3.3.1 Design

Loading a fuel channel with a combination of regular fuel bundles and bundles that are 1.3 cm (0.5 in) longer than regular fuel bundles (long fuel bundles) results in a fuel string length that can be managed to minimize the inlet gap and position the 13th bundle bearing pads away from the burnish mark. This solution therefore addresses the power pulse problem as well as preventing bearing pad interaction with the highest stress area of the pressure tube. The

Unit	3	1	2	4
FSISP Installation Complete Date	January 1994	February 1994	March 1994	July 1994
Reactor Power to 70%FP	March 1994	March 1994	April 1994	August 1994
Reactor Power to 75%FP	October 1994	October 1994	October 1994	October 1994

Table 1 - Flow Straightening Inlet Shield Plug Installation and Power Increase Dates

design of the long fuel bundle is identical to the regular fuel bundle except the distance between the center bearing pad and the intermediate bearing pad on each side of the bundle center line is increased by 6.3 mm (0.25 in). This design has been tested in-reactor and approved for use.

3.3.2 Installation

Although it is desirable to minimize the average core gap from a power pulse perspective, it is also necessary to ensure that the gap in any one fuel channel does not get smaller than a specified minimum gap size. A postulated large break LOCA causes the fuel string to expand and therefore reduce the gap. If the gap is too small prior to the event, the fuel string expands into compression which may lead to pressure tube failure. The gap management system must therefore continually function with high reliability to ensure that the average core gap is maintained low enough to prevent power pulse yet not exceed the allowable minimum gap size in any one fuel channel. Channels are fuelled with sufficient long fuel bundles to prevent fretting on the burnish mark and reduce the gap at the inlet. During the normal course of fuelling, spent long fuel bundles removed from the channel must be replaced and additional long bundles installed to account for pressure tube creep. However at the same time, the inlet gap in each fuel channel must be maintained greater than the minimum allowable gap size. This management system impacts a number of work group activities i.e. new fuel loading, spent fuel storage, generation of fuel change orders, fuel handling.

3.3.3 Outcome

Pressure tube fretting at Bruce A has been identified as a concern, however, levels of fretting are more severe at sister stations. The installation of the FSISPs in the inner zones of all units will help to minimize fretting damage.

The core physics at Bruce A is different than sister stations. The unique peaked neutron flux makes the response to fuelling operations more difficult to simulate and reduces the ability to accurately predict in advance, fuelling sequences long enough to allow for the required control of long fuel bundle fuelling.

Also, the requirement for a minimum gap has great potential to limit the maximum achievable power output from the Bruce A units. A minimum gap size 3.2 cm (1.25 in) would be required to achieve 92%RP which is the full electrical equivalent of the Bruce A units. The currently approved minimum gap size is 5.1 cm (2 in).

This gap management solution utilizing long fuel bundles was not pursued at Bruce A. However, Bruce A has pursued a contingency plan involving the loading of one long fuel bundle at one time into all inner zone channels to gain a temporary reduction in average core gap without violating the minimum gap requirement of 5.1 cm (2 in). This would permit an increase in reactor power to 80%FP for a period of time. Although approval has been received to proceed with this program, Bruce A has chosen not to implement it at this time.

4. REVERSAL OF FUEL BURNUP PROFILE

4.1 Fuelling With The Flow

Changing the direction of fuelling from against the flow to with the flow, can be used to maintain a reversal of the fuel burnup profile. The most irradiated or spent fuel bundle would reside at the outlet end of the fuel channel and the new fuel at the inlet. Therefore, given the postulated inlet header failure, the relocation of the fuel string would introduce negative reactivity as irradiated fuel would replace newer, more reactive fuel in the central high flux region. The size of the inlet gap would no longer be of concern and the remaining power pulse caused by voiding could be negated by the shutdown systems. There has also been some indication that having new fuel at the inlet of the channel does not cause fretting damage to the same extent as highly irradiated fuel[2]. After reviewing the alternatives and performing business case evaluations, Bruce A has committed to resolve the power pulse problem by pursuing reordering the fuel bundles and FWF.

4.1.1 Design

Prior to implementing FWF, it is necessary to reverse the order of the fuel bundles in the fuel channels. Without doing this, the most irradiated fuel would be pushed back through the core, causing a loss in reactivity and subsequent shutdown. In addition, new fuel string supporting shield plugs were designed to prevent fuel bundle end plate cracking during a hot shutdown once the channels have been reordered. These are discussed in following sections.

Software design changes to the fuel handling control system, namely the controller software, protective software, operations and sequences, control console, termination cabinet and wiring are required to implement a fully automatic operation of the FWF process.

Control software changes have been made to permit synchronization of the rams based on encoder inputs. A channel status map is contained within the control software, to record the state of various channel parameters. The status for each fuel channel is updated as changes are made such as new shield plugs installed, channel reordered and channel ready for FWF. Also the map is automatically interrogated prior to a change being made, in order to verify that the change is appropriate. This verification step serves as one of at least two barriers to performing erroneous operations. The fuelling machine ram encoder signals have been directly wired from one control 'quadrant' to the controller of the other fuelling machine on the trolley, to provide better ability to track the ram motion of the other machine for FWF. Additionally, signals have been wired between the systems which indicate status of ram brakes and clutches and facilitate ram and fuel position sensing. Power supply and brake release logic changes have been made to ensure the brakes will hold the fuel string during a Class III or IV power failure.

New outputs have been assigned to command the redundant brakes added for fuelling machine ram and charge tube axial and rotary drives. These F/M signals are used in the control and protective logic to prevent applying excessive forces to the fuel string. Protective software changes include additional interlocks to ensure latch operation and the fuel is not damaged. A separate set of protective logic has been provided for 'abnormal' or maintenance operations.

The operating data includes new sequences and operations that define FWF operation. The sequences for fuelling against flow have been retained since during the transition to FWF, both modes of fuelling will be required.

Console panels have been modified to add the protective normal and abnormal indications and the redundant ram, charge tube axial, and charge tube rotary brakes. Interlock bypass switches have been allocated for FWF and charge tube rotary interlocks. All operations executed by the control computer, as well as those executed manually while the controller is active will be logged. This information will be used for confirmation of proper completion of operation if manual intervention becomes necessary.

The fuelling carriers required modification to permit them to open the latches and a taper is required on the back of the ram head to allow it to return through the latches.

Basically all design work was done by General Electric Canada and commissioning and testing done by Bruce A Fuel Handling Operations. Bruce A Engineering Services Department functioned as the design authority.

4.1.2 Installation, Commissioning and Testing

In order to permit installation and commissioning of FWF without adversely affecting the north and south trolleys from normal fuelling or to adversely affect fuel handling outage support work several initial changes were required. The purpose of these changes was to have the trolley quadrants converted to a state where changing from FWF commissioning mode over to regular fuelling mode on a frequent basis could be done expeditiously. These changes were declared in-service in May 1995.

Commissioning work plans were prepared based on GE commissioning test specifications by Fuel Handling Technical. The work plans were implemented by the Fuel Handling Operations shift crews. Completed commissioning work plans were reviewed by Fuel Handling Technical and GE.

Due to the demands on Fuel Handling systems and resources to provide regular fuelling for 4 units at 75%FP with 2 trolleys, subsequently reduced to 3 units, support outage work such as defuelling channels for CIGAR inspections and mini-SLAR operations, supply fuel handling operators to SLAR operations, defuel unit 2 and perform preventative and forced system maintenance, progress on commissioning FWF software has been delayed. A total of approximately 13,000 man-hours of engineering and operations labor have been used to date to perform FWF commissioning related activities which commenced in late 1994.

4.1.3 Status and Plans

A full set of interlock checks and follow-ups have been completed on the south trolley. The north trolley system interlock checks are scheduled for May 1996. Controller software checks, follow-ups and tests have been completed. North and south trolley protective checks have also been completed.

The FWF control and hardware changes were tested on the Maintenance Area Test Facility and the Service Area Rehearsal Facility. These facilities provide the necessary mockups to test the interface between the fuel handling system and fuel channel components. An on-reactor test is also planned. Two outer zone channels will be fuelled with new fuel (done to reduce contamination if fuel damage occurs) and FWF operations will be fully tested.

Hardware modifications to the fuel handling carriers and ram heads were completed at site with components designed and supplied by GE.

Fuel Handling operations staff have been trained. Operating documentation has been revised. The Operations ECN walkdown and design authority approval is expected to be completed in the second quarter. Submissions have been made to the AECB and approval is expected prior to the putting FWF in-service by the end of the second quarter.

4.2 Reordering

Reordering reverses the burn-up profile in each channel so at the next fuelling visit, the channel can be fuelled with the flow. Bruce A undertook extensive design and safety analysis to support the selected reorder scheme. In addition, a reorder trial was performed to test the selected reorder process.

4.2.1 Design

The selected reorder scheme is a 12 bundle cascade which is done against the flow. The process is started by displacing 12 fuel bundles in the seed channel with new and depleted fuel. The fuel from the seed channel is then loaded into a neighboring channel, in order, with least irradiated first to most irradiated last. The displaced fuel is then loaded into the next channel and the cascade continues for approximately 14 channels. The final channel loaded is the original seed channel. The displaced new and depleted bundles are then used to fuel for reactivity gain as usual. The Bruce A Fuel and Physics section are responsible to conduct SORO simulations predicting maximum channel and bundle powers during the reorder push of the selected channels to be reordered.

4.2.2 Testing

Reordering of fuel bundles using the 12 bundle cascade scheme while at power was analyzed to show that the degree of power ramping of high burnup bundles would not lead to increased fuel defects. However, a reorder test was done to verify the conditions and parameters under which production reordering will be performed. In August of 1995, fuel bundles from 12 channels containing F3SPs were successfully reordered in unit 4. The conclusions were that the test confirmed predictions and verified limits pertaining to the reactor regulating system, maximum channel power peaking factor, maximum channel power, neutron overpower protection and fuel performance. A FWF Reorder Specification has been prepared incorporating test results which specifically describes the conditions and parameters under which reorder will be safely performed.

4.2.3 Plans

It is planned to reorder the 280 inner zone channels on each unit at a power level of approximately 70%FP before returning to complete the 200 outer zone channels for safety and economic reasons. Based on the reorder trial on unit 4, it is estimated that the reordering of one unit will take approximately 9 weeks. A dedicated fuel handling trolley will be used to reorder batches of approximately 14 channels at a time, with about 1 to 2 days of regular fuelling between batches. The fuelling for reactivity gain between reorder batches will be either fuelling against the flow or fuelling with the flow. Therefore, reordering will not commence until FWF has been declared in-service and new fuel string supporting shield plugs installed.

4.3 Fuel String Supporting Shield Plugs

4.3.1 Design

The most significant impact of reordering and FWF, is the relocation of the most irradiated fuel bundle from the inlet to the outlet. The outlet bundle is supported by the latch fingers reacting against the outer elements of the most downstream fuel bundle and is subjected to the highest hydraulic load (6675 N). Although fuel bundle end plate cracking due to PHT pump pressure oscillations had not been seen in Bruce A reactors, a test was done to confirm the integrity of highly irradiated bundle end plates in this new configuration. In the fourth quarter of 1993, 12 fuel bundles were positioned in the reverse flow direction into fuel channel position #1 after cycling through the reactor in Unit 2 and left there for 7 weeks of operation. They were removed from the core after a hot shutdown. External visual inspection of the end plates showed no through wall cracking. However, a short time after the completion of the design review for FWF, scientists at Ontario Hydro Technologies warned against the potential for delayed hydride cracking in the highly stressed fuel bundle end plate during a hot shutdown. A total of 6 fuel bundles from the test and 2 control bundles were sent to Chalk River Laboratories for destructive examination. The 6 bundles displayed numerous cracks, up to several millimeters into the spigot to end plate weld. As a result, a change to the design of the outlet shield plug was made to fully support the outlet fuel bundle end plate, reduce the stress and prevent cracking.

A nosepiece was designed to be added to the front of the body of the existing outlet shield plug to fully support the end plate of a fuel bundle against the hydraulic load on the fuel string (figure 2 see next page). The 410 stainless steel nosepiece is a cylindrical arrangement which fits over the end of the existing cast iron shield plug body, increasing its length by approximately 7.6 cm (3 in). It is attached to the body by a special bolt which limits the nosepiece to rotation on a thrust bearing. The back of the nosepiece is keyed to the shield plug liner locking lug during installation and removal thus preventing relative motion between the nosepiece and the fuel bundle. A formed leaf spring in the back of the nosepiece provides a detent between the nosepiece and the shield plug body. This ensures that nosepiece orientation, relative to the shield plug, will be correct during installation.

New shield plugs have been manufactured and delivered to site. It is planned to replace the existing outlet shield plugs with the new F3SPs. A shield plug handling system designed by GE, has been installed and commissioned in the fuel bay. This system is used to handle and load the F3SPs and old shield plugs. Modules, each holding 8 F3SPs were designed for transportation, storage and handling. The module consists of 8 horizontally arranged tubes which are welded to a handling frame. Locating lugs in each tube serve to control the orientation of the F3SP which facilitates pickup by the fuelling machine ram and charge tube assembly.

The modules are handled in the fuel bay by a module handling tool which is suspended from the fuel bay crane. The handling tool provides shielding and functions to attach to the module frame and rotate in a horizontal plane. The fuel bay crane has been modified to permit remote infrared control to reduce radiation dose to the operator. Modules are lifted by the handling tool from the 615' elevation up to positioning

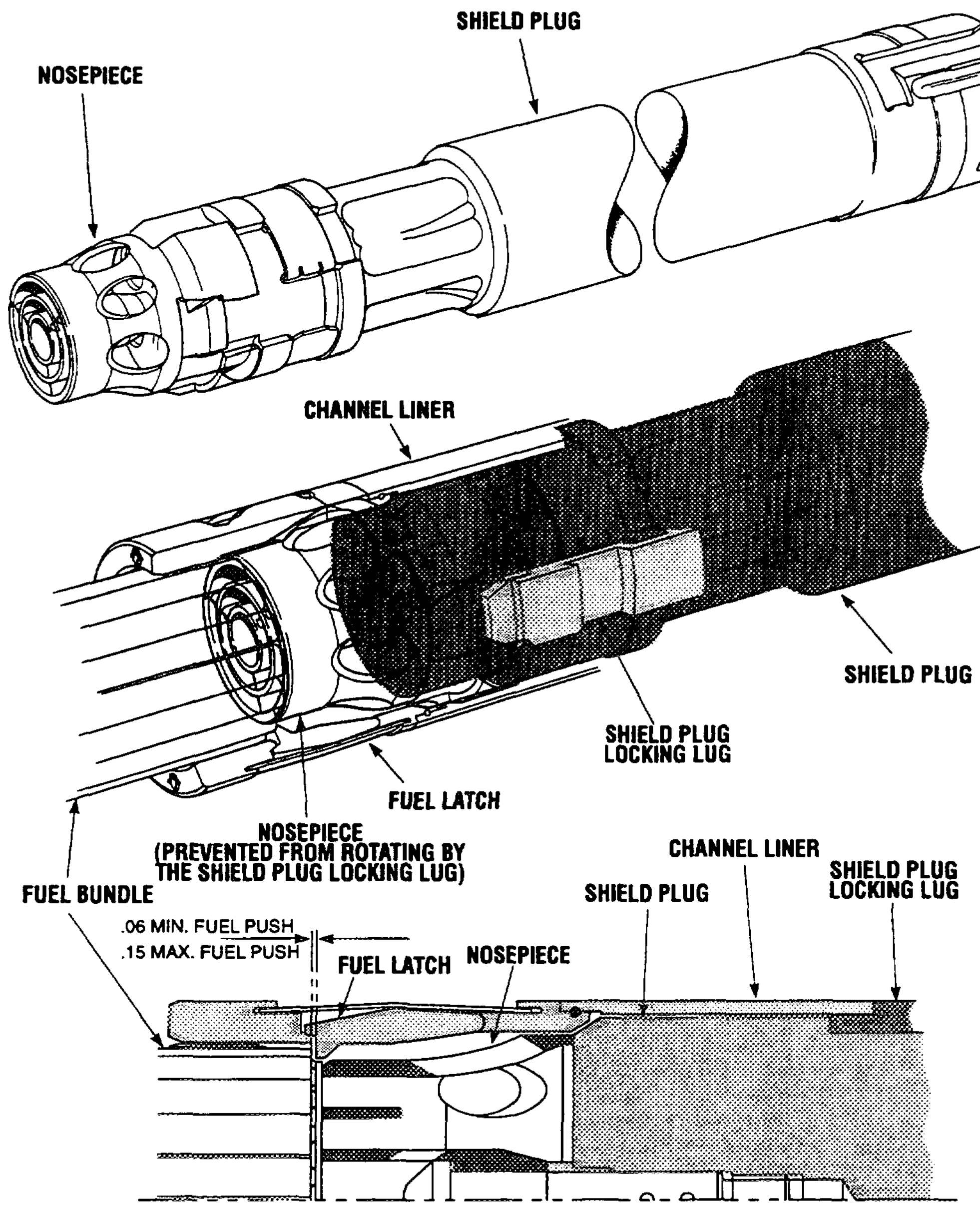

Figure 2: Fuel string supporting shield plug.

mechanisms, mounted on each end of the fuel bay gantry. The positioning mechanism supports and shields a module and positions the module tubes with respect to the ancillary port. The positioning mechanisms are also operated by remote infrared controllers.

The fuelling machine exchanges old shield plugs for new F3SPs at the ancillary ports. The modules containing the old shield plugs are loaded into shielded storage containers located on the 615' elevation. Each container holds 14 modules (112 shield plugs) and when full will be transported to the Radioactive Waste Operations Site (RWOS) for storage in concrete trenches. Lifting beams have been provided to handle the containers by both the fuel bay crane and the crane used at the RWOS. The module handling tool is also used to handle the shielded lids on the container. Specially designed portable steel tanks (1.2 m wide by 0.8 m thick by 1.9 m high) containing water are used to provide additional shielding for the storage containers and the control center. A number of video cameras have also been strategically located to facilitate operations when handling the old radioactive shield plugs. The control center is located at the extreme north end of the fuel bay so that both the activities on the 615' elevation and at the fuel bay level can be seen by the operators. Video monitors, infrared controllers for the fuel bay crane and positioning mechanisms and electrical controls for the module handling tool are located at the control center.

4.3.2 Testing and Installation

An extensive out-reactor testing program was conducted to verify the performance characteristics of the new shield plug and the impact on interfacing components such as the fuel channels components and fuel. Parameters tested included pressure drop, vibration, strength and crudding. Functional tests were performed on MATF to confirm the fuelling machine and fuel channel interface. A total of 14 prototype F3SPs were used in a delayed hydride cracking test in-reactor prior a unit 4 outage in March 1995. Each prototype supported highly irradiated fuel bundles which were subject to a hot shutdown. These bundles were then defuelled and inspected by a specially designed ultrasonic end plate inspection tool developed at Ontario Hydro Technologies. Underwater fuel bay inspections using this tool showed no cracking. The results were confirmed by sending a number of these fuel bundle elements with controls to Chalk River Nuclear Laboratories for destructive examination. Controls were selected from the previous unit 2 test bundles which displayed cracking.

Changes to fuel handling control software necessary for the installation and removal of the F3SP were included in the latest release of the FWF control software. Specific opdata testing has also been completed. The channel status map will also be used to monitor the installation of the F3SPs and serve as a barrier to preventing erroneous operations.

The shield plug handling system has been installed and commissioned. Training of fuel handling operators is complete.

4.3.3 Plans

It is planned to start installation of F3SPs into the 280 inner zone channels of the first unit this summer and complete all three unit inner zones by the end of the year. This is a prerequisite to reordering. A dedicated trolley will be used on a continuous basis to remove the old shield plugs and install the F3SPs. Based on experience with the FSISP program, it is estimated that the inner zone of one unit can be completed in 2 weeks. Once all inner zones are complete, the 200 outer zone channels will be converted by the first quarter of 1998 for ease of configuration management.

5.0 CONCLUSIONS

The impact of the power pulse problem on the production of electricity at Bruce A over the past 3 years has been significant. Although almost all station work groups and several Ontario Hydro service organizations have been affected to some degree, the Fuel Handling section has experienced a extraordinary demand on human and equipment resources. Not only were solutions born, but developed and ultimately will be implemented by them. The engineering changes to the fuel handling system to convert to fuelling with the flow were interrupted in August 1994 with the discovery of the end plate cracking problem. The need to replace outlet shield plugs with a new design has added to the work load and resulted in delays to the schedule. It remains as a challenge to Fuel Handling in next two years, to implement power pulse solutions by installing F3SPs, reordering fuel bundles and placing FWF in-service, in addition to supporting SLAR outages and providing normal fuelling in order eliminate reactor operating power restrictions imposed by the power pulse problem.

6.0 REFERENCES

1 M. Trelinski, "Darlington NGS Unit 4, In-Service Report, DGS-IR-4-03641.3-03, May 1995 In-Service Inspection of Fuel Channel Pressure Tubes Using PIPE and OPIT", August 1, 1995.

2 D. Jarron, "Darlington NGS Unit 4, In-Service Report, D-IR-4-31110.01 Rev 0, August 1993 In-Service Inspection of Fuel Channel Pressure Tubes Using PIPE", January 21, 1994.

FUEL STRING SUPPORTING SHIELD PLUG (F3SP) FOR ONTARIO HYDRO - BRUCE NGSA

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ABSTRACT

A reactor "power pulse" problem was identified for the Ontario Hydro Bruce generating stations. On a postulated inlet header break, the fuel strings in a large number of channels could relocate toward the upstream end, resulting in a power pulse.

The solution adopted for Bruce GSA is to change the direction of fuelling, from against the flow, to fuelling with the flow. In this revised fuelling scheme, given a postulated inlet header failure, the fuel bundle with the highest burnup would relocate into the reactor core and introduce a negative reactivity during the accident. However, this fuelling configuration results in a highly irradiated fuel bundle residing in the most downstream position against the latch. The latch supports only the outer ring of elements, not the end plate. A resulting high stress on the end plate coupled with high levels of hydrogen and deuterium may result in Zr hydride assisted cracking in the end plate during hot shutdown conditions. (In fuelling against flow, this is not a problem, since the latch supported bundle is not irradiated and has only low levels of hydrogen and deuterium.)

A fuel string supporting shield plug (f3sp) which supports the bundle end plate has been developed as a solution to the fuel bundle end plate cracking problem. It would replace the existing outlet shield plug in all channels.

This paper will describe the f3sp design, associated fuel handling operation and qualification for reactor use.

1.0 DESIGN (SEE FIGS 1A and 1B)

The f3sp uses the existing outlet shield plug design, modified to include a nosepiece that protrudes through the latch to fully support the end plate of the most downstream fuel bundle. All other functions of the shield plug are retained in the f3sp;eg, diverts the primary heat transport system flow from the pressure tube to the end fitting/liner annulus and provides shielding against radiation streaming.

The downstream fuelling machine removes and installs the f3sp during each channel fuelling cycle (while the upstream fuelling machine removes and installs the existing inlet shield plug). When installed on the shield plug locking lug, the nosepiece extends forward enough to support the fuel string, resulting in a minimum fuel string push of 0.06 inches off the latch supported position.

Main components of the nosepiece assembly are; fuel support plate, nosepiece body, attachment bolt, thrust bearing, and detent spring. The assembly is installed complete on an outlet shield plug (s/p) body with the special attachment bolt. The s/p body requires machining to accommodate the new nosepiece, additional to the design of the standard outlet shield plug.

A major design intent was to prevent relative rotary motion between the f3sp and the fuel during s/p installation/removal operations using the f/m, because the fuel is unproven under these conditions (see also section 2.0). Since the Bruce style fuel channel and fuel handling system was based on a rotary motion for shield plug lock/unlock the design evolved as follows. Relative rotary motion between the fuel bundle and the nosepiece during s/p installation and removal is prevented under normal operation by a thrust bearing between the nosepiece and the shield plug body. The effective friction torque radius at the bushing is less than half that at the nosepiece/fuel bundle interface, biasing rotation at the bearing rather than at the fuel. This bearing also limits the amount of torque that can be transmitted to the fuel string. In the event of binding;eg, crud build up, relative motion is limited by keying the nosepiece to the shield plug locking lug.

The fuel support plate and nosepiece body are joined during assembly to form the subassembly that rotates relative to the remainder of the f3sp. This subassembly is radially supported on the attachment bolt with axial contact against the thrust bearing. The bolt and bearing are locked to the shield plug body at assembly. Further detail component descriptions follow.

The fuel support plate is held captive on the bearing portion of the attachment bolt. The fuel support plate of type 410 stainless steel is 1.00 inch thick with a flow hole pattern similar to a fuel bundle end plate. The outer, intermediate and inner rings on the fuel support plate are raised to restrict contact with the bundle end plate to these areas giving full support to all three rings of 37 element fuel. The inner ring of the fuel support plate is extended towards the s/p and provides the interface with the attachment bolt. Only the center fuel element is unsupported. (The outlet shield plug for 500 - 600 Mwe reactors only supports the outer two rings of the bundle end plate.) The fuel support plate is attached rigidly to the nosepiece body by an interference fit and single plug weld while two axial dowel pins are also placed at the joint diameter to carry rotary loads.

The nosepiece body of type 410 stainless steel is a cylindrical arrangement supported off the outer diameter of the fuel support plate. Two axial projecting wings extend outboard from the nosepiece and mate with two sections milled in the outer diameter of the s/p body. The wings key on the shield plug locking lug when the s/p is being installed or removed. In the event of binding between the nosepiece and the thrust bearing due to crud build up or similar restriction, this keying limits the potential relative rotation between the nosepiece and the fuel bundle to 3 degrees maximum.

The attachment bolt of type 410 stainless steel serves as the axle for the fuel support to turn on but restricts axial movements of the nosepiece relative to the s/p. The bolt shaft diameter, at the interface with the fuel support is chrome plated for wear and low friction reasons. It is shrunk and threaded into the nodular cast iron s/p body and provides a cantilever support for the fuel support. The bolt is locked by two radial pins of stainless steel. These pins are a press fit and are further retained by coining their access hole.

The type 410 stainless steel used in the majority of the nosepiece assembly was chosen for its corrorsion resistance, high strength and lower activation properties compared to other high strength stainless steels.

The small cylindrical thrust bearing of Nitronic 60, having good anti galling properties, is installed on the attachment bolt and carries the full axial load of the fuel string under static conditions or during rotary motions when the f3sp is installed or removed.

A formed leaf spring of inconel 718 rivetted into a slot in the larger of the nosepiece body wings provides a detent between the nosepiece and the s/p body. The purpose of the detent is to prevent rotation of the nosepiece relative to the s/p when the s/p is not locked on the liner lug. This ensures that nosepiece orientation, relative to the s/p, will be correct when the s/p is reinstalled. The spring is attached to the nosepiece by a clamp plate and inconel rivets.

To prepare the s/p body for installation of the nosepiece the following machining is required on the inboard end of the s/p (additional to the design of the standard outlet s/p); a 5/8 - 11 UNC tapped hole in the center of the inboard end face for the attachment bolt, a 0.748 inch diameter hole leading into the tapped hole for attachment bolt location, a 0.850 inch diameter counterbore leading into the 0.748 diameter for location of the thrust bushing, a spotface is also provided to give a quality mating surface for the bushing, two sections are milled on the outside diameter at the inboard end of the s/p in which the nosepiece wings sit - one section covers about 94 degrees while the other section covers about 136 degrees, in the larger of the two milled sections two grooves are machined to mate with the detent leaf spring. The standard shield plug design of the bayonet slot and c-spring arrangement is retained so that there is no change to the way it is held in the fuel channel.

2.0 OPERATION

Compared with the existing s/p, operations associated with installation or removal of the f3sp are similar since the existing shield plug bayonet slot and c-spring arrangement is retained. A design intent of the f3sp is to minimize any rotation against the fuel and the operating sequence was developed accordingly:

- The f3sp is picked up by the f/m and is unlocked from its f/m magazine site. The f3sp is advanced into the fuel channel with the nosepiece aligned to pass over the shield plug locking lug. The advance is halted at the position shown in figure 2-1.
- The f/m then rotates the f3sp clockwise to stall against the shield plug locking lug. See figure 2-2. This precludes any rotary motion of the f3sp against the fuel string during installation. During removal, if the f3sp rotary bushing is seized, there could be a maximum 3 degrees of rotary motion, in a counter-clockwise direction, against the fuel string (this is the clearance between the nosepiece and the liner locking lug). This potential rotation of the downstream fuel bundle has been shown to be acceptable. (Immediately after removal, the fuel bundle pair interfacing the f3sp are removed from the channel and discharged to the irradiated fuel bay.)
- The f/m then advances the f3sp at slow speed to stall axially in the fuel channel against the fuel latch cage. See figure 2-3. During this motion the f3sp will pick up the fuel string from the fuel latch and thus the f3sp and f/m are carrying the fuel string hydraulic load for the remainder of the sequence.
- The f/m then retracts the f3sp a small amount from the hard stall position. See figure 2-4.
- The f/m then rotates the f3sp about 45 degrees clockwise to stall, to lock on the shield plug locking lug,. See figure 2-5.
- Finally the f/m is disengaged from the s/p and the channel closures can be installed prior to the f/m's unclamping from the fuel channel. Once the f/m unlocks from the f3sp the hydraulic load on the fuel string will push the f3sp back into hard contact with the shield plug locking lug but the fuel will still be a minimum of 0.06 inches off the fuel latch. See figure 2-6. The f3sp makes no contact with the pivoting fuel latch segments and thus they remain closed under their spring bias (see figure 1A).

The same basic cycle is repeated in reverse for removal of the f3sp. The f/m stalls into the f3sp to pick it up. The f/m is retracted a small amount. The f/m rotates the f3sp about 45 degrees counterclockwise to stall to unlock from the lug. The f/m retracts the f3sp from the fuel channel to its temporary f/m magazine storage site and the fuelling sequence continues (the fuel string will be hydraulically pushed into contact with the fuel latch until the f3sp is reinstalled at the end of the fuelling sequence). Note that bundle support by the latch is acceptable under normal fuelling conditions. It is only if this condition exists during a hot shutdown process that the potential for hydride assisted cracking occurs.

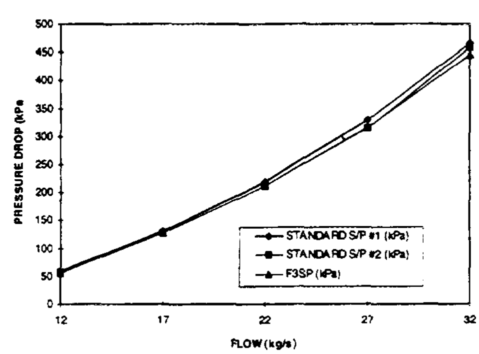
3.0 QUALIFICATION TESTING

The following qualifications were performed in various test rig channels representing the Bruce fuel channel geometry.

3.1 Pressure Drop

This testing was carried out in a representative test channel at GE Canada Laboratory in Peterborough, Ontario.

The f3sp pressure drop was compared to that of the standard outlet shield plug. The intent was to make the pressure drop similar. The nosepiece geometry is in part a result of this development testing. The final test results showed that for the reactor operating conditions of 300 ° C temperature, 9.5 MPa pressure and 27 kg/s mass flow, the f3sp pressure drop was within about +/- 7 kPa (+/- 1 psi) of the standard s/p, depending on the particular manufacturing tolerances of components used in the test comparison.



3.2 Fuel Latch Vibration

These tests were carried out using a representative test channel at Stern Laboratories in Hamilton, Ontario.

When the f3sp is installed it supports the fuel string off the fuel latch, leaving the latch segments unloaded except for a light spring preload. Fuel latch segment vibrations were checked at flows up to 50 kg/s which verified that the motions were insignificant (less than 0.8 μm).

Similarly, the f3sp nosepiece vibration was checked in a defuelled channel at flows up to 50 kg/s (this would be the case in a defuelled channel). Again motions were verified insignificant (less than 1.2 μm).

3.3 Strength Testing

The f3sp was subjected to several abnormal loadings to verify its integrity. All testing was done using pertinent sections of fuel channel of Bruce design. The loads were applied using a tensile test machine. All were completed acceptably without any loss of function or significant permanent deformation. These tests were carried out in the GE Canada laboratory.

The f3sp was pushed back against the shield plug lug at loads up to 35000 lb to represent an outlet LOCA condition where the full system pressure is applied to the f3sp.

The f3sp was stalled hard forward against the fuel latch cage at 10000 lb to represent the abnormal forward axial stall of the component in the fuel channel during s/p installation using the f/m.

The f3sp was stalled hard forward against a flat plate at 10000 lb to represent the abnormal forward axial stall of the component in the ancillary port during f/m pick-up.

The f3sp nosepiece was purposely snagged in a fuel channel endfitting liner tube while pulling on the outboard end to a load of 10000 lb to represent the abnormal retract axial stall of the f/m in an abnormal load case were the f3sp nosepiece seizes in the fuel channel during axial withdrawal from the channel by the f/m.

The f3sp nosepiece was purposely seized on its axle while torquing the nosepiece at loads over 360 ftlb to represent the abnormal stall of the f/m in an abnormal load case where the f3sp nosepiece seizes during f/m removal or installation.

The f3sp nosepiece was fully rotated back and forth on its bushing while under an axial load of 2000 lb while immersed in water to represent the normal loading seen by the thrust bushing when the f3sp is installed or removed from the fuel channel while supporting a fuel string. Three hundred rotation cycles were completed.

The above testing verified f3sp strength as documented in a stress report utilizing finite element analysis which also covered normal and abnormal load cases.

3.4 Long Term Flow Tests

These tests were carried out in the representative test channels at GE Canada Laboratory in Peterborough, Ontario.

The purpose of these tests was to verify that the f3sp lock/unlock mechanism remains operable during the extended time period that the s/p sits statically in the channel between f/m visits. The test was also used to record any damage to fuel, fuel channel or f3sp during the test duration, such as from vibration. Two tests were completed, both for a 3 month duration, one at 300 °C, 10 MPa and flow of 28 kg/s, the other at 300 °C, 10 MPa and reduced flow of 0.5 kg/s. The higher flow conditions represent typical on-power conditions while the lower flow represents maintenance cooling flow. Neither of these extended tests resulted in the nosepiece seizing on its axle and f3sp/fuel/pressure tube inspections before and half way through the high flow test indicated that wear of these components was virtually zero.

3.5 F/M Functional Tests

These tests were performed by Ontario Hydro at Bruce A on their representative test channels located in the maintenance area test facility. One of the station production fuelling machines was used.

The purpose of these tests was to verify the fuelling machine/fuel channel interface and the specific f/m computer control. The fuel string hydraulic load was simulated by spring loading the bundle residing against the fuel latch (against the f3sp when installed).

The tests were completed in a cutaway channel so that the operating motions could be verified correct.

Abnormal operating motions were also tested to verify f3sp integrity during forward axial stall in the fuel channel with the f3sp purposely misoriented to prevent passage over the liner lug and to verify that f/m rotary motions away from the shield plug lug can not detent the f3sp nosepiece.

3.6 Vibration Testing

These tests were performed based on Ontario Hydro test specifications using a representative test channel at Stern Laboratories in Hamilton, Ontario.

The purpose of this testing was to show that the vibrational characteristics of the f3sp are not detrimental to the fuel channel nor fuel nor itself. This was done by comparing fuel/fuel channel vibration in the existing design (latch supported) with the shield plug supported design.

3.7 Reactor Prototype Tests

Fourteen f3sp's were loaded by production f/m's into Bruce unit 4 to demonstrate that their fuel support function would prevent cracking of the end plate during a hot shutdown. This was shown to be the case. Prior to their installation the particular channels had been partially reordered to locate the 13th bundle (an irradiated bundle) from the inlet position to the outlet position to be supported by a f3sp.

Additionally the f/m channel pressure drop data recorded during operation on these channels showed no excessive differential pressures across the fuel string and shield plugs which can be attributed to the f3sp and thus no measurable effect on the hydraulic characteristics of the fuel channel is expected.

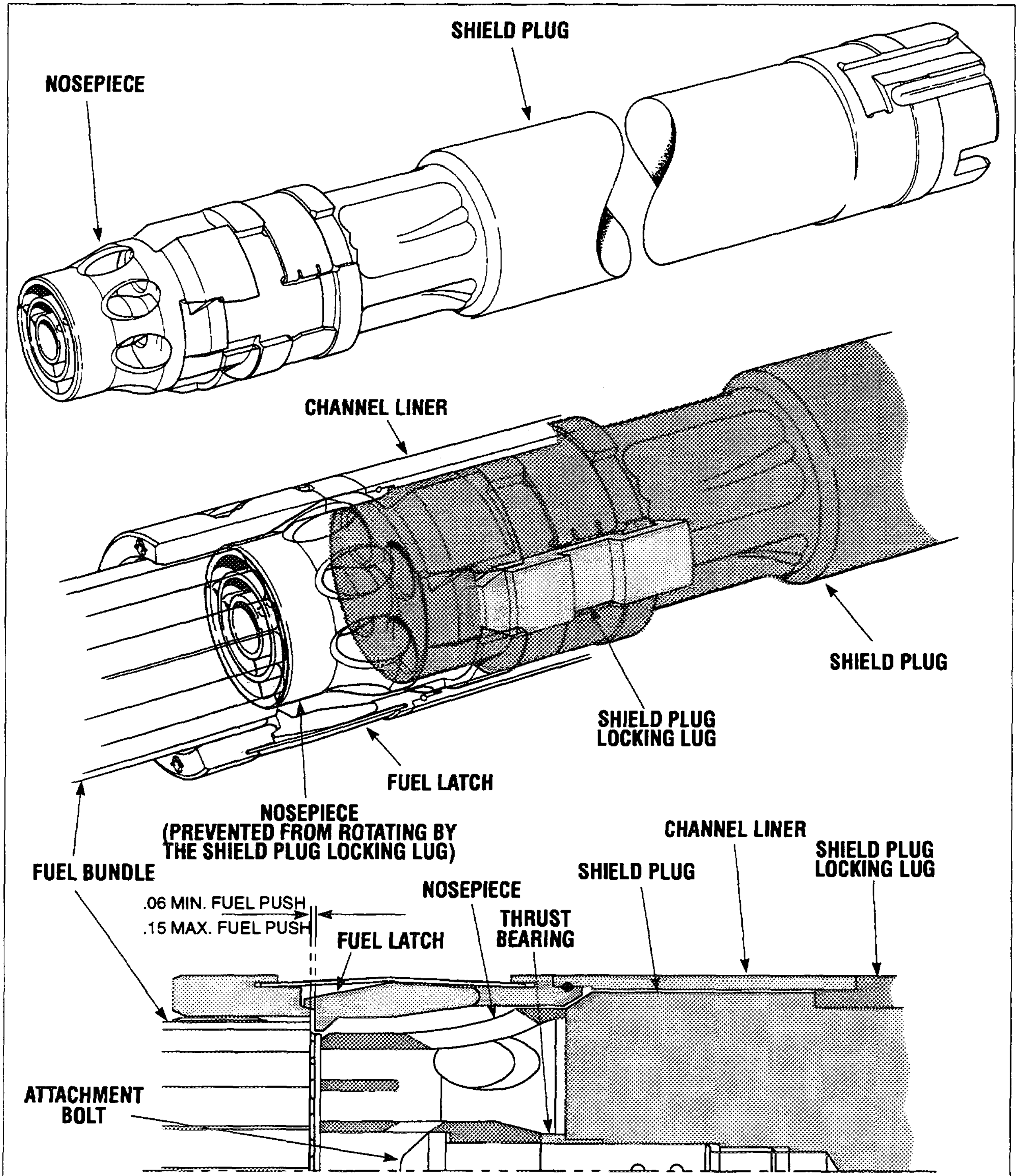
4.0 CURRENT STATUS

It is planned to begin installing production f3sp's in the reactors at Bruce NGSA in 1996.

5.0 ACKNOWLEDGEMENTS

The author wishes to thank Ontario Hydro BAND and GE Canada for their approval and support in preparing this paper.

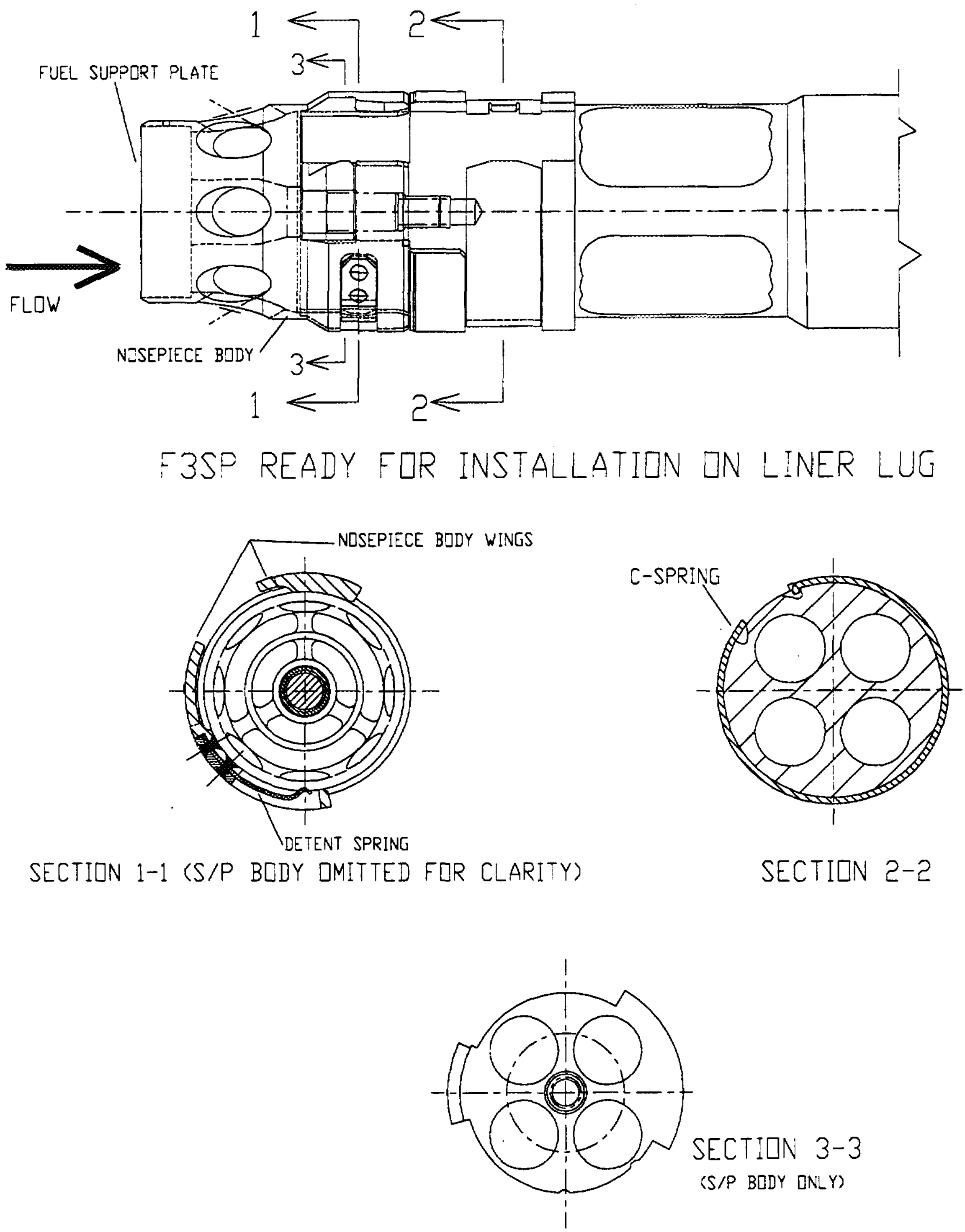
FIGURE 1A– FUEL STRING SUPPORTING SHIELD PLUG



FUEL STRING SUPPORTING SHIELD PLUG ROTATED TO LOCK INTO LINER SHOWN AT CTR 61.5° (1026_B)

CG

FIGURE 1B - FUEL STRING SUPPORTING SHIELD PLUG



2.0 OPERATION: F3SP INSTALLATION SEQUENCE

FIGURE 2-1

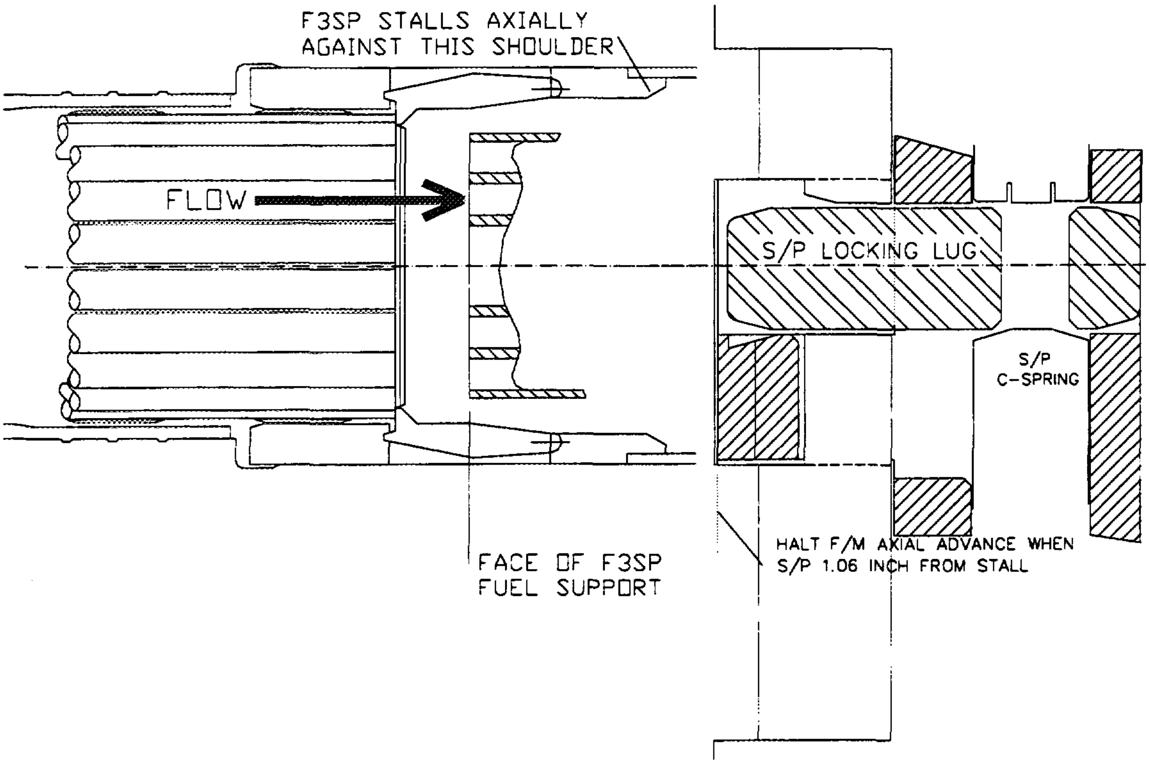


FIGURE 2-2

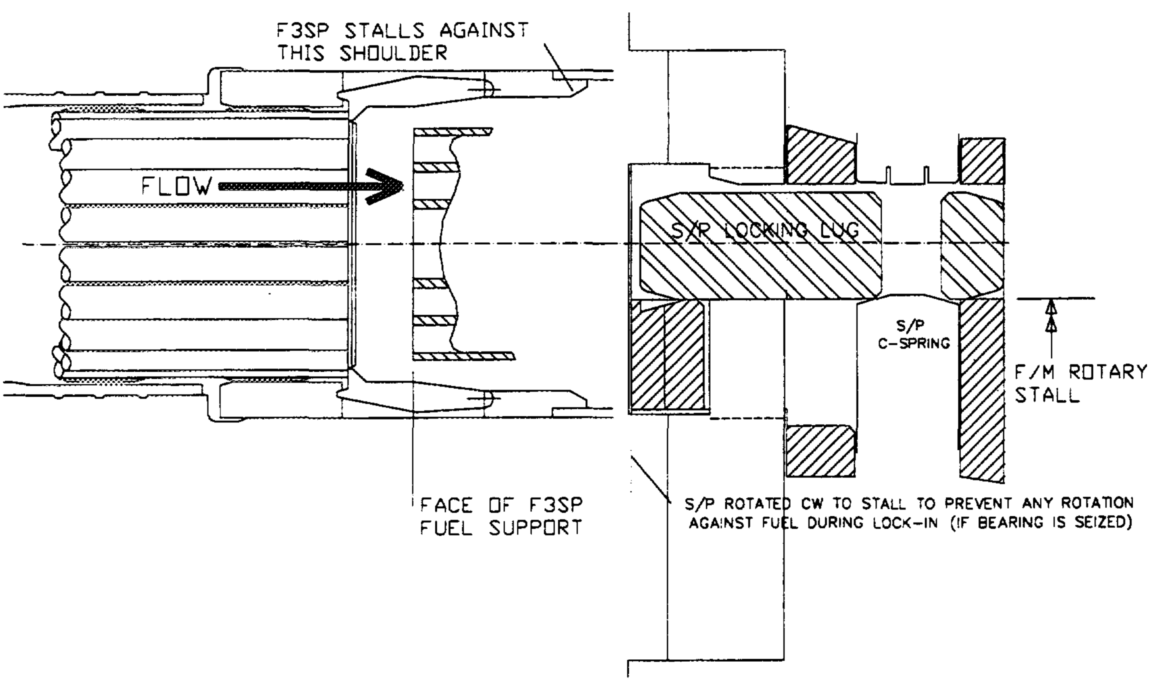


FIGURE 2-3

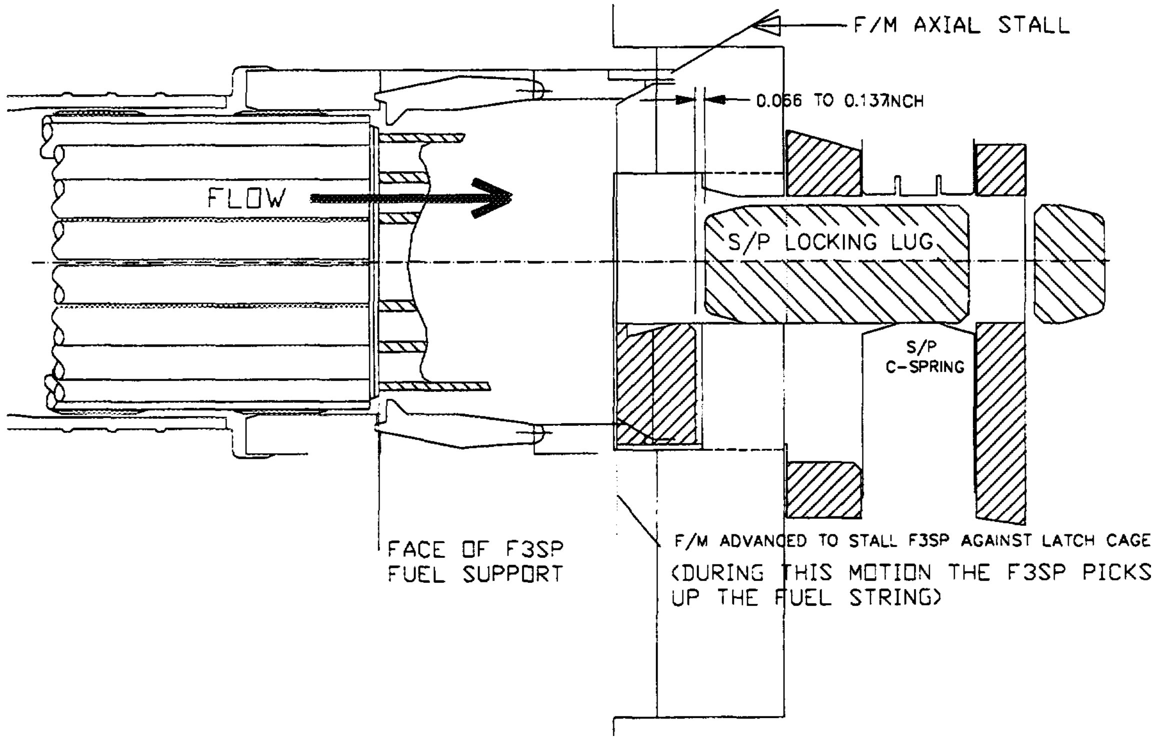


FIGURE 2-4

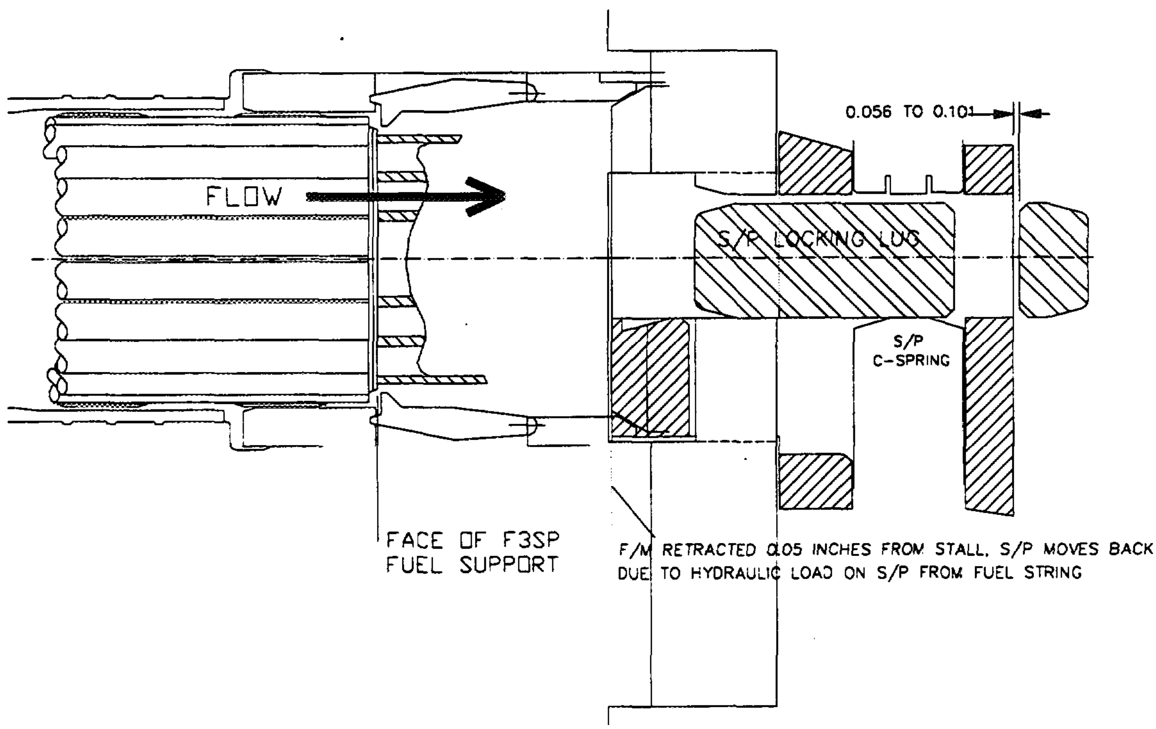


FIGURE 2-5

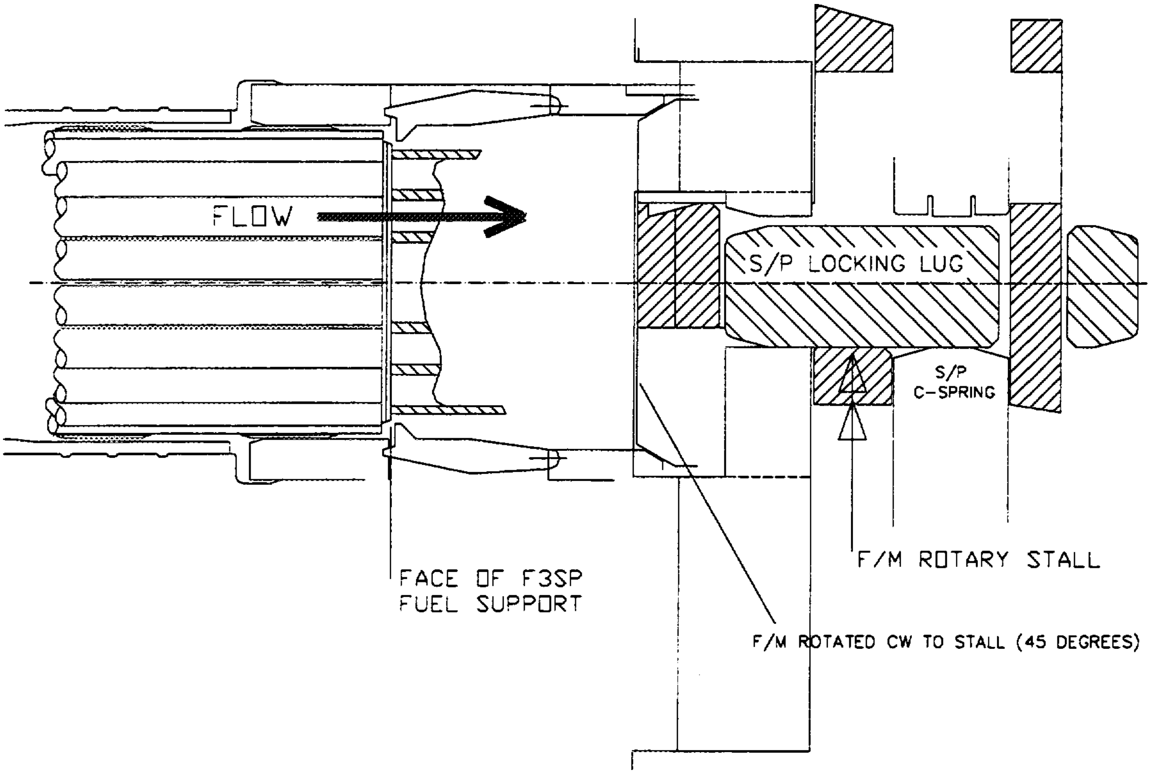
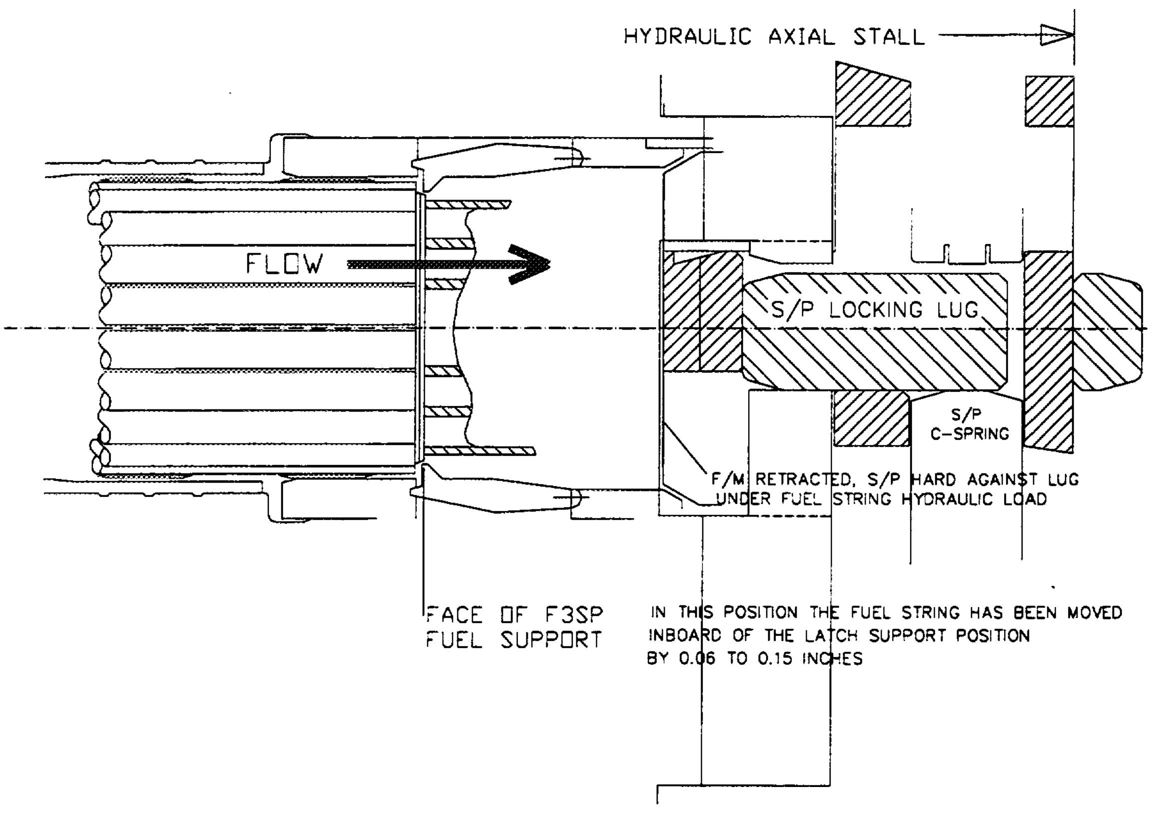


FIGURE 2-6



FUELLING WITH FLOW AT BRUCE A

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ABSTRACT

Fuelling with flow is the solution chosen by Bruce A to overcome the potential power pulse caused by a major inlet header failure. Fuelling with flow solves the problem by rearranging the core to place new fuel at the channel inlet and irradiated fuel at the channel outlet. The change has a significant impact on the Bruce A fuel handling system which was designed primarily to do on power fuelling in the against flow direction.

Mechanical changes to the fuelling machine include a modification to the existing ram head and the replacement of standard fuel carriers with new fuelling with flow fuel carriers having the capability of opening the channel latch.

Changes to the control system are more involved. A new set of operational sequences are required for both the upstream and downstream fuelling machines to achieve the fuel change. Steps based on sensitive ram push are added to reduce the risk of failing to close the latch at the correct position to properly support the fuel string. Changes are also required to the protective interlocks to allow fuelling with flow and reduce risk.

A new fuel string supporting shield plug was designed and tested to reduce the risk of endplate cracking that could occur on the irradiated bundle that would have been supported directly by the channel latch. Some operational changes have been incorporated to accommodate this new shield plug. Considerable testing has been carried out on all aspects of fuel handling where fuelling with flow differs from the reference fuelling against flow.

INTRODUCTION

Early in 1993 the potential for a power pulse accident on a Bruce or Darlington reactor was identified. Following a postulated inlet header failure the fuel in half the reactor core channels would shift toward the inlet end. Since the freshest fuel exists at the outlet end and approximately half of the most downstream bundle has never been irradiated, this fuel shift immediately causes a power spike. When this occurs on 240 channels simultaneously, an uncontrolled power pulse results.

Many potential solutions to the problem were put forth and following methodical evaluation, two solutions appeared most feasible: long fuel bundles and fuelling with flow. The long bundle concept solves the problem by minimizing the gap which exists between the most upstream bundle and the inlet shield plug, thereby limiting the potential fuel string shift. This is achieved by introducing fuel bundles that are longer than standard and continually monitoring the remaining gap. This may seem straightforward but one must appreciate that due to channel creep the combination of normal and long bundles becomes unique for every channel. The number and position of long bundles must always be tracked, the minimum gap size in any channel cannot be violated, the fuelling machines must be precisely loaded with the correct combination in the correct order in the correct magazine positions and there can be no deviation from the planned fuel changing mission once the machine is loaded. Much effort has gone into development of the long bundle program, the details of which are outside the scope of this paper. It has currently been implemented in limited form at Bruce B and Darlington.

Fuelling with flow solves the power pulse by reversing the core to place the freshest fuel at the inlet end and the irradiated fuel at the outlet end. The fuel string shift following an inlet header break then results in a decrease in reactivity as the most irradiated fuel enters the core and the freshest fuel leaves. This is the solution chosen by Bruce A. This paper discusses the changes required to the fuel handling system along with testing completed to support the fuelling with flow program.

BACKGROUND

Although the Bruce A reactors and fuel handling system were designed to be fuelled against the flow, there was always a requirement to be able to hold a latch open to allow fuel to pass back through. Initially this was intended to be done by manual operations with special tooling under shutdown conditions, but later evolved into the flow defuelling technique developed as part of the breakdown channel defuelling system. This technique has been used successfully at all stations with latch endfittings. Experience with flow defuelling contributed significantly to the fuelling with flow program at Bruce A.

Fuelling with flow does not result in any major changes to the fuel handling system. The new fuel is now inserted at the channel upstream end and the irradiated fuel discharged at the downstream end. New control system OPs and Sequences have been written to achieve this. Also a new fuelling with flow fuel carrier has been provided in the fuelling machines, and a new fuel string supporting shield plug has been installed at the channel outlet end. The new fuel mechanism ram stroke has been extended to accommodate the new fuel carriers. Changes to some protective interlocks have been provided and some changes have been made to control hardware. Testing of some critical fuelling with flow parameters was performed in the GE Canada lab and Chalk River Nuclear Labs. Full testing of fuelling machine sequences are being carried out at Bruce A.

MECHANICAL HARDWARE

Fuelling with flow requires inserting new fuel at the channel upstream end while discharging irradiated fuel at the downstream end. Although the original fuelling machine fuel carriers are capable of the new operations at the upstream end, they are unable to open the channel latch to allow passage of fuel out of the channel at the downstream end. New fuelling with flow fuel carriers were therefore designed and supplied. The new design is shown in Figure 1. For economic reasons the design reuses the original fuel carrier body, truncated, with the new nosepiece riveted on.

The fuelling with flow fuel carrier has a protruding, thin walled nose which can open the channel latch and hold it open during the fuel change. The new carrier also has a flat shoulder machined on it, specifically to allow stalling against the shield plug lug in the end fitting. An operation stalling the carrier against the lug is carried out during the fuel accept sequence to verify the charge tube axial encoder prior to opening the latch. The new carrier incorporates inserts to operate the existing breakdown channel defuelling ram catch adapter. It is now unnecessary to replace the fuelling with flow fuel carriers with defuelling carriers when grappling is required.

Testing indicated a potential for bundle endplate cracking of irradiated fuel held by the channel latch. A new fuel string supporting shield plug (F3SP) has been designed and supplied, and is shown in Figures 2 and 3. This shield plug features a nosepiece which protrudes just through the channel latch and supports the fuel string off the latch fingers. Relative rotation can take place between the shield plug body and the nosepiece. In service the nosepiece is prevented from turning by the end fitting shield plug lug while the shield plug body rotates to lock or unlock from the end fitting. This ensures no relative rotation between the nosepiece and the fuel which otherwise might result in fuel damage.

The stroke of the new fuel mechanism's fuel transfer ram is too short to reliably push bundles past the lip of a defuelling carrier. Because of this shortfall, equipment was designed to retrofit the four mechanisms with different ram position detection components. The same retrofit allows the ram to provide sufficient additional stroke.

The chamfer has been increased on the four cruciform sections on the back end of the fuelling machine ram head. This improves the ability of the ram head to open the channel latch in the retract direction. Channel creep results in the need for the downstream ram to penetrate through the latch and any change in the orientation of the ram head during retract will result in contact between the ram head and the latch fingers. The ability to open the latch smoothly prevents the ram from hanging up or possibly causing damage.

OPS and SEQUENCES

New fuelling machine control operations and sequences had to be written to achieve fuelling with flow. The fuel change occurs while the channel latch is being held open by the fuel carrier at the downstream end. Any operations while the latch is open are critical since an incident could result in the fuel string passing uncontrolled through the latch. It is especially important that the fuel be held in the correct position while the carrier is withdrawn to close the latch to ensure that the latch fingers drop between bundles correctly to properly support the fuel string. Consequently the fuelling with flow ops and sequences contain more steps specifically added to verify correct position or operation.

New sequences were also written for installation and removal of the new fuel string supporting shield plug at the outlet endfitting. New Ops in the sequences check the type of channel end being visited to ensure that the appropriate sequence is being used.

PROTECTIVE INTERLOCKS and CONTROL

In the fuelling with flow mode of operation, the channel latch is opened to accept fuel into the fuelling machine on each fuel change. Holding the latch open with the fuelling with flow fuel carrier increases the hazard, since there is a greater period of time during which system failure, either due to power failure or system error, could lead to fuel being washed back through the open latch into the fuelling machine, leading to a difficult recovery process. For this reason, a number of interlock changes have been made to the system.

While the ram is supporting the fuel string, as the string is retracted to receive the irradiated fuel at the downstream channel end, at least one of the ram clutches must be engaged so that the ram is not pushed back freely by the fuel string. Hardwired interlocks prevent releasing the ram brakes at the downstream end unless one of the clutches has been engaged.

Similarly, while the charge tube has advanced to a position where the fuel carrier is holding the latch open, if the axial drive brake is released and the clutch is not engaged, the charge tube would be pushed backward by the fuel string, leaving fuel improperly positioned relative to the channel latch. Hardwired interlocks prevent releasing the charge tube axial brake unless the clutch is engaged.

A number of protective system interlocks have been changed to ensure that the fuel string is properly supported while the latch is open, and to ensure that the fuel will be left properly positioned after the latch is closed. Another set of changes prevents advancing the fuel string against a braked ram during the various sensitive ram push operations, and otherwise pinching the fuel string between two rams at high torque. Torque limitations on the fuel supporting shield plug required a limit on charge tube rotary torque while installing or removing the shield plug.

Interlock changes are incorporated to both ram and charge tube axial drives to ensure proper positioning of fuel relative to the latch. For the ram drive interlocks, one change specific to ram motion applies. When retracting the fuel string into the downstream fuel carrier while it is holding the latch open, the ram can be retracted only to the position where the bearing pad at the upstream end of the second bundle is over the channel latch line, so that as the latch is closed it cannot catch the bearing pad. At this point, the third bundle is still upstream of the latch, and there is no danger of it coming through the latch as the fuel carrier is withdrawn. The charge tube axial interlocks are mainly related to ensuring that the fuel is properly positioned before the fuel carrier can be retracted to close the latch. Compensated axial motion is used when closing the latch such that the charge tube and fuel carrier are retracted while the ram is driven forward relative to the charge tube to hold the fuel string stationary with respect to the latch. This forward motion of the ram inside a fuel carrier in a lip down orientation was previously prevented due to the possibility of damaging a bundle against the lip, but is now allowed over specific defined ranges.

To ensure that the forces on the fuel string are kept low enough to prevent fuel damage, a number of new protective interlocks have been applied. Signals are generated from the upstream fuelling machine to indicate that both of the ram brakes and clutches have been released, and therefore the ram is 'floating'. This condition is required during ram position sensing operations which are done more often during fuelling with flow. While discharging fuel from the upstream fuel carrier, the ram must be advanced behind the fuel to move it along within the carrier until it reaches the area of channel flow and is carried by the flow into the channel. Since there is the possibility of advancing this ram to pinch the fuel string between the two ram heads or against the latch at the downstream end, the speed and torque for this ram advance motion are limited to lower values than when advancing the ram within a fuel carrier at the downstream end.

Controller program functions have been added to accommodate the fuelling with flow mode of operation. The new functions ensure that fuelling with flow is being carried out at appropriate channels and each fuelling machine is executing the right sequence for the end at which it is operating. There is capability of monitoring the motion of the rams of the two fuelling machines to ensure that they are properly synchronized during the fuel transfer. As the downstream ram retracts to accept fuel bundles into the carrier, the upstream ram advances to move fuel along so that it is carried into the channel by the flow. The upstream ram follows along behind the fuel to ensure that it continues moving into the channel. Maintaining a gap between the ram and the fuel ensures that the fuel string is not pinched between the two fuelling machine rams, while minimizing the gap reduces the impact as each bundle is swept by the flow into the slowly moving fuel string. This synchronization is achieved by having ram encoder feedback from both fuelling machines monitored by each controller simultaneously. The ram encoder signal is connected directly as an input to the controller of the other fuelling machine on the same trolley, thereby avoiding delays inherent in communicating the data over the interprocessor communication links.

During the changeover from fuelling against flow to fuelling with flow, it is important to keep track of the fuel ordering in each channel since both types of fuelling will be in service until the changeover is complete. A channel information data file, CHANN.TD, has been created to store the characteristics of each channel of each reactor so that the control system can determine whether the proper fuelling approach has been selected for the channel. Other pertinent data stored for each channel include: channel data updated, channel quarantined, shield plug not installed, channel defuelled, long bundle(s) in channel, flow restricting outlet shield plug installed, and F3SP installed.

The operation of all of the fuelling machine brakes has been changed to a 'command to release' from the previous 'command to engage'. In this way, on loss of computer control, the brakes revert to the braked state, holding the machine components in their current positions.

For additional reliability, Class II power has been provided to both of the brake circuits for the fuelling machine ram, charge tube axial and rotary drives. The Class II power is switched within the fuel handling power distribution cabinet so that as long as the normal Class III supply is available, it is used and the Class II is standby. If the Class III is lost, Class II is immediately switched in with only a momentary delay in the supply to the brakes. This change will prevent the fuel string passing through the latch uncontrolled in the event of a loss of power during the fuel change while the latch is being held open.

TESTING

A series of tests were performed in the GE Canada lab to demonstrate the feasibility of fuelling with flow at Bruce A and establish the key flow rate parameters. These tests are summarized as follows:

1. Check the release of fuel bundles past the lip of the upstream fuel carrier at normal full flow conditions.
2. Establish the maximum flow at which fuel bundles will successfully move past the lip of the upstream carrier.
3. Determine the effect of a bundle impacting the fuel string over a distance of 45 inches under full flow conditions.
4. Establish how far the fuel carrier can retract from full forward stall in the latch before a bundle no longer passes through.
5. At full flow conditions establish the point of relative movement during rotation of the upstream carrier when the new fuel is in contact with the fuel string.
6. Establish the minimum flow requirements to consistently wash two bundles past the lip of the downstream fuel carrier.

The results indicated no reportable damage to any of the fuel bundles, fuel carriers or channel components during any of the tests. The minimum flow rate to successfully receive two bundles over the lip of the downstream carrier was established to be 20.9 Kg/s at 300°C. The series of tests showed that pursuing fuelling with flow at Bruce A would be a viable option.

Testing at CRNL indicated a possible problem with cracking of endplates of irradiated bundles supported on the latch. This concern led to the design and implementation of the fuel string supporting shield plugs in the outlet endfittings.

Testing of the new Ops and Sequences, and the new fuel handling protective programs are being performed at Bruce A by Ontario Hydro with support from GE Canada Fuel Handling personnel.

CONCLUSIONS

With appropriate changes to the fuel handling equipment and control system, the reactors at Bruce A can be refuelled in the fuelling with flow direction. The current test program together with previous flow defuelling experience established confidence in this fuelling method. The replacement of outlet shield plugs with the new fuel string supporting shield plugs will eliminate the risk of bundle endplate cracking at the latch.

ACKNOWLEDGEMENTS

The author would like to acknowledge the contributions of Ontario Hydro, AECL and the GE Canada fuel handling team to the overall program. The work was carried out as part of the Power Pulse program funded and managed by Ontario Hydro.

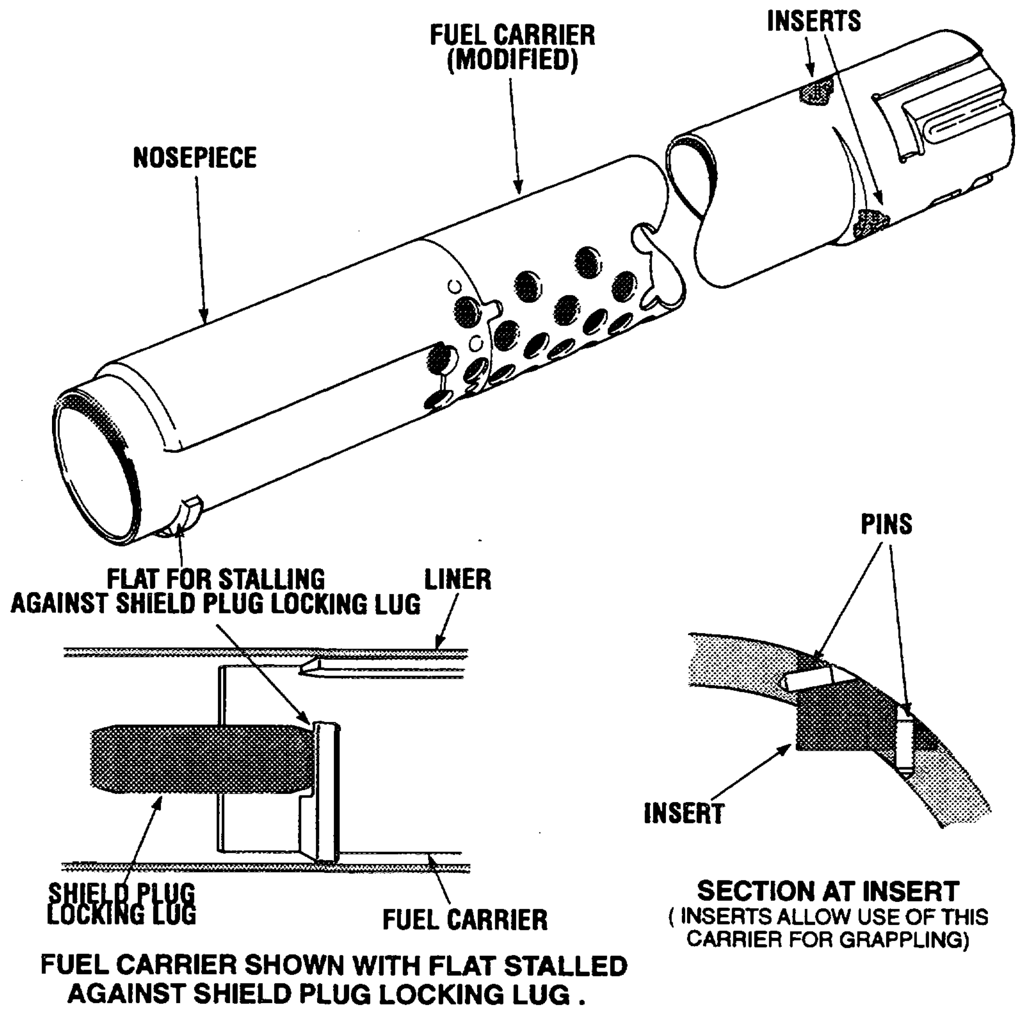
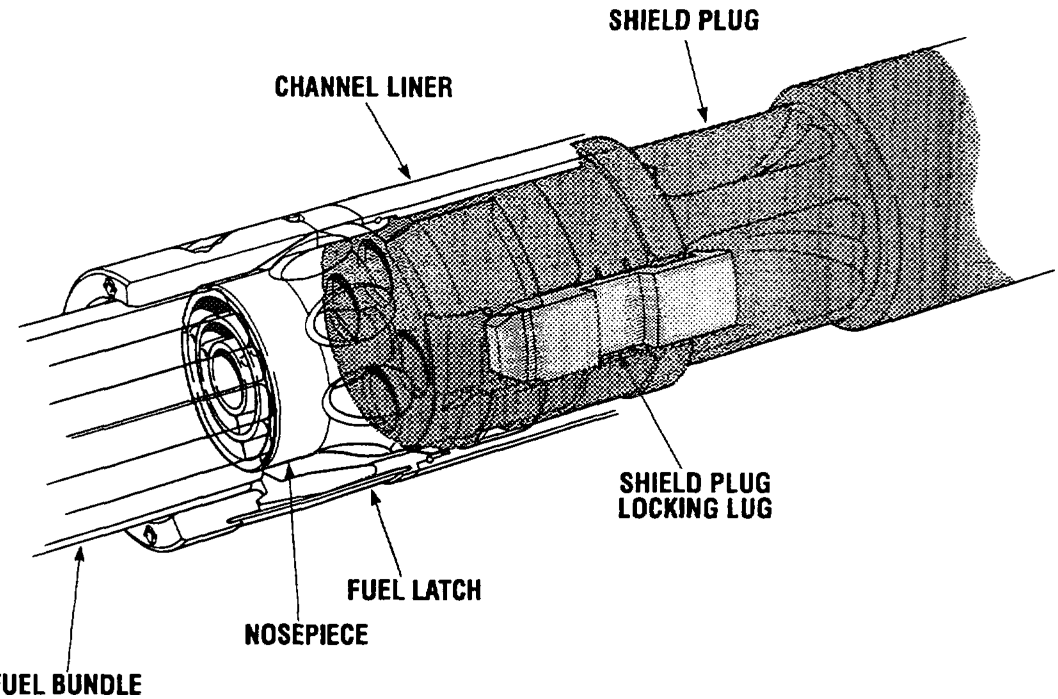


Figure1 - Fuelling With Flow Fuel Carrier



FUEL STRING SUPPORTING SHIELD PLUG ADVANCED INTO LINER

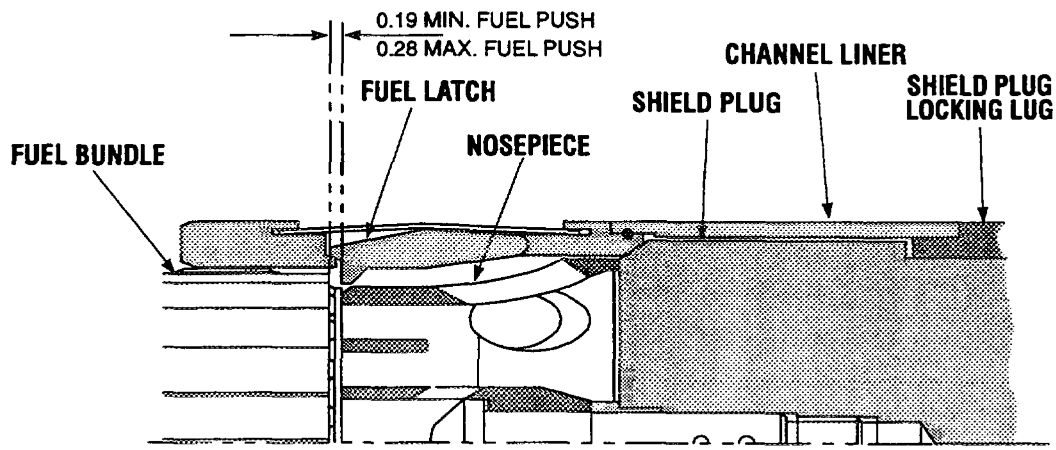


Figure 2 - Fuel String Supporting Shield Plug - Unlocked

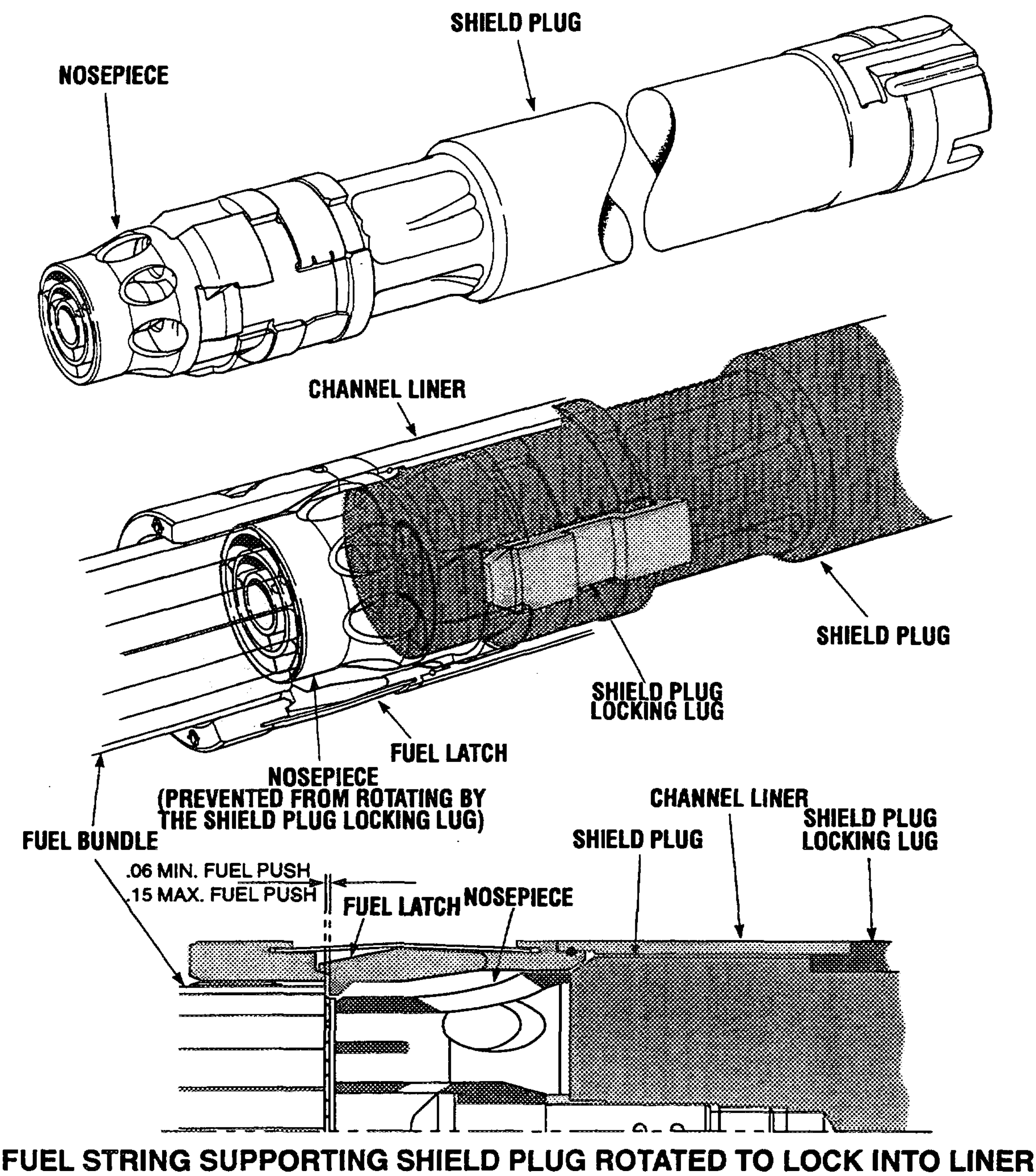


Figure 3 - Fuel String Supporting Shield Plug - Locked

BUNDLE 13 POSITION VERIFICATION TOOL DESCRIPTION AND ON-REACTOR USE

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ABSTRACT

To address the Power Pulse problem, Bruce B uses Gap: a comprehensive monitoring program by the station to maintain the gap between the fuel string and the upstream shield plug. The gap must be maintained within a band. The gap must not be so large as to allow excessive reactivity increases or cause high impact forces during reverse flow events. It should also not be so small as to cause crushed fuel during rapid, differential reactor/fuel string cool downs. Rapid cool downs are infrequent.

The Bundle 13 Position Verification Tool (BPV tool) role is to independently measure the position of the upstream bundle of the fuel string. The measurements are made on-reactor, on-power and will allow verification of the Gap Management system's calculated fuel string position.

This paper reviews the reasons for developing the BPV tool. Design issues relevant to safe operation in the fuelling machine, fuel channel and fuel handling equipment are also reviewed. Tests ensuring no adverse effects on channel pressure losses are described and actual on-reactor, on-power results are discussed.

1. INTRODUCTION

Gap Management is a comprehensive monitoring program to maintain the gap between the fuel string and the upstream shield plug (see Figure 1). The program defines a safe range of gap between the upstream shield plug and the thirteenth bundle. The gap is to be small enough to lessen the reverse flow impacts and limit increased reactivity. As well, the gap is to be large enough to ensure that the fuel string is not compressed due to differential cooling during rapid reactor cool down conditions.

When channel creep lengthens the fuel channel, the gap between the thirteenth bundle and the upstream shield plug increases. As the channel creeps, increasing numbers of special long bundles are used in the fuel string to maintain the proper gap.

The BPV tool assists in the process by obtaining independent measurements of the gap. These measurements are used to determine correction factors for the gap calculation.

2. GAP MANAGEMENT REQUIREMENTS FOR BPV TOOL

The Gap Management program contains algorithms and data bases to continually track the position of bundles. The recorded bundle lengths in the fuel strings are combined with knowledge of fuel channel lengths to calculate the existing gaps. The program requires verification of this gap by means independent of the program inputs (bundle records, fuelling machine end fitting position measurements, fuelling machine fuel string position measurements). The gap calculations must also account for varying reactor conditions. Again, the need for an independent tool capable of operating in reactor conditions led to the requirement for the BPV tool.

The gap is maintained through knowledge of the gap and the use of two sizes of fuel bundle (19 inch standard and 19.5 inch 'long', 483 and 495 mm respectively). The fuel string is composed of all standard length bundles when the fuel channel is in its early years of use. As the radiation induced creep lengthens the pressure tube, the two end fittings grow apart. This causes the gap to increase. When the gap is determined to be approaching the upper tolerance, one of the new bundles installed will be a long bundle. The gap is thereby reduced by half an inch. The long bundle is advanced upstream with each subsequent fuelling operation until the it eventually reaches the discharge position. At this time the new fuel charge must have the same combinations of bundle types to maintain the fuel string length and the gap.

The gap management algorithm, utilized by the Bruce B fuel handling control software, incorporates a basic equation to define the position of upstream bundle 13. During each channel fuelling operation, upstream and downstream fuelling machines have their rams in contact with the fuel string. The ram encoders supply two of the variables needed to measure the gap. The allowed gap, the accuracy of the fuel handling system, and the fuel channel current length are constants that have been determined. Other variables are dependent on what combinations of fuel bundle types are required to be inserted and discharged. These fuelling operation variables are specified when the operator determines the bundle combinations used.

The basic equation is affected by the power level, the specific station, reactor temperature, and the use of flow injection. These effects are grouped into the single correction factor K_p. This factor can only be determined by actual measurements in representative conditions. The BPV tool was developed to assist in this role. It is designed to be installed in any channel of an operating Bruce B reactor. Measurements retrieved from the tool can be used to assign values of K_p for each combination of power level, channel position, etc.

An additional role of the BPV tool is to assist in recovery from some accident cases due to bundle combination errors. If the gap management system's gap algorithm detects an error, the BPV tool can be used to measure the actual bundle position independently.

3. BUNDLE 13 POSITION VERIFICATION TOOL DESCRIPTION

The BPV tool obtains measurements completely independent of the channel and fuelling machine components because it incorporates a mechanical recording device. This device is basically a sliding depth gauge and takes its reading directly from the fuel bundle relative to the end fitting.

The BPV tool (see Figure 2) is composed of three major components:

- tool body
- fuel sensor
- spring plunger

All three parts are made of the same type stainless steel.

3.1 Tool Body

The tool body (see Figure 2) is a modified fuelling machine fuel carrier made of stainless steel with chrome plating of its sliding surfaces. The outboard end (relative to the reactor centre) has slots and recesses to facilitate connection to the fuelling machine charge tube. Along the length are holes for the primary heat transport coolant flow. A relief of the outer diameter for most of the length on one side provides clearance when the body is warped by the stratified temperatures encountered during service. A slot with gauge markings, in conjunction with a pin mounted in the fuel sensor, indicates the measured position of the fuel bundle. A bulkhead is pinned within the bore of the body. The bulkhead restrains the spring of the spring plunger. A tubular extension of the bulkhead restricts the fully extended reach of the fuel sensor to no more than 1.5 inches (38 mm) past the inboard end of the body. The inboard end enters the chamfer at the end fitting spacer. The spacer line is a fixed plane through the end fitting at the end of the channel liner tube. The shield plug, fuel carrier and BPV tool all abut the chamfer at this line (see Figures 1, 2 & 4).

3.2 Fuel Sensor

This part is contained with the spring plunger and body (see Figures 2 and 3). It functions as a depth gauge probe. A chromed, segmented flange provides sliding contact with the spring plunger bore. The flange is designed to yield if the sliding surfaces jam. The axial position of the sensor within the plunger is held by friction pads. The springs within the pads are shimmed to provide sufficient friction to resist the drag of the coolant flow yet slide with an acceptable force when the fuel bundle is encountered.

A set of concentric rings at the inboard end distributes any forces against the fuel bundle. The outermost ring has three pads to support the inboard end and allow the sensor to slide along the bores of the pressure tube, liner spacer and tool body. Radial holes allow coolant flow to reach the centre of the fuel bundle.

3.3 Spring Plunger

The spring plunger is tubular. The bore carries the fuel sensor. The outer diameter has chromed bearing rings to slide within the body. The axial limits of the plunger stroke are set by the bulkhead and a step in the body's bore near its outboard end. Normally the parts are kept in the retracted outboard position by a large spring between the plunger and bulkhead. During measurement operations the fuelling machine ram compresses the spring and advances the plunger and fuel sensor.

4. BPV TOOL OPERATION

The BPV tool is brought out of storage and placed in a trough attached to the ancillary port (a containment penetration accessible to the fuelling machine within containment and by personnel in the irradiated fuel bay room). A hand tool is supplied by which the indicator pin can be pre-set. When the BPV tool is ready, the fuelling machine engages the tool and stores it in its magazine.

After locking onto the target end fitting, the fuelling machine engages with the BPV tool and advances it into the end fitting to stall, positioning the tool nose at the spacer line. Fuel coolant flow will be constantly flowing through the tool.

The plunger, with its sensor having been pre-set in its fully extended position, is advanced to the measuring location by the fuelling machine ram (see Figure 4). When the sensor encounters the thirteenth bundle before the plunger has been fully advanced, relative sliding between the stalled sensor and the advancing plunger causes the indicator pin to slide back along the groove. Retracting the ram allows the gauge to be fully stored within the BPV tool's body. When the plunger has returned to its home position, the pin lines up with the scale inscribed on the body and allows the detected position of the bundle to be read.

The scale is set up to read positive numbers if the thirteenth bundle is outboard of the spacer line. Thus a reading of positive 2.5 inches would indicate that the bundle might encounter the flow straightener if the shield plug were re-installed. A zero reading indicates that the bundle end plate is in the plane of the spacer line. If the bundle is farther inboard than the spacer line, negative measures are recorded.

It should be noted that the second fuelling machine will lock onto the other end of the fuel channel for the duration of the BPV tool visit. With two fuelling machines in use, the channel pressure drop is monitored for any excursions beyond what is normal during fuelling operations.

The BPV tool is then retracted into the fuelling machine and returned to the ancillary port. The fuelling machine locks on the port, the port is opened manually and the trough installed. The tool is then be slid onto the trough. The reading on the tool's scale is recorded as the thirteenth bundle position.

The tool could either be removed and stored or reset for further measurements.

5. DESIGN ASPECTS FOR RELIABLE IN-REACTOR USE

5.1 Accommodating Flow Injection

The Bruce B fuelling machines incorporate a flow injection system. While the fuelling machines are locked onto a fuel channel, cooling water is circulated through the machine's magazine. To guard against the ingress of hot water from the channel and any debris or crud that may come with it, sufficient cooling water is injected to create a slow flow out of the fuelling machine. When considering tools or components that leave the relatively cool fuelling machine interior (between 27 and 93 Celsius, or 80 to 200 Fahrenheit) and enter the hot fuel coolant flow (up to 305 Celsius, or 580 Fahrenheit), thermal effects must be considered.

The tool was designed with similar materials throughout such that thermal expansion has virtually no effect on the measurement. However, a small error is unavoidable based on the differential movement of the expanded parts. The expansion/contraction applies only to the range of sensor shift (-1.5 to +4 inches = 6 inches). The net effect on measurement is well below the desired +/- 1/16 inch accuracy required.

A less obvious thermal effect is due to the fact the flow injection water does not mix for some length down the end fitting. The stratification causes a considerable temperature difference along the top and bottom of any component installed. The differential thermal expansion causes components to bow, sufficiently in some cases to cause interference within the end fitting liner bore.

The BPV tool accommodates the bowing effect by being designed with large clearances and ensuring long components were not over constrained.

Commissioning tests included differential heating of the tool's body to duplicate the worst case bowing. Stroking the tool's components in this condition ensured that no jamming would occur in the reactor that could cause excessive forces against the fuel bundles.

5.2 Coefficient of Friction

The sensor relies on its friction pads to limit the force against the fuel bundles during initial installation and the measurement operation. Extensive commissioning tests were performed to prove that the design force was not exceeded regardless of whether the tool was heating up, cooling down, wet or dry.

Tests uncovered one unique condition in which the friction pads locked. The sequence of events was:

- the tool was exposed to simulated coolant water,
- then allowed to dry, and
- bowed by applying heat.

Previous tests had shown the coefficient of friction between certain components to be within the expected range for aluminum bronze against stainless steel. Within this range, the geometry is such that window jamming should not occur.

An interesting trend developed after investigating the possible combination of applied forces and spring forces with various coefficients of friction. As the friction coefficient increased the static model could be shown to jam in one state. However, at higher coefficients the jammed case would develop a different set of equilibrium forces and theoretically not jam or be statically indeterminate. In that condition, another equilibrium state would exist, jamming and locking at other coefficients of friction.

Tests on the tool were able to duplicate the coefficient of friction and show that it corresponded to the lowest value responsible for the geometry to lock. This low value was twice as high as normal for the combination of materials. The high coefficient of friction was thought to be due to the extremely clean condition of the tool after being exposed to the hot, basic pH, coolant. It was also thought that the lithium hydroxide used to adjust the water's pH may have formed a thin deposit of crystals. Stroking the components once or twice always reduced the coefficient of friction to within the expected range

It is noteworthy that the jamming case was not classical window jamming, but in fact, the friction pad pistons stalling all on one side of their bore hole. The unique equilibrium cases were caused by the relative movement of internal components as the tool body bowed. By changing the friction pad tip profile and by moving them to a plane not sensitive to body bow, the jamming is avoided, even at the high coefficients of friction.

5.3 Tool Failure Safety

As the tool is to be pushed against the fuel string through commands of the fuel handling control software, several tool failure modes were investigated.

The first to be investigated was the failure of an extended sensor to retract (due to spring failure or particles jamming in the sliding surfaces of the sensor). In such a case, the tool would not fit within the fuelling machine magazine. The fuelling machine magazine would be commanded to rotate regardless. The sensor is strong enough to stall the magazine drive without itself being deformed. This would allow for a contingency manoeuvre in which the fuel string would be advanced by the opposite fuelling machine to push the sensor back into the tool body.

To guard against particles preventing retraction of the sensor, two design features were incorporated. On the outboard end of the sensor, bearing surfaces were designed with the ability to fold away from jammed particles. At the other end, generous clearances and small footprint pads allowed jammed particles to be dislodged by jogging the sensor.

The folding bearing pads also guard against excessive forces being applied to the fuel string when the fuelling machine ram extended the plunger.

5.4 Fuel Channel Pressure Drop

The BPV is an intrusive tool in that it must place a stiff sensor within the coolant flow up against the upstream bundle. If the flow were to be blocked, fuel bundle temperatures would rise. Normal fuel handling control software monitors the pressure differential. If a flow blockage is deduced, the operators are warned and the BPV tool would need to be removed.

The sensor head was designed to avoid blocking flow to the bundle. Ample flow areas are machined through the head and the central tube is cross drilled to deliver water flow to the bundle centre. Up stream of the bundle, coolant flow enters the tool body radially through holes that are in the same area of the radial holes in the liner. All plunger, spring and bulkhead components are placed behind the area of the entering flow.

Tests were performed in the GE Canada fuel channel rig. The sensor was placed in various positions and pressure taps were used to obtain pressure differentials along the BPV tool. The pressure drops recorded were well below the design limits. Drag on the sensor was deduced from some of the pressure differentials. This allowed the establishment of friction pad settings.

6. SAFETY ASPECTS OF BPV TOOL MAINTENANCE

The safety concerns of tool maintenance fall into two categories. First, ensuring the tool operation is reliable and will not damage fuel channel components. Second, ensuring that handling the highly preloaded springs is safe for the maintainers.

Prior to being sent on its mission, the sensor friction is checked. The hand tool supplied to pre-set the indicator pin is used to jack the sensor back and forth. The capacity of the hand tool is lower than the friction loads that would damage the fuel bundles or latch. If the sensor is difficult to move, a test is performed to apply loads to the sensor and document the maximum force required to push the sensor. The test results confirm that the force limiting friction pads are performing within the acceptable limits.

Visual inspection prior to the mission would confirm that the sliding surfaces are intact.

If the plunger is suspected of being jammed, tests and test fixtures are provided to check the plunger movement. Stacked weights added to the test fixture depress the plunger and allow confirmation of the spring preload and stiffness.

The tool's highly preloaded internal spring presents a potential hazard to the maintainers. A set of maintenance tools and maintenance instructions are provided. Using these, the spring can be safely locked during maintenance and overhaul.

7. SITE MEASUREMENTS

The BPV tool has been commissioned and used successfully at site. Full commissioning was performed on the Bruce B fuel channel test facility. There a fuelling machine head installed the tool into a test channel with full reactor operating temperature, pressures and flows. Indicator pin readings were found to duplicate the known position of the thirteenth bundle in the test channel.

Subsequent use on a number of reactor channels has provided fuel bundle positions under operating conditions. These readings have confirmed the estimated K_p correction factor for one set of reactor conditions.

ACKNOWLEDGMENTS

The author would like to thank Ontario Hydro and GE Canada for the use of their data and to acknowledge Patrick Henry of GE Canada as the originator of the BPV tool concept.

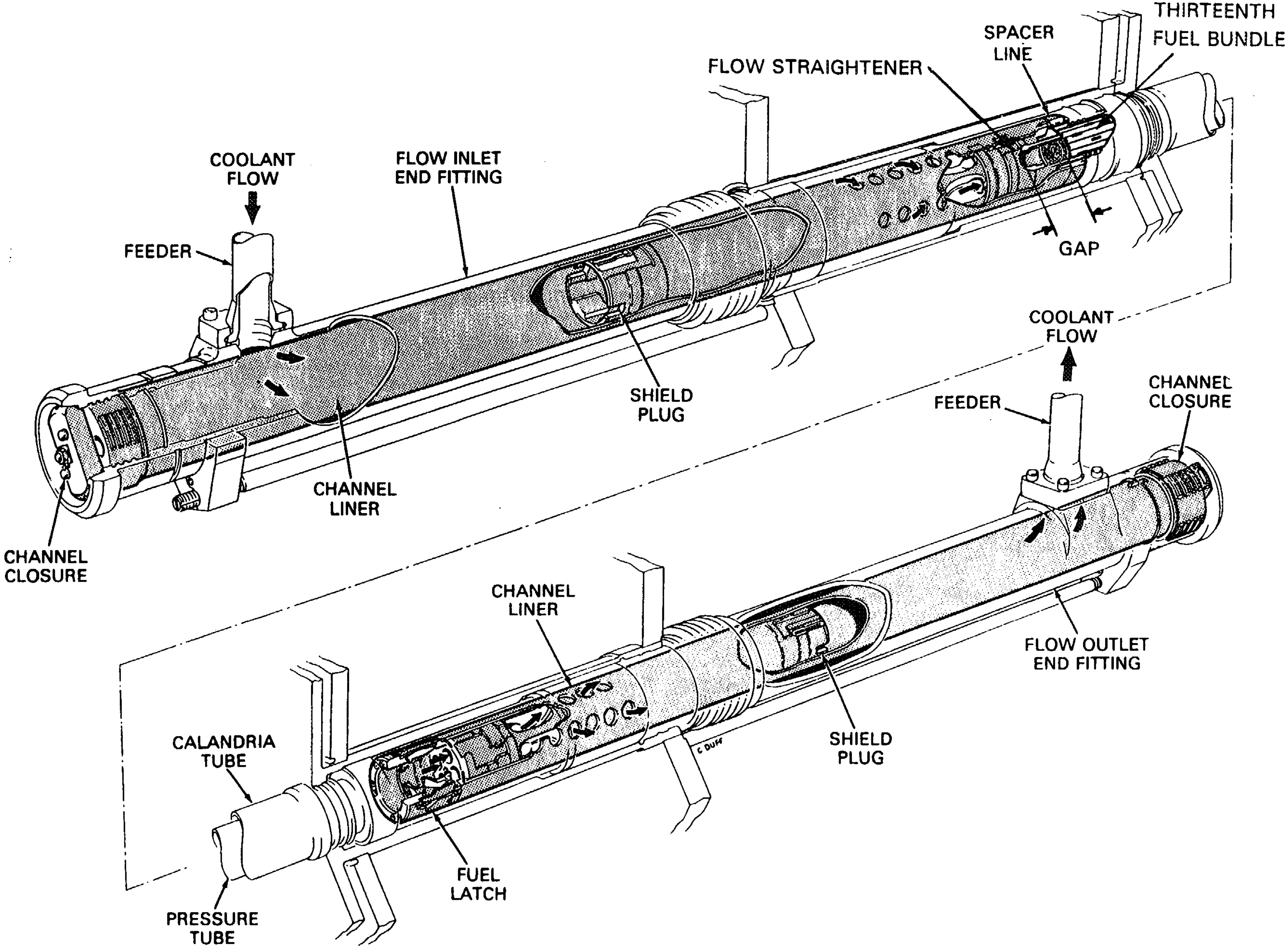

FIGURE 1 -- BRUCE B FUEL CHANNEL COMPONENTS

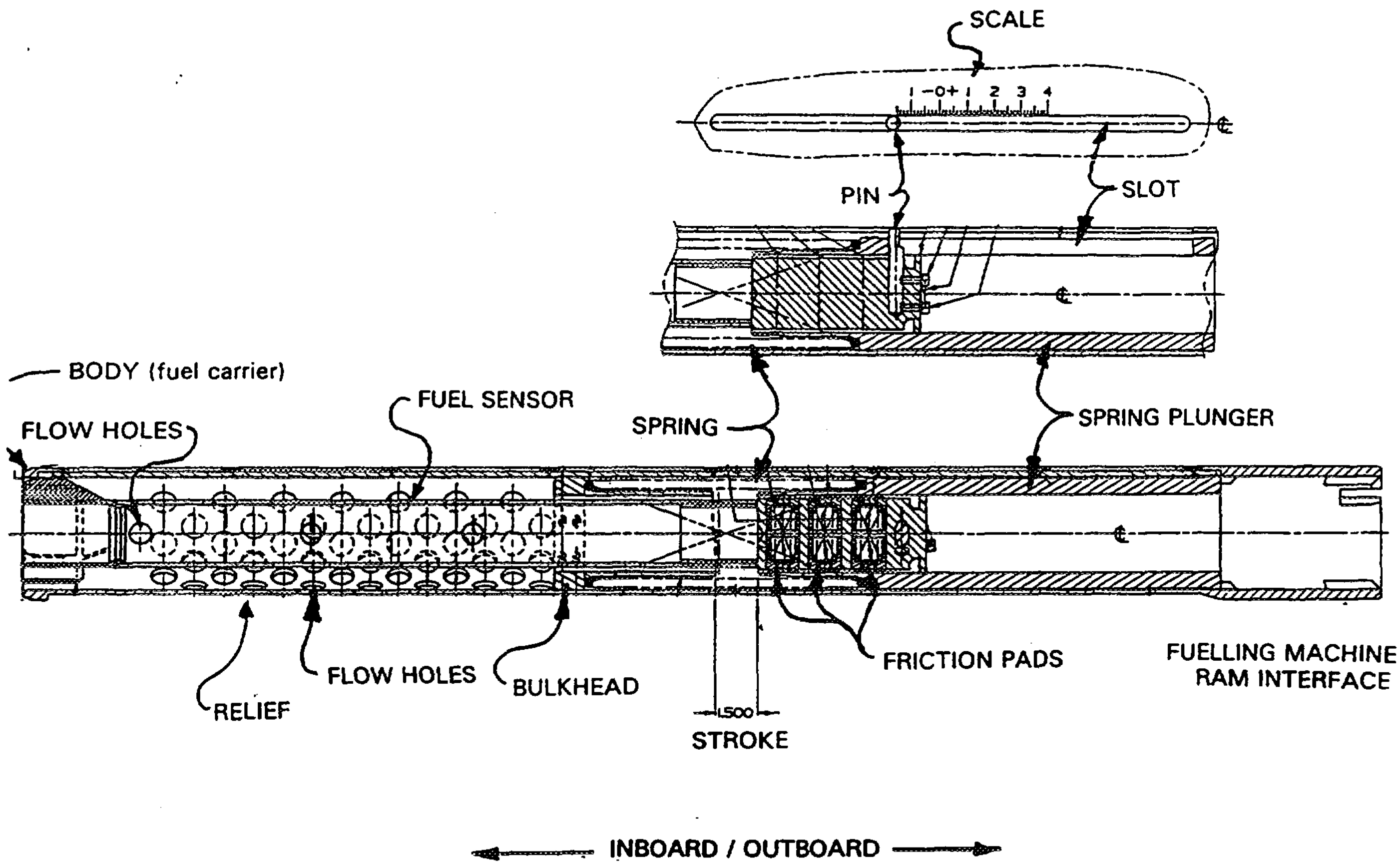

FIGURE 2 -- BUNDLE THIRTEEN POSITION VERIFICATION TOOL

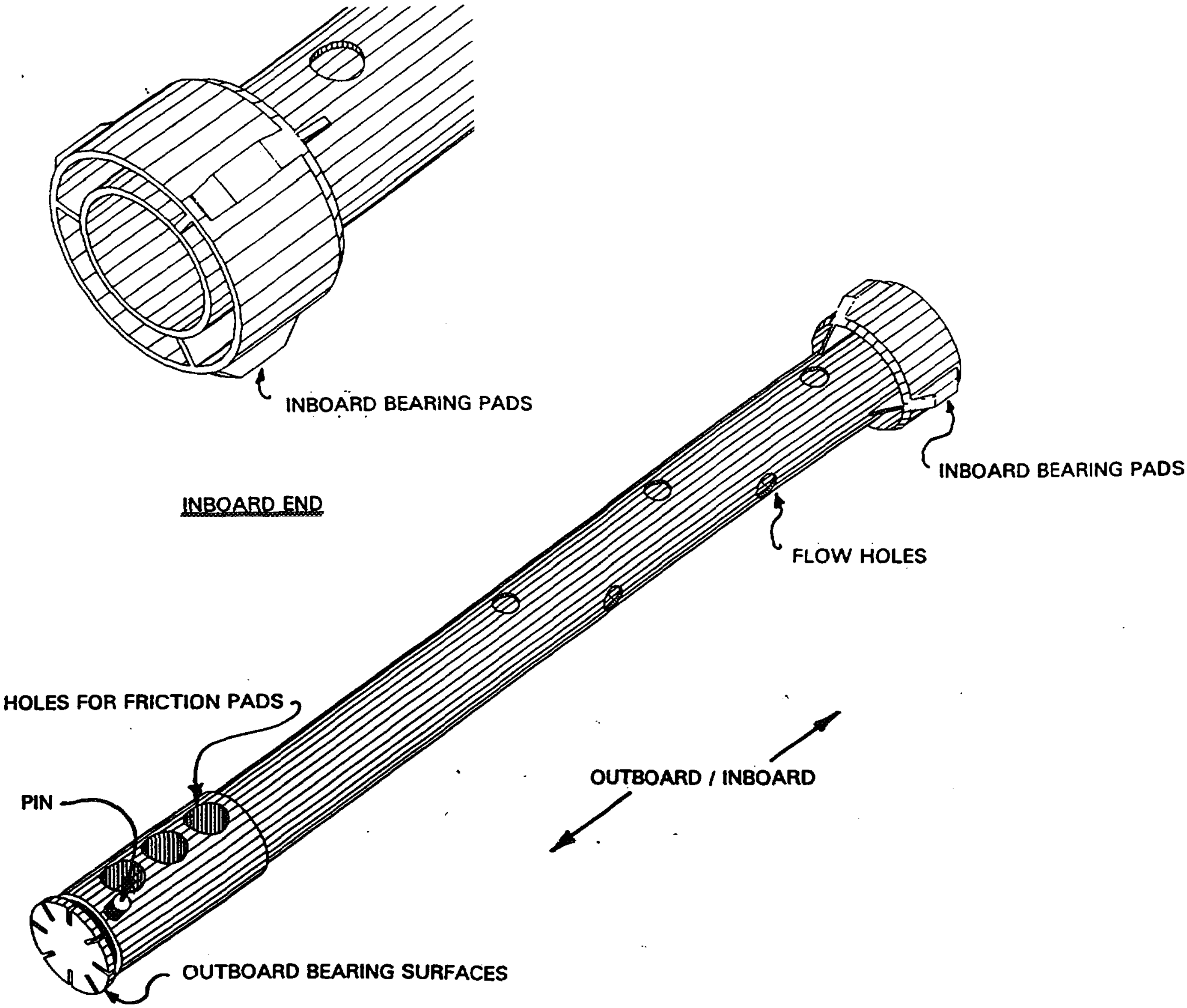

FIGURE 3 -- FUEL SENSOR

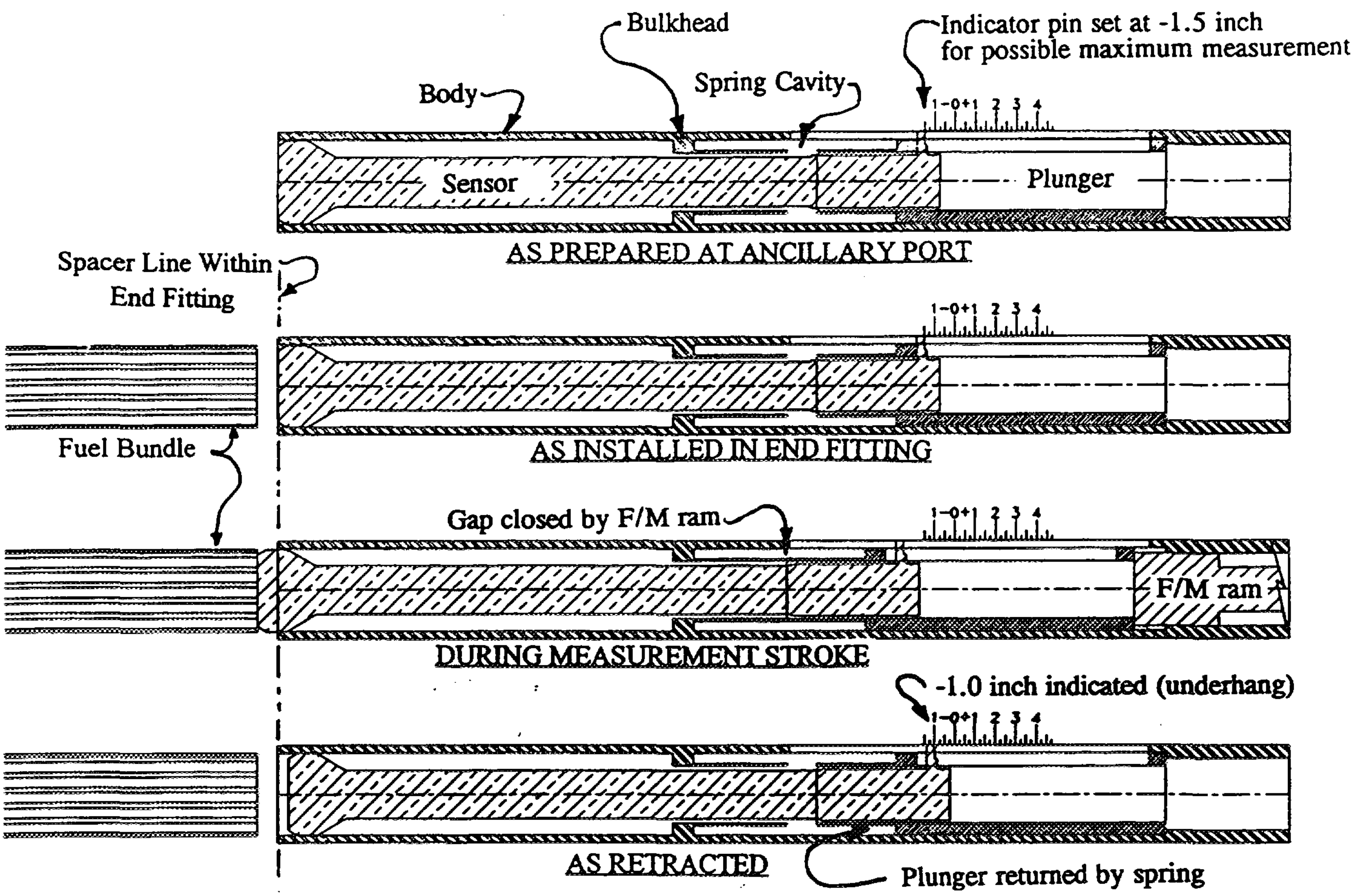

FIGURE 4 – BUNDLE THIRTEEN POSITION VERIFICATION TOOL OPERATION

THE OPERATION AND MAINTENANCE OF THE SLAR SYSTEM AT BRUCE A

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Abstract

The SLAR (Spacer Location And Repositioning) system at Bruce A consists of two (2) Delivery Machines and a Fuelling Machine Trolley equipped with the D_2O and Air auxiliary systems. The Delivery Machines are designed to perform all the Fuelling Machine operations and have the capability to rapidly defuel/refuel a reactor channel and traverse the SLAR tool to locate and reposition the spacers in the channel. The number of functions that a Delivery Machine must perform makes it more complex as compared to the operations of a Fuelling Machine.

The paper discusses the operation of the SLAR Delivery Machines and the problems encountered with the operation and maintenance of this system at Bruce A.

1.0 Introduction

SLAR is a maintenance program developed for the rehabilitation of the fuel channels in the CANDU reactors. It was determined that the P/T (Pressure Tube) to C/T (Calandria Tube) contact can be detrimental to the safe and long term operation of a reactor. The P/T to C/T contact can be caused by the displacement of the fuel channel spacers. The contact area can lead to the formation of Hydride blisters leading to a failure of the P/T over a period of time. Hence, Bruce A implemented the SLAR program to reposition the spacers in Units 3 and 4 reactor channels.

The SLAR program at Bruce A has been successful in completing Three (3) campaigns since 1993 on Units 3 and 4. A total of Four Hundred and Forty Nine (449) channels have been SLAR'd in approximately Three Hundred (300) days of critical and non-critical reactor outage time. The SLAR Channel processing times have varied depending on the size of the spacer and the relative position of the channel in the reactor. The 'Fat' spacers in the Unit 3 channels were found closer to their design position than the 'Thin' spacers in the Unit 4 channels because of their tighter fit. The spacers in the top row reactor channels were displaced more from their design location than the channels closer to the reactor core.

2.0 The Operation of the SLAR System

The SLAR system at Bruce A consists of two (2) D/Ms' (Delivery Machines, Fig. 1) and a Fuelling Trolley (Fig. 2) complete with its D_2O and Air auxiliaries. The D/Ms' are designed to operate within the Fuelling Machine (F/M) envelope. This has made the design of these machines more complex, as they must perform more operations, compared to a F/M.

After the D/Ms' remove the channel plugs, the hydraulic ram within the M/T (Module Tube, Fig. 3) is used to defuel the channel into the F/T (Fuel Tube, Fig. 4) of the D/M on the opposite side. The hydraulic ram also traverses and rotates the SLAR tool during its operation within the channel. The fuel is returned to the channel by a hydraulic piston in the F/T and the channel is closed.

2.1 The SLAR Delivery Machines

The basic design of the SLAR D/Ms' is modified by separating a F/M at the flat head and leaving the front end unchanged (Fig. 1). The remaining assembly consists of an indexing turret in place of a flat head on which three(3) tubes are mounted:

1. A Mechanical Ram Tube, containing the existing charge tube and ram ballscrews and input drives

2. A Fuel Tube

3. A Module Tube containing the SLAR tool and the hydraulic ram.

By rotating the turret, each of the tubes can be aligned with the snout of the head. The F/T and the M/T are longer than the mechanical ram tube and are retracted when the head enters or leaves the reactor vault.

2.1.2 Fuelling Trolley
The South Extension Trolley has been dedicated for the SLAR equipment. The trolley is parked in the ESA (East Service Area) where the testing and maintenance on the SLAR equipment is performed.

The South Extension trolley (Fig. 2) is modified by the installation of two (2) D_2O hydraulic drive units, connected to the existing auxiliaries. These units are used to power the hydraulic devices on the new tubes. The hydraulic power supply (Charging Pump) of each hydraulic drive unit are interconnected to provide backup.

SLAR tool calibration devices are provided in each F/M head (Calibration sleeve) and on the transport trolley (Calibration channel between the two (2) D/Ms').

The trolley catenaries and power track remain unchanged.

2.1.3 Delivery Machines Drives
The SLAR head can handle the normal D/M head operations to automatically manipulate the channel components. The extra requirements of the SLAR head has led to the addition of seven (7) new drives in addition to the seven (7) F/M drives for a total of fourteen (14) drives per side. The new drives are:

1. Turret Rotary
2. Telescopic
3. Umbilical Cable Drum
4. Ram Tether
5. Piston Tether
6. M/T rotary
7. F/T rotary

The controls for the original seven (7) F/Ms' drives have been replaced with controls for fourteen (14) DC servo drives complete with brakes and encoders except for the umbilical cable drum motor which has no brake. Since spare conductors are not available in the catenary or power track for the drives, multiplexing is used where drives share the existing conductors. The maximum number of drives required for any SLAR operation is three (3).

The system is normally operated in a semi-automated mode with the operator initiating the operating sequence stored in the F/M computer.

2.1.4 Slar Tool
The SLAR tool (Fig. 5) used to locate and reposition the Spacers consists of a hydraulic bending tool to raise the P/T, eddy current system for locating the spacers and measuring the gap between the P/T and C/T, ultrasonic probes to detect cracked hydride blisters on the lower section of the P/T, and LIMs' (Linear Induction Motor) to reposition the spring from within the P/T.

The operation of the SLAR tool in the channel is controlled by the SCADA (Supervisory Computer And Data Acquisition) computer which is slaved to the D/M control computer. The LIM power, control and inspection signal from the tool use the umbilical cable connected to the tool. The one hundred and ten (110) ft. of umbilical is wrapped around the cable drum as the tool is extended and retracted in the channel and is connected by slip rings at the Cable Drum.

2.1.5 Operation Control
The SLAR equipment is maintained and tested in ESA prior to going to the unit for the SLAR campaign by the F/H and SLAR personnel. The SLAR campaign at the unit is an involvement of the various work groups, required from preparing the Reactor Vault to the operation of the SLAR equipment.

The operation of the SLAR equipment is performed by F/H SLAR trained operators who are responsible for all D/M, trolley and bridge operations. After the D/Ms' have been homed on the channel, its C/Ps' and S/Ps' removed and the channel defuelled the control of the SLAR tool is transferred to the inspection personnel on the SCADA system to process the channel. The SCADA system however remains slaved to the F/H control computer. When the processing of the channel has been completed, the control is returned back to the SLAR operator to return the fuel to the channel and close it up.

2.1.6 Reactor PHT system
During SLARing, the PHT (Primary Heat Transport) System is put in a low level drain state as the D/Ms' can only operate at a maximum channel pressure of thirty two (32) psig. The limit is set because of the pressure

force exerted on the turret seal.

2.1.7 System Surveillance

To monitor the D/M D_2O supply pressures and record the performance of the various drives, a datalogger is used with each machine. This is in addition to the information available from the F/H computer and Operator logs. The datalogger has fourteen (14) input channels for monitoring the analog signals from the D/Ms', the data from which is graphically displayed on its computer monitor and is stored to its hard drive (Fig. 6 & 7 show the graph of some of hard drive data). The collection capability has proved to be a very useful tool in monitoring and diagnosing the problems with the machines.

2.2 Operation Problems

2.2.1 Temperature Transient on Channel

When SLARing, on three (3) occasions the channel temperature went up from thirty (30) degree Celsius to sixty eight (68) degree Celsius after the fuel was returned to the channel. The channel temperature was monitored for five (5) hrs. until it returned back to normal. Inspection of the fuel from one of the channels showed no visible heat deformation or damage, and all bundles inspected had normal wear patterns on the bearing pads and endplates.

The possible cause of the transient was attributed to the stagnated flow in the channel. With the PHT system in the low level drain state for SLAR, the flow in the channels is reduced. The SLAR tool, when inserted in the channel restricts the flow. When the fuel is returned to the channel the flow stagnates. The heat is eventually removed from the channel via IBIF (Intermittent Buoyancy Induced Flow) and rejected to the header/boilers. This causes a gradual rise in the channel temperature over a short period of time till the flow in the channel gets established.

As a result of these temperature excursion incidents, flow switches are being installed to monitor the cooling flow to the F/T. Also the COT (Channel Outlet Temperature) is monitored for a period of time three (3) hrs. after the fuel has been returned back to the channel.

2.2.2 Crud in the Channel

The D_2O for the operation of the F/T and M/T is taken from the header supplying the Circulating Pumps which in turn is supplied by the D/M heads. This supply is unfiltered and contains the crud that gets transferred to the head from the E/F (End Fitting) when the C/Ps' (Closure Plugs) are removed. The crud in the system effects the PRVs' (Pressure Reducing Valve) pressure settings controlling the M/T and F/T operations. The F/T side sees more crud as it gets removed from the channel by the fuel bundles in the defuelling operation. The crud problem gets worse the longer the unit is down.

To handle the crud problem a spare D_2O filter assembly has been installed on the D/M trolley of the F/T side handling the most F/T operations. This has reduced the problems with the PRVs'.

2.2.3 Slar Drive System

The SLAR drive system on each Delivery machine consists of fourteen (14) individual permanent magnet DC motors that can be operated with a variable speed and torque. Each motor has a tachometer and a brake (Except the Cable Reel). The motors are fed from three (3) DC buses using a multiplex system. Eight (8) of the motors are Contraves and six (6) are PMI. A number of problems have occurred with the PMI amplifiers and motors.

Amplifiers

The PMI Model #CX amplifiers output Transistors (PTC7001) and Optocoupler had failed on numerous occasions. The PMI amplifiers were obsolete and could not be repaired. The PMI amplifiers have been replaced with standard off the shelf Copley Control amplifiers. These amplifiers required an edge filter to increase the inductance in the motor circuit and an electronic potentiometer module to replace the built in peak current potentiometer.

The control OP's had to be revised to limit the torque because these amplifiers are capable of sustaining higher current. The total cost of this modification has been less than the original price of the PMI amplifier.

SLAR Umbilical Cable Drum Drive Motors

The cable drum (Fig. 8) must be controlled in synchronous with many other drives in order to keep the umbilical cable under a relatively constant tension while releasing or taking in cable. The control signal is derived by a single loop micro-controller fed with appropriate input signals. An LVDT (Linear Voltage Displacement

Transducer) is used to monitor the cable tension and its conditioned signal is fed to one input of the loop controller. During turret, telescopic or Z motions, the controller uses the feedback from the LVDT to readjust the cable tension. In M/T axial motion the controller will ratio the tachometer feedback from the module axial drive and use the LVDT signal to trim the motor speed to achieve the correct tension.

This control system places a heavy demand on the cable drum drive motor. The motor must achieve high acceleration rates and output torques as well as rapid changes in direction. Monitoring the tachometer feedback has shown the motor operating at two thousand (2000) RPM in one direction and two (2) seconds later running at two thousand (2000) RPM. in the opposite direction.

Numerous failures of the original cable drum motors led to a detailed analysis which revealed the following deficiencies:

- The motor power rating of 1.41 HP was less than the expected load of 1.6 HP.

- The motors were overheating due to the light thermal mass of the pancake style armatures with printed circuit windings and the high peak current.

- High pulsed current which were constantly reversing were suspected of partially demagnetizing the motor permanent magnets.

The motors were replaced with 1.66 HP units with conventional copper wound armatures and a higher current rating. These new motors have performed well with no failures to date.

2.2.6 Multiplexing System

The problem also existed with the design of the multiplexing system, the amplifiers were being powered before the motors contactor was picked up. This was suspected of causing some of the PMI motor failures. To correct the problem, timers have been installed that delay the power up of the drive amplifiers until the multiplexed motor contactor relay is picked up.

2.2.7 Vault Modifications for the Delivery Machines

Prior to operating the D/Ms' in the unit vault, it has to be modified to extend the machines. This requires removal of rails, catwalk and ACUs' (Air Conditioning Units) in the vault. The work requires about ten (10) days to complete during which the inspection equipment for the SLARing is also setup inside the vault.

2.3 Maintenance Problems

2.3.1 SLAR Tool Installation

The maintenance of any equipment inside containment poses a problem. The SLAR system has been no exception to this. The complexity of the system requires a number of electrical, mechanical and control systems to work together that complicates the diagnosing and maintenance of this equipment even more.

SLAR Tool Changes

The SLAR tool and umbilical cable must be replaced with a new unit several times in each outage. The SLAR tool is changed in a V-trough mounted on the trolley between the two (2) delivery machines. To avoid problems, tool changes must be completed by trained personnel following a complex and detailed procedure. Some of the problems encountered with tool changes at Bruce A are:

- The umbilical cable wrapped around the tether cable as it was fished through the M/T. This eventually led to the tether cable cutting through the umbilical cable.

- The coaxial and twisted pair signal cables from the eddy current and ultrasonic probes in the tool must be terminated using small, fragile connectors in an awkward location beneath the cable drum. Poor connections have led to erratic signals from the SLAR tool probes causing delays in the channel processing.

- The stainless steel sleeve and potted section of the umbilical cable are a very tight fit inside the M/T ram #1. Slightly oversize sleeves and out of round potting have made connecting the tool to the ram very difficult.

- The SLAR tool and umbilical cable are heavy and awkward to handle in the limited space available on the trolley. At least one back injury has occurred while attempting to manoeuvre the tool into position for installation in the delivery machine. To correct this, the working platforms on either side of the Calibration channel had to be lowered to provide better access to the V-trough to install the tool. Also, a jib has been installed to handle the tool.

The sag/droop developed in the M/T and F/T when left in the extended positions gave problems with sticky M/T and F/T operations. This was resolved by leaving the machines in the parked or collapsed stage. Minor problems resulting from these were difficult to diagnose at the time again because of the complexity of the machines.

The F/H SLAR personnel have been used to work on the equipment. The maintenance procedures had to be tested and revised in the field when the work was done as there was no mockup to train the personnel at the expense of time and dose.

SLAR was supposed to be a short term project, but has extended longer than expected. This poses problems with the availability of the spares such as amplifiers and motors for the drives and wearing of the equipment, as it was not designed to operate for this extended SLAR operations.

3.0 Conclusion

The SLAR equipment has performed well in carrying out the three (3) Slar campaigns on Unit 3 & 4. The system is scheduled for the SLAR campaign on Unit 3 in May of 1996 and Unit 4 sometimes in 1997. The duration of the SLAR project has created some problems with assigning permanent SLAR personnel to operate and maintain the equipment but because the personnel assigned for SLAR are from the F/H group the experience remains in the group.

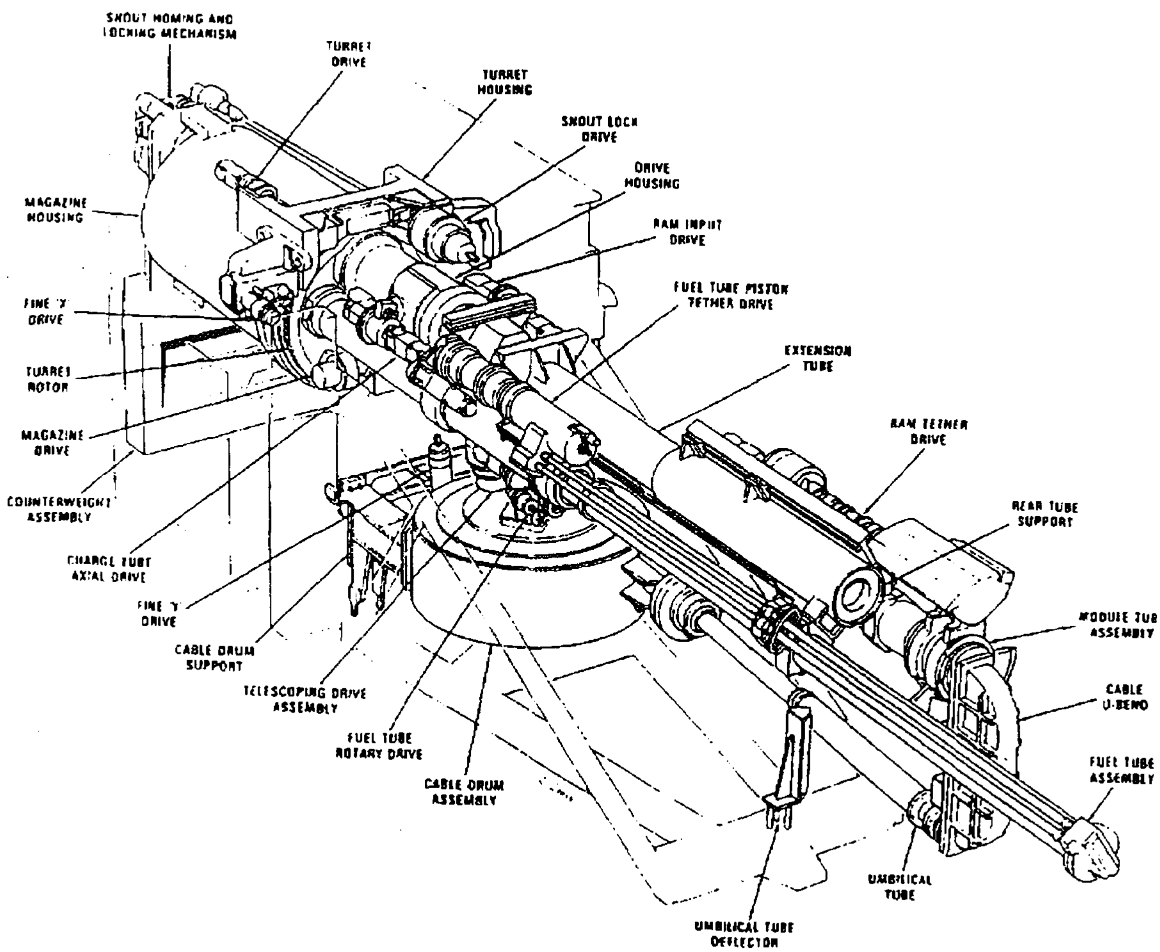

FIGURE 1. SLAR DELIVERY MACHINE HEAD

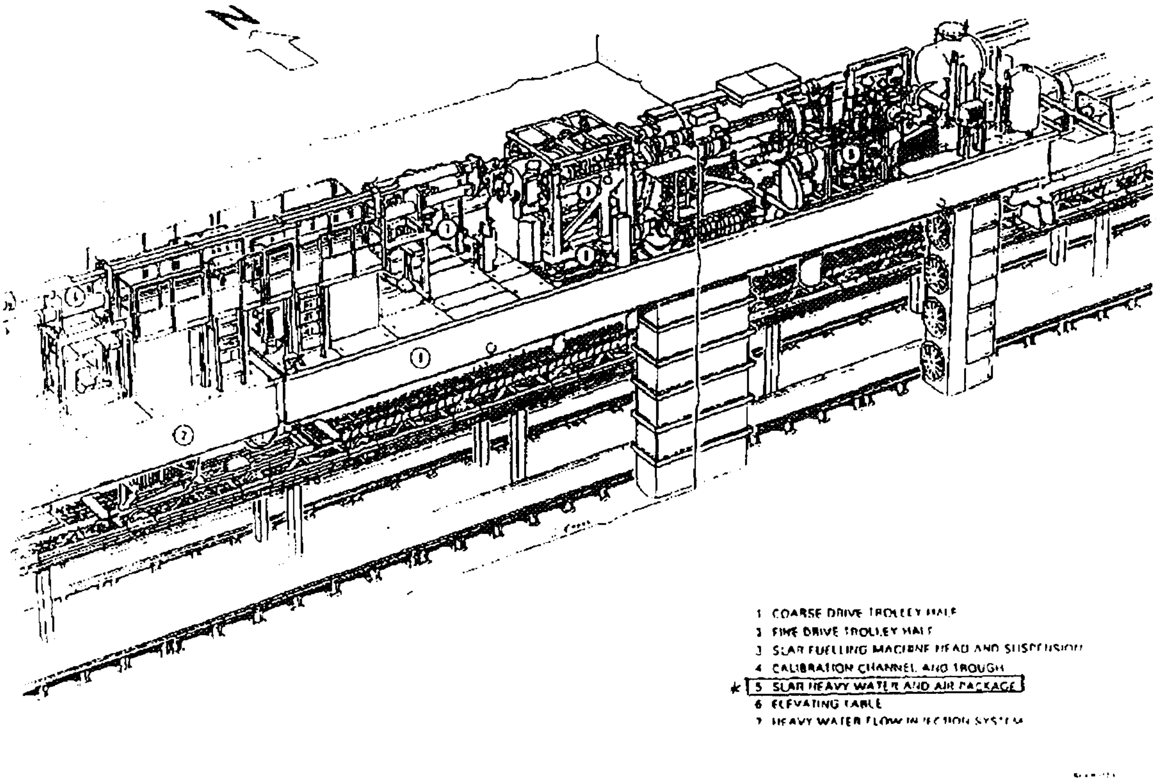

FIGURE 2. SLAR SYSTEM INSTALLED ON THE SOUTH FUELLING TROLLEY

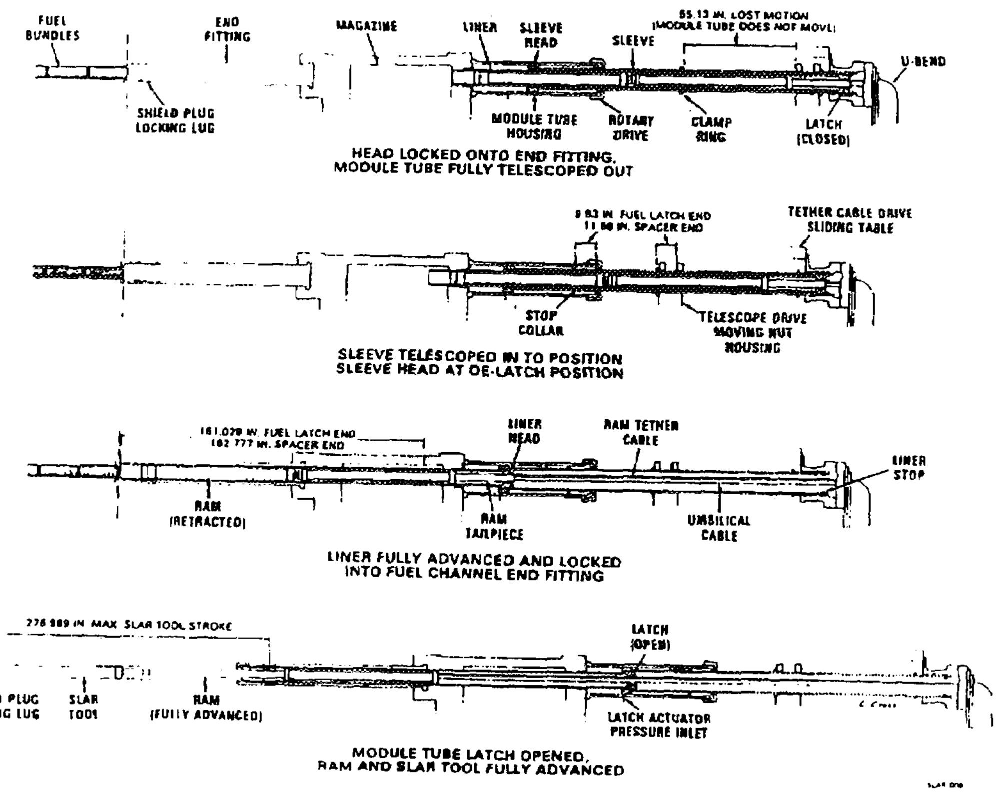

FIGURE 3. MODULE TUBE OPERATION

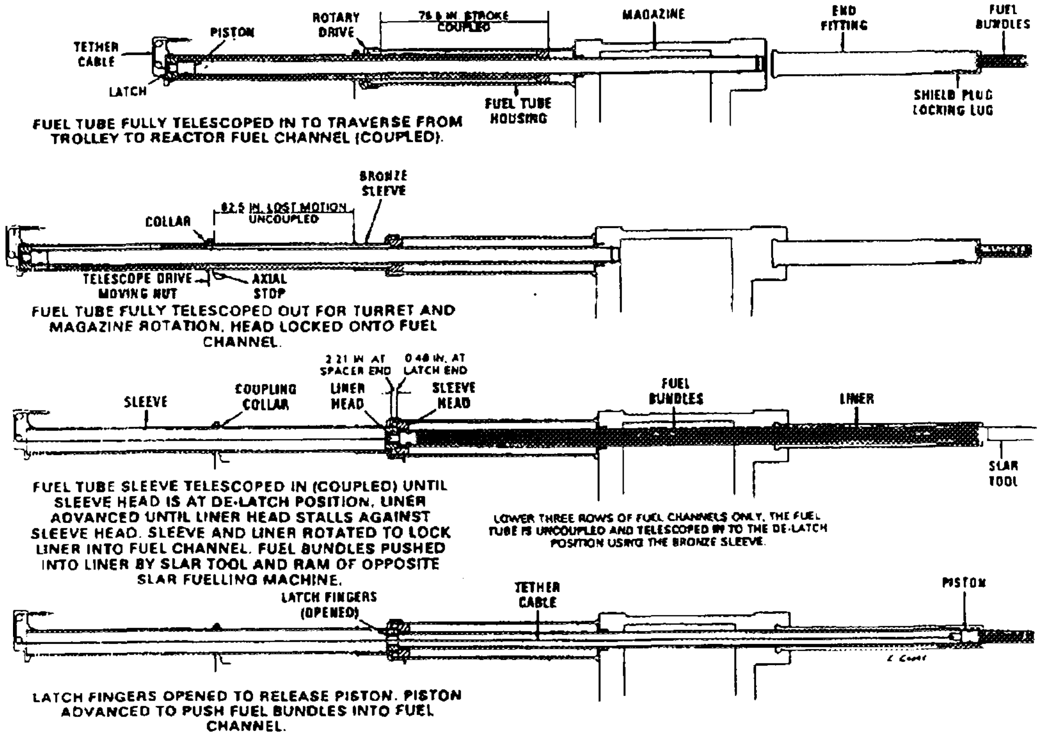

FIGURE 4. FUEL TUBE OPERATION

OVERALL DIAGRAM OF MARK III SLAR TOOL AND COMPONENTS

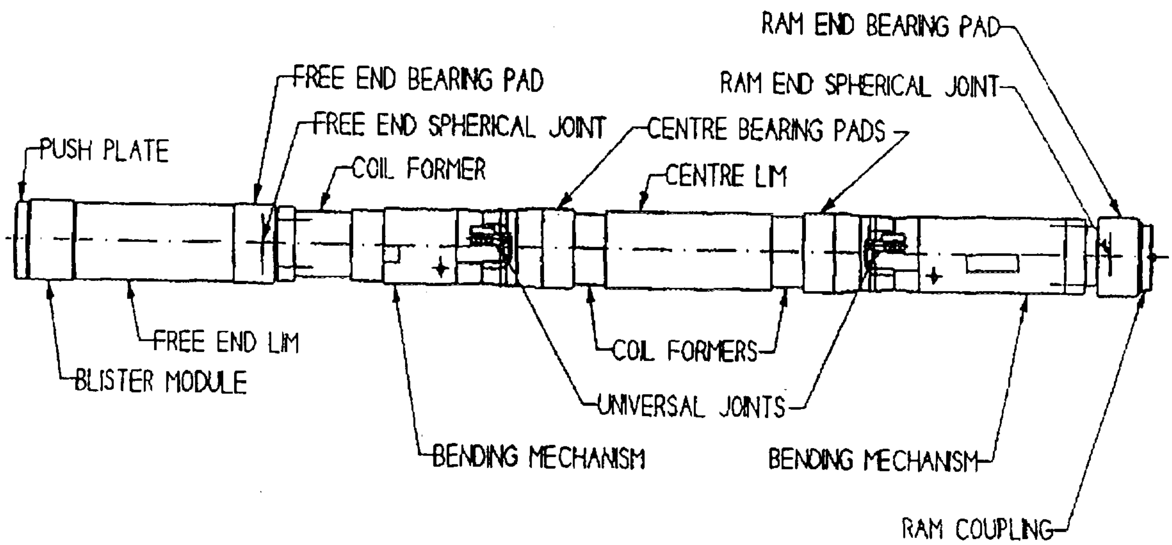

FIGURE 5. SLAR TOOL

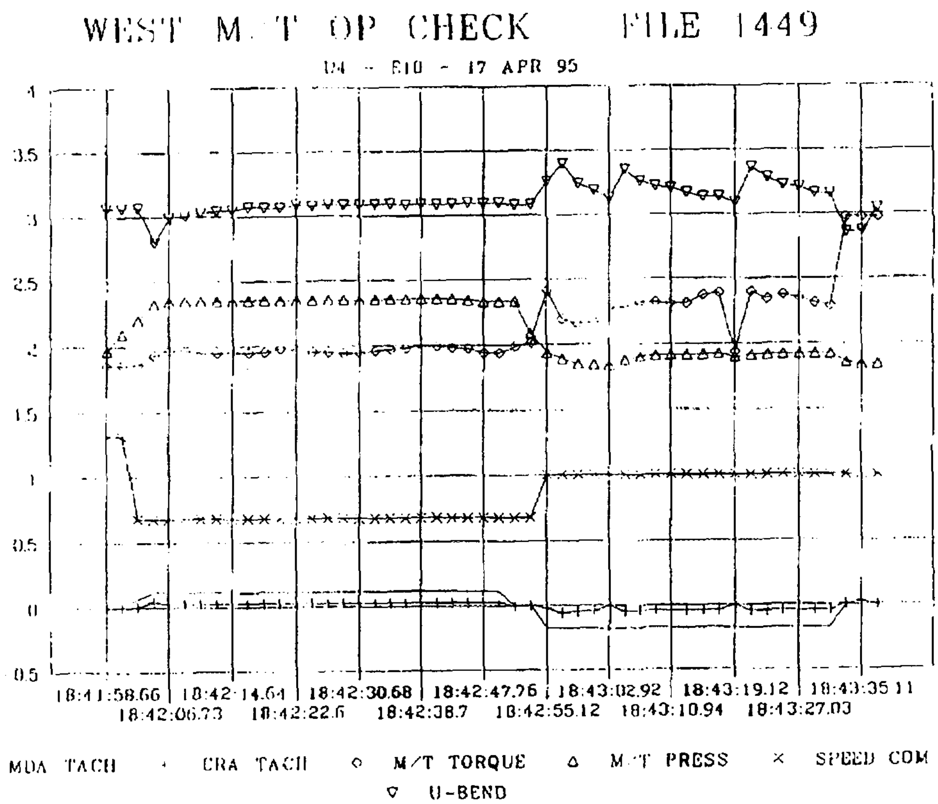

FIGURE 6. MODULE TUBE GRAPH (DATALOGGER)

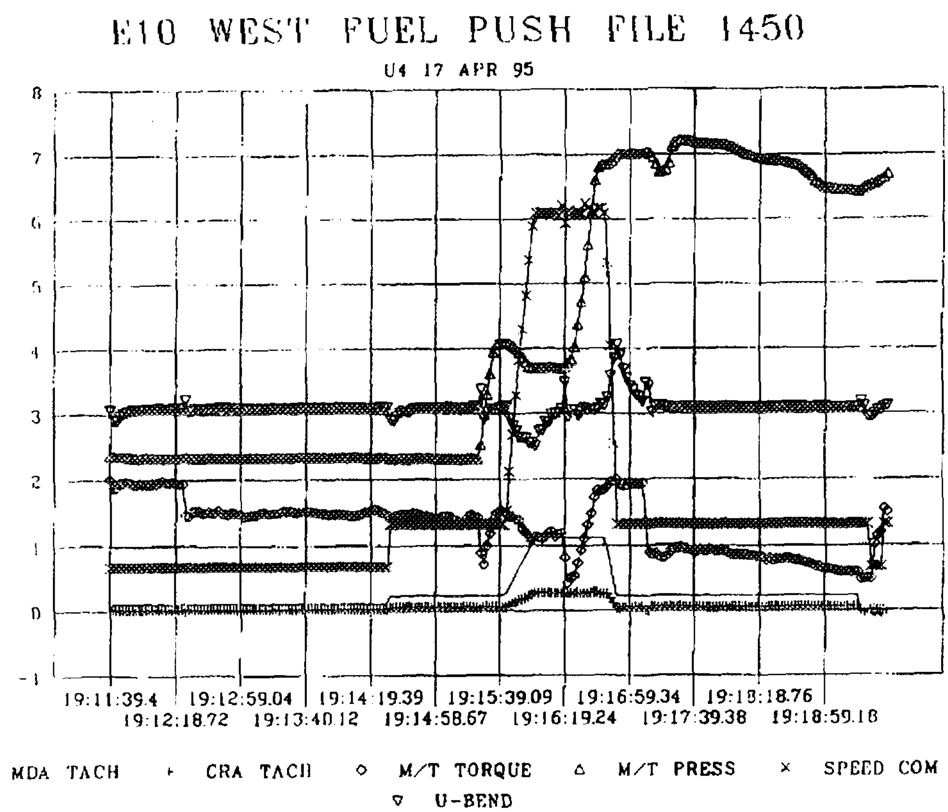

FIGURE 7. FUEL TUBE GRAPH (DATALOGGER)

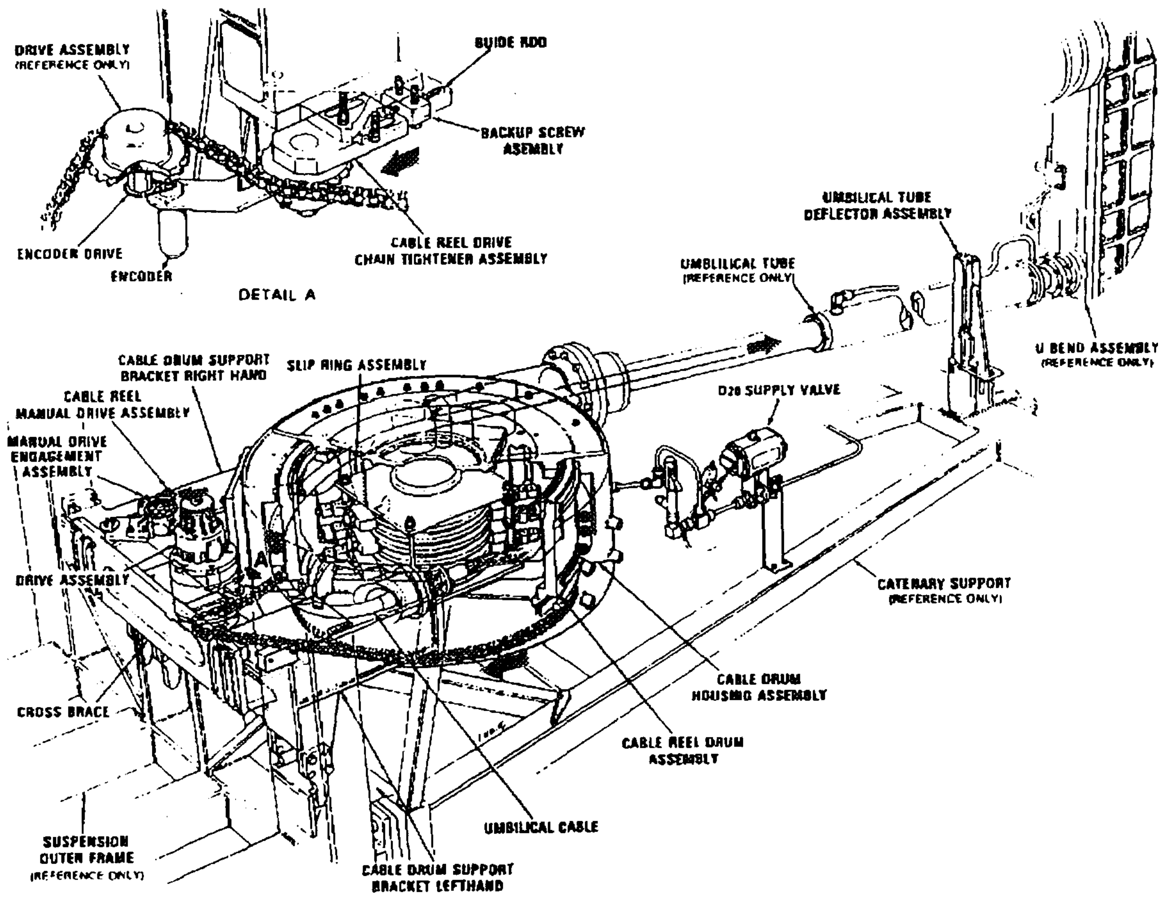

FIGURE 8. CABLE DRUM

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MINI-SLAR DELIVERY SYSTEM

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ABSTRACT

In the Spring of 1993, a need to complete Spacer Location and Repositioning (SLAR) on the Bruce 'A', Unit 1 Reactor was identified. An alternate SLAR delivery system was required due to conversion constraints that prevented the existing Bruce SLAR System from being used in Unit 1.

A Portable SLAR Delivery System called MINI-SLAR Delivery System was developed, designed and fabricated in a 14 month period, then used to successfully SLAR 109 channels.

The system is a portable remotely operated Nuclear Class I registered fitting that is independent of the Fuelling Machine, allowing the station to continue normal Fuelling and Maintenance activities. It is designed to a Level "D" faulted condition of HPECI Pressure thus minimizing PHT Heat Sink configuration requirements and minimizing outage set-up times. The system is based on a modular design allowing for easy fabrication, assembly and repair. It consists of a Snout Assembly, a Closure Plug Assembly, Shield Plug Assembly, SLAR Ram assembly, Work Table Assembly and Control Panel. Controls are through a Programmable Logic Controller with software tested and certified to a Software Quality Assurance of Level III.

INTRODUCTION

In the spring of 1993, the Canadian Nuclear industry was rocked by Ontario Hydro's decision to cancel the retubing of Units 1 & 2 Reactors at Bruce 'A'. It was decided that Unit 2 would be "laid up" beginning in September of 1995 and Unit 1 would be run until it met its Fuel Channel Life Limits.

In developing a "Life Cycle Management" Strategy for Unit 1 Fuel Channels, the station quickly realized that it must address the grave issue of potential pressure tube to calandria tube contact that results from channel annulus spacers moving from the design location, or face a shutdown of that unit as early as the Spring of 1997. At Bruce, Units 3 & 4, Spacer Locating and Repositioning (SLAR) has been quite successful and is performed using two specially converted fuelling machines. This system could not be used, however, as the Fuelling Machine Trolley could not reach Unit 1 and the equipment could not be converted to another trolley in time to meet project commitments. Thus was born the need for an alternate delivery system.

Although the need for SLAR of Bruce Unit 1 fuel channels was first identified following the cancellation of the proposed retubing of Unit 1 in the spring of 1993 work on a delivery system did not begin until the summer of 1994. In spite of a late start to the design process and the complex challenge of designing, building and commissioning a replacement for the SLAR delivery system in just 14 months, the project succeeded in completing SLAR of 109 channels in unit 1 in 31 days of SLAR processing at a very substantial saving to Bruce A. The purpose of this paper is to describe the process followed to develop the mini-SLAR system, to discuss the distinct aspects of this approach and examine what made this project successful. The technical description of the mini-SLAR mechanical design has been documented by others [1.] and this paper concentrates on the organizational aspects of the project.

The mini-SLAR delivery system is a small scale fuelling machine-like device. It was used in 1995 to deliver a SLAR tool into a fuel channel from tooling mounted on a fuel channel maintenance platform. The design requirements of the delivery system were to manually home and clamp onto a fuel channel under shutdown conditions, extending the heat transport pressure boundary, remove the closure and the shield plug and install a liner sleeve into the end fitting from either the inlet or outlet ends of a fuel channel. The target channel processing rate for the system was 2 channels per day. Both the SLAR processing and the delivery machine systems performed extremely well overall. The average time required to perform SLAR operations once on the channel was 1.8 hours, and the time required to move between sites was 1.3 hours, with an average 4.3 hours required for tool calibration, equipment maintenance, tool replacement and other support activities [2.]. SLAR processing rate was slow initially, and gradually improved as the crews became more experienced with the equipment and resolved some initial operational problems.
Ultimately the SLAR processing of Unit 1 was completed in less time than originally scheduled, and processed 25 percent more channels than originally planned. This has made it possible for Unit 1 fuel channels to continue to operate safely until the year 2000.

What distinguished the Mini-SLAR project from other similar tooling development jobs pursued at Bruce, such as West Shift, SLAR and Retube, was that the project management group was a site based team, highly accessible to and in regular contact with station personnel as well as the design organizations. The project team strongly influenced the design, but more importantly ensured that the design adapted to the needs of the site organization. The availability of existing retube tooling as a basis for some of the design and development gave the project a running start in the testing area, heading off many potential problems and guiding the direction of the project based on solid practical experience. The opportunity to assemble and play with simple and sometimes crude prototype equipment may not have been "doing it right the first time" but it led to some very interesting results.

- "Americans don't do it right the first time, they just don't. But when that second time comes around, not even the stars are out of reach."

"Doing it right the first time and zero defects may be expectations, but as messages to Americans they're more debilitating than motivating; they make Americans feel controlled and restricted."

from the book INCREDIBLY AMERICAN, RELEASING THE HEART OF QUALITY

CONCEPT DEVELOPMENT

Upon determining the need for a Delivery System, a "Skunk Works" Team of Engineers, Technical Staff and Trades was assembled. The team developed an approach of mounting a number of existing Retube Tools on a sliding plate. A fixed plate with a snout that attached to the End Fitting would remain in a permanent position with the sliding mechanism behind to align individual tools to an access port in the fixed plate. An 'O' Ring Seal would form the pressure boundary between boundary and the two plates. With little more than sketches, the two plates were fabricated and rudimentary cycle tests and pressure tests were undertaken.

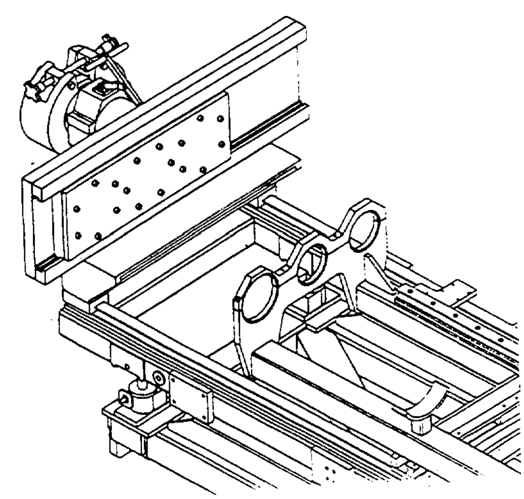

FIGURE 1
SLIDING PLATE
TEST ARRANGEMENT

Having passed these tests, the team now had a concept tool, a reference that was physical, against which improvements could be measured and could be used to validate the feasibility of the project. The team could now proceed with confidence in scoping out the project, seeking out proper approval for funding, then move on to designing and building a prototype.

PROJECT TEAM

The project team members had four distinct roles, to manage, design, build and implement. While one person was primarily responsible for each of these areas, the entire project team was collectively involved in key decisions and met frequently to review progress. The team consisted of staff with a varied background from fuel handling, operations, retubing, SLAR, fuel channel design, and mechanical maintenance.

Members of the project team brought widely varying individual view points and experience and as a result contributed to a broader internal review of the design which was constantly evaluated by the team in addition to more formal review by staff in the customer's organization. In addition, the team members had a wide and varied range of personal contacts. This provided a source of informal communication that was important to an understanding of the customer's need at all levels.

CUSTOMER CONTACT

The close working relationship between members of the project team and station staff resulted in ongoing formal and informal discussion of design requirements and design preferences. The articulation of design requirements required frequent meetings and discussion with station staff at every stage of the project.

Numerous design ideas resulted from discussion with station staff during design review, information presentations and training. More importantly, many of the staff who were eventually involved in the end use of the system, were also involved and had opportunities for input during the design and testing process. Frequently, the groups of staff visiting the training and mock up facility to witness prototype system operation, gained an impression that the design had already been completed and was not subject to change. This impression was discouraged. Suggestions for improvement were encouraged and acted upon whenever possible. The site organization was encouraged to develop a sense of design ownership and involvement.

USE OF EXISTING FACILITIES AND COMPONENTS

Site partnership in the design process was possible because of the existence of many tools and components that were used in the design and testing of the delivery system. Key site individuals with direct experience in developing tools of this type were involved in testing and experimentation with various equipment already available to the site. As the design progressed off site, it was possible for a parallel development effort to proceed at the station.

DESIGN FLEXIBILITY

It was necessary for the design to adapt to changing design requirements resulting from field development feedback and evolving customer requirements. Examples of this are the addition of shield plug and liner handling to the basic function of the system, the increase of design pressure from 5 to 8 MPa, shortening of the system to fit into airlock 2, as well as complete redesign of the support structure, the snout locking mechanism, and the sliding plate drive system to resolve performance problems. The use of a PLC control system to automate steps in the process allowed for the closure plug removal sequence to be modified and tested during the final weeks before the equipment was put into service. By implementing a modular design of hardware and software, it was possible for hardware and software development and testing to proceed in stages. The modular nature of the delivery system is illustrated in the following overview. This shows how the shield plug, closure plug and SLAR ram modules are independently mounted stand alone devices built into a modular support frame.

INFLUENCE OF PROTOTYPE TESTING

The use of a complete prototype system for testing and training was an invaluable aid to implementation as it allowed many potentially serious problems to be resolved ahead of time. Examples of this were the need for a Z drive brake on the shield plug removal tool, the need for rotary stops on the shield plug drive to prevent over torquing of the latch mechanism, and the identification of various design improvements to provide smoother, more reliable operation.

The control system was extensively proof tested on the prototype system to establish its performance characteristics. This proved to be vital to fine tuning the control software for smooth and consistent operation of the system. Although apparently straight forward, the protective stops and interlocks built into the controls for equipment protection created practical performance problems which were resolved through a process of testing and modification. A list of hardware and software deficiencies were gradually resolved through a process of realistic full scale simulation of the field operation of the equipment.

Training of 40 fuel handling operators and mechanics on the operation of the system was possible using the realistic rehearsal facilities used for prototype testing. These included a full scale mockup of the reactor vault, maintenance platform, and F/M bridge full length pressurized channels as well as dry cutaway channels for viewing the internal operation of the equipment, and a remote control console, complete with viewing and communication systems.

Integration testing using the actual SLAR inspection system was made possible by the parallel development of a transportable SLAR control trailer. The interface with the SLAR control system was designed to emulate the existing fuelling machine interface to provide the least possible redesign of the existing SLAR system.

Bruce A has been historically dependent on outside sources for design support and has not developed a strong design capability or culture. Site is more comfortable with practical results rather than a dependence on analysis and procedures of design. From the outset, the site team based many decisions on actual performance of field equipment, and on the intuition and judgement of field personnel rather than accepting the paradigms of a traditional design approach.

The sliding plate mechanism is a case in point. This is an unconventional design, not recommended by various designers, but supported by the site team as a simple solution to the problem of tool indexing. In spite of concerns raised by experts to the advisability of this approach, the site team went ahead to fabricate and demonstrate the design. In a more traditional conservative design culture this simple approach would have been rejected although in practice it has performed reliably with minimal development. This points out how a risk averse tendency in our traditional design approach can result in missed opportunities. When necessity is the mother of invention, some interesting and useful discoveries can be made.

Is this just luck? Is it equally likely that in a future design project, this design approach would fail to produce acceptable results? The answer to this question may be yes, but if we look at the track record of the "conventional" design approach we see that there are any number of unsuccessful designs that have failed to produce results in spite of the best available design talent being brought to bear.

We believe that in the field of nuclear station maintenance equipment design, the most demanding problems are not technical ones. The technology associated with what we need to do is not only readily available, but in most cases it has become relatively commonplace. The most important issue is design acceptance by field personnel and the field implementation organization. As designers, by focusing on more elegant technical solutions we risk alienating the people who have to use our products at the end of the day.

The Mini-SLAR project is an illustration of an alternative organizational approach intended to address field involvement and acceptance issues. There were both positive and negative aspects to the project, and some important lessons to be learned.

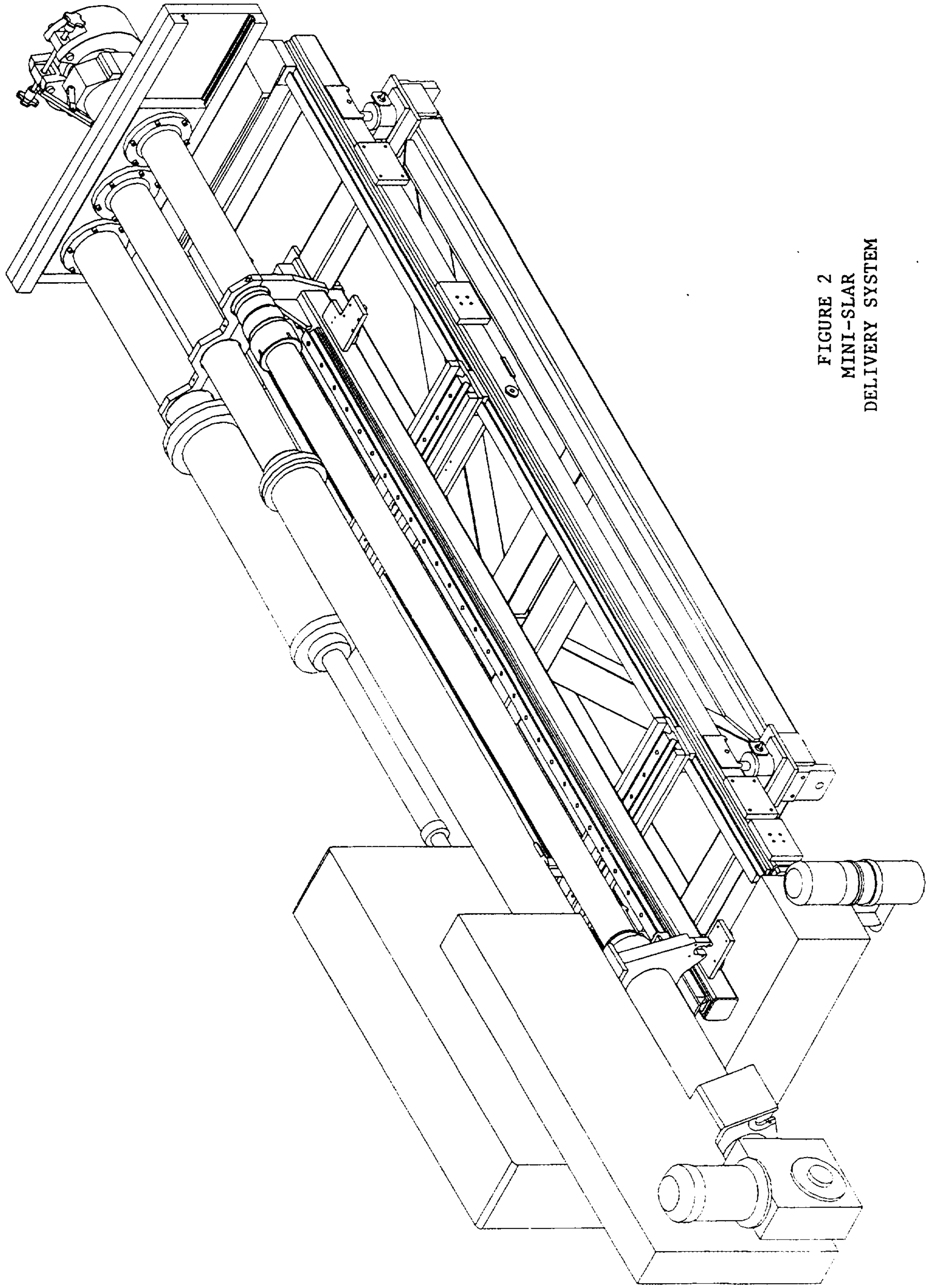

FIGURE 2
MINI-SLAR
DELIVERY SYSTEM

The project team was proud of the fact that in spite of a short implementation schedule, it was able to deliver a system capable of replacing the existing fuelling machine based SLAR delivery system, with a project justification, design build and test duration of approximately 14 months.

The business case for Mini-SLAR was based on a need to SLAR an estimated 110 fuel channels in Unit 1 by spring of 1995 or face the need to perform a series of CIGAR inspections, scrape samples and fuel channel replacement of 27 channels at risk. Based on a realistic estimation of the cost of channel inspections and replacements, the mini-SLAR project was expected to save Bruce A a significant capital cost associated with channel replacement, and 100 outage days based on then current multi fuel channel replacement techniques. The actual performance of the system was somewhat better than expected. A total of 109 channels were SLARed in 31 days of SLAR processing time, exceeding the target rate of 2 channels per day by a significant margin. All but one channel required spacer repositioning to extend its operating life [2]. An estimated 27 channel replacements were avoided, and work was completed with minimal impact on the critical path duration of a planned 103 day boiler chemical clean outage. Total direct and indirect savings to Ontario Hydro are considerable and Hydro has gained some very valuable experience.

Like any successful job there were many hurdles to overcome, the most demanding one was project schedule. Many activities had to be performed in parallel in order to meet the required completion date. The resulting process operated in the reverse of the usual design process. Fabrication and testing preceded the formal design activity, and the GE Canada and NTS SLAR design teams had to frequently adjust their priorities to respond to the practical requirements of the fabrication and testing schedule. Equipment was assembled and tested in an emerging prototype form and the design organization, while trying to analyze and formally document the design was at the same time responding to frequent requests for changes from the field organization. The flexibility demonstrated by our key designer, GE Canada, while no doubt frustrating for them, was key to ensuring the final product was accepted in the plant. In spite of the effort made to maintain frequent contact with the design group, it was apparent that the design process could have been more efficient if the designers had been working together with the fabrication and testing team at the station. The circumstances of this project allowed the design and field organizations to be on a more equal basis than most design build projects. From the beginning, the field organization had a rough prototype to work with using various retube equipment on hand. Whenever possible existing components were used to reduce the design and test time. Significant upgrades of the equipment were required but many of the basic mechanical concepts were available for testing before the design process started. This allowed the field team to begin hands on experimentation and to provide early feedback to the designers to correct various deficiencies in the equipment and to refine the understanding of the operating characteristics of the system. Valuable design input was provided by the field organization based on the experience with the mechanical prototype system that contributed to a more useable piece of equipment when it went into service. The control system, and the SLAR inspection system were dovetailed into the design later in the project schedule, and required an intense dedicated effort by site staff as well as by NTS SLAR site team and GE Canada designers. Because of the modular nature of the design it was possible to marry the control systems for the delivery system and the SLAR inspection system with a minimum of testing problems. Much of the credit for this accomplishment is due to the effective communication between GE Canada and NTS SLAR designers who developed a detailed interface specification at an early stage in the design.

SITE FABRICATION AND TESTING

Fabrication and testing of the prototype delivery system at site resulted in a much greater site awareness of the project although not everyone was comfortable with the way the work was allocated. Half way through the fabrication and testing process, the team had to negotiate a revised work assignment to reflect the emerging trend toward greater operations staff participation in projects of this type. This was disruptive to progress, but in the end the staff who were involved in the development and testing were able to be closely involved in the field support of the equipment and were largely responsible for finding and efficiently solving problems during implementation. The construction and operations trades staff involved in this job worked closely and effectively together exhibiting a high standard of professionalism and dedication.

At the time the Mini-SLAR system was built the Bruce site organization was not authorized to fabricate this type of nuclear class 1 equipment. However, the MCCR granted a concession for the assembly and testing of the pressure boundary, and the fabrication of the NF support structure for this project. In future, a site fabrication certification will make the assembly of this kind of equipment more practical. The advantages of site fabrication and assembly were the responsiveness to schedule requirements, direct supervision and high standard of workmanship. The fabrication of the equipment on site contributed to a sense of design ownership and a highly effective review of manufacturability of the design. The hands on involvement of field personnel led to a detailed understanding of the maintenance requirements for the equipment.

INDEPENDENT OPERATION

The use of the station fuel handling systems for reactor maintenance and inspection has been a common practice to the extent that the fuel handling system availability for fuelling is becoming restricted. The original strategy considered for Unit 1 SLAR was based on extensive use of the fuel handling system for shield plug and liner sleeve handling. This was reviewed with station fuel handling staff at the outset of the project and it became immediately apparent that a design strategy with the least involvement of station fuel handling systems was a basic design objective. Ultimately the mini SLAR system was able to replace the fuelling machines for all aspects of the project except fuel removal and replacement. Defuelling was completed as a batch process using flow assisted defuelling methods in a period of 17 days following shutdown. Fuel replacement was accomplished by transferring fuel from Unit 2 during the Unit 2 layup defuelling at a substantial fuel cost saving.

Independence from station systems was assured by providing independent power supplies, and the capability of operating on channel independent of the state of the heat transport system. The system was designed to withstand fuel channel pressures up to 8.3 MPa and was capable of operation of up to 1.38 MPa. This made it possible to perform SLAR operations during all phases of boiler chemical cleaning. Defuelling at the start of the outage also made the vault accessible throughout the balance of the outage up to the start of refuelling. This provided unrestricted access for other outage work to proceed in parallel.

FUTURE PROJECTS

The mini SLAR system has numerous potential future uses. In the immediate future it is being adapted for mounting in the fuelling machine carriage with a remote homing and locking capability, for use in draining Unit 2 during lay up. In addition to improving drawing efficiency this equipment is also useful in providing pressure relief during fuel channel isolation. This alleviates the need for the F/M during channel isolation. A second mini SLAR system will be built and used to perform SLAR on Unit 4 in 1997, using two SLAR systems in parallel. As an upgrade for this project, the Unit 4 system will be designed to defuel the Unit 4 channels by pushing fuel into a fuelling machine at the opposite end of the channel. This method of defuelling allows for economical fuel shuffling, avoiding the disposal of partially utilized fuel. Defuelling during a shutdown with this technique is much more efficient than the fuel grappling alternative. It avoids the need to operate the heat transport pumps to permit flow assisted defuelling, and can be performed from either the upstream or the downstream end of a channel.

The use of a mini-SLAR type device for general fuel channel maintenance and inspection has many advantages over current practice. The equipment can be picked up by the fuelling machine bridge, and requires a very little vault time to install. Once in place, the equipment can deliver a variety of special purpose tools to aid in inspection and maintenance activities while placing few demands on the existing fuel handling system. Since the delivery system is modular in design, it can be easily reconfigured to accommodate special purpose devices in place of or in addition to the existing modules. Examples of some potential future applications are the delivery of tools for channel inspection and gauging, isolation, closure seal repairs, pressure tube wet scrape, fret blending and replication, and visual inspection. In the longer term similar devices may be used to assist in fuel channel replacement.

CONCLUSION

Site based project teams are assuming an important role in providing specialized design and equipment needs for OHN. The mini-SLAR project demonstrated how a site project management team can be more responsive to the specific needs of an individual project, and provide greater levels of site involvement in the design while maintaining a high standard of quality. Emphasis on using available resources to speed the development cycle, keeping the design simple and flexible and encouraging very broad participation by the user organization were key elements of the success of this project. Future projects should address the need for qualified site fabrication capability, and a closer working relationship between the design organization and the fabrication and testing group. These factors would improve the efficiency of the design feedback process and ensure an effective site fabrication and testing process

REFERENCES

1. Design of the Mini-SLAR System for Bruce A, M. Gray, GE Canada Nuclear Products, Candu Maintenance Conference Oct.,1995.

2. Mini-SLAR of Bruce NGS A Unit 1 Pressure Tubes in 1995, S. Motomura, Ontario Hydro NTS SLAR, ISMP-BGA-00773.1-14, Dec 31, 1995

CANDU 9 FUELLING MACHINE CARRIAGE

CA9700730

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ABSTRACT

Continuous, on–power refuelling is a key feature of all CANDU reactor designs and is essential to maintaining high station capacity factors.

The concept of a fuelling machine carriage can be traced to the early CANDU designs, such as the Douglas Point Nuclear Generating Station. In the CANDU 9 480NU unit, the combination of a mobile carriage and a proven fuelling machine head design comprises an effective means of transporting fuel between the reactor and the fuel transfer ports. It is a suitable alternative to the fuelling machine bridge system that has been utilized in the CANDU 6 reactor units.

The CANDU 9 480NU fuel handling system successfully combines features that meet the project requirements with respect to fuelling performance, functionality, seismic qualification and the use of proven components. The design incorporates improvements based on experience and applicable current technologies.

1. INTRODUCTION

CANDU reactor operation is based on the concept of continuous, on–power fuelling, which is carried out by the remotely controlled Fuel Handling System. The fuelling machine head, a pressure vessel which contains new or irradiated fuel during its transport between the reactor and the transfer ports, can be supported either by a bridge system, such as that used in the CANDU 6, or by a mobile carriage such as in Douglas Point NGS. For the CANDU 9 480NU design, a carriage system has been adopted.

An important consideration in the selection of the carriage system for CANDU 9 was having a design that would lend itself to seismic qualification for future sites with potentially higher seismicities than those of the previously constructed CANDU stations. Other design goals included reduced construction time and improved on–power maintainability of the carriage and the fuelling machine head. Simplicity and the maximum use of standard, commercially available components were also a part of the design intent. Supplier and operator input was incorporated.

In the following sections, an outline of the requirements applied to the fuelling machine carriage concept is provided, along with a detailed description of the carriage system functional and structural design. A summary of the benefits of the carriage system is included.

2. DESIGN REQUIREMENTS

The key CANDU 9 480NU project requirements, related to the fuelling machine carriage system performance and functionality, included:

i) meeting the fuelling demands of the CANDU 9 480NU reactor in support of the target lifetime capacity factor of 90%;

ii) access, with the required positioning accuracy, to all fuel channels, the new and irradiated fuel transfer ports, the rehearsal facility and the ancillary port;

iii) drive mechanisms compatible with computer control in the fully automatic, semi automatic and manual modes;

Design requirements related to safety and licensing were:

iv) meeting the requirements of the CAN/CSA N285 series of standards for Class 1C pressure boundary component supports, with references to the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF;

v) application of quality assurance programs during design and construction, based on the CSA CAN3–N286 and Z299 standards;

vi) carriage design licensable in Canada;

vii) seismic qualification to ensure structural integrity during a 0.2 g Design Base Earthquake (DBE);

viii) providing safe and reliable transport of the fuelling machine head, thus ensuring the integrity of the contained new and irradiated fuel and of the interfacing components;

ix) meeting the environmental qualification requirements for normal operating conditions and postulated accident conditions;

x) satisfying the current human factors requirements with respect to the system layout, equipment maintenance and overall minimization of dose expenditures;

Other requirements:

xi) minimizing construction time.

3. CARRIAGE SYSTEM DESIGN

The design of the CANDU 9 480NU carriage is functionally based upon the Douglas Point NGS carriage. There are two dedicated fuelling machine and carriage assemblies per reactor that operate in concert to fuel a single channel. Correspondingly, there are two independent fuel transfer systems to handle new and irradiated fuel.

Each CANDU 9 480NU carriage (Figure 1) operates between the fuel transfer ports inside the maintenance lock and the fuelling machine vault. Once in the vault, the carriage turntable rotates 90° so that the fuelling machine faces the reactor. The carriage positions the fuelling machine head at the selected lattice location in the X (lateral) and Y (vertical) directions, engages the seismic locking mechanisms, and then advances the fuelling machine head in the Z direction. Once the head arrives at the pre-stop position, X and Y correction measurements are established. If necessary, the head is retracted clear of the end fitting and X and Y homing corrections are carried out by the fine X and carriage Y drives, respectively. The repositioned fuelling machine is then advanced to contact the end fitting with sufficient force to stall the Z drive. At this point, a standard CANDU fuelling sequence commences.

The carriage is designed to the requirements of the CAN/CSA N285.0, General Requirements For Pressure Retaining Systems and Components in CANDU Nuclear Power Plants and CAN/CSA N285.2, Requirements for Class 1C, 2C, and 3C Pressure Retaining Components and Supports in CANDU Nuclear Power Plants. These standards recognize the unique design of CANDU fuel handling systems and allow pressure vessel supports such as the carriage to be designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. Through conformance to these design standards and codes, the carriage system becomes eligible for licensing in Canada.

The carriage assembly structure consists of six main sub-assemblies: the base, turntable, columns, outer elevator, inner elevator, and guide plate (Figure 1). The movements of the carriage sub-assemblies relative to one another and with respect to fixed supports facilitate the fuelling machine positioning at the reactor. The fixed supports consist of a floor rail and an upper guide beam. The carriage rail runs parallel to the reactor face and into the fuelling machine maintenance lock. The upper guide beam is centered over the rail, spanning its full length. The guide beam supports the carriage structure in the Z direction by trapping the guide plate within a slot in the bottom of the guide beam.

The base is a structure on two wheels that provides the X motion and half of the rotation freedom of the carriage. It is a welded assembly consisting of two beams, internal bracing and a circular skirt. The beams run parallel to the rail with cross members spanning the rail. Additional bracing, perpendicular to the main members, supports the skirt. One wheel of the base is driven by an electric motor through a reduction gear train, while the second wheel is an idler that, when required, can be driven by the backup X drive. The base is guided along the rail by guide rollers that contact the outside edges of the rail. Hold down hooks, that run in slots in the sides of the rail and are bolted to the main beams, prevent lifting of the carriage during an earthquake. The skirt supports a mounting ring at the top and four outriggers at the bottom. The outriggers are provided for construction and maintenance purposes and are capable of supporting the carriage assembly independently of the wheels. Finally, the inner race of a slewing ring is bolted to the mounting ring at the top of the skirt.

On top of the base, the outer race of the slewing ring is bolted to the underside of the turntable. The outer race is manufactured with an external gear cut in its outside perimeter. For carriage rotation, this gear meshes with a pinion fixed to the base through another reducer and an electric motor. The turntable structure is of a welded, sandwich style construction. It has internal cross-bracing that runs diagonally between the main members and it is enclosed on top and bottom with steel plates. Access openings are provided for the construction of the turntable, and for inspection and replacement of the bolts between the inner slewing ring cage and the base.

The column sub-assembly is bolted to the top of the turntable and it consists of two vertical columns and two horizontal crossbeams (Figure 2). The vertical columns are internally braced and have Y guide tracks along their front and back sides. The internal bracing at the bottom of the columns is in an X pattern facing upwards, providing high torsional rigidity. The bracing in the active portion of the columns is arranged in a zig-zag pattern with the edges of each brace corresponding to a reactor lattice position. These angled braces support loads imparted by the outer elevator Y guide bearings to the columns through the Y guide tracks. The top outside stub of each column houses a radial thrust bearing which holds the ballscrew. The lower stub on each column supports the ballscrew tensioner, while the Y drive seismic brake and gear reducer are bolted to the outsides of the columns below the stubs. The Y drive electric motors and drive trains are mounted on the turntable for improved access. The lower horizontal crossbeam between the two columns serves as a mechanical lower stop position for the outer elevator. Permanent shielding is bolted to the lower crossbeam that will shield the underside of the fuelling machine head magazine during maintenance. The upper crossbeam houses the bearings which in turn hold the shaft of the upper guide plate. These bearings provide a pivot point to allow rotation of the turntable and the columns relative to the base and the upper guide beam. The mating surfaces between the upper crossbeam and the columns are provided with keys that limit seismic shear loads through the bolts.

The outer elevator is positioned between the columns and allows the vertical movement of the fuelling machine. The outer elevator consists of a frame, four pedestals, outer crossbeams, a Z drive mounting beam and Z drive tracks (Figure 3). The frame fits between the columns with extensions on the sides of the frame protruding ahead of and behind each column. The pedestals are bolted to these extensions and they house the Y guide bearings. The Y guide bearings provide support to the outer elevator in both the X and Z directions by riding along both the face and sides of the Y guide tracks mounted on the columns. An outer crossbeam spans between the front and back pedestals along the outside of each column, thereby surrounding each column. This outer crossbeam is the mounting point for the Y drive ballnut. The Z drive mounting beam bolts across the back two pedestals and serves to transfer Z drive reaction and seismic forces from the fuelling machine to the columns through the pedestals and Y guide bearings. The Z drive tracks are mounted on the bed of the outer elevator and provide a track for the inner elevator to ride along. The Z drive is an Acme screw which is turned by an electric motor through a reducer. The screw, while rotating, drives an Acme nut along its length. The nut itself is fixed to one end of a seismic spring pack which in turn is fixed to the inner elevator. The spring pack is comprised of a tuned stack of hard Belleville spring washers. Its purpose is to allow relative movement of the fuelling machine with respect to the carriage, thereby limiting the carriage seismic inertial forces imparted to the end fitting should the fuelling machine be clamped on to an end fitting during an earthquake.

The inner elevator moves relative to the outer elevator to provide the Z motion of the carriage. The inner elevator consists of a base, a yaw turntable, and a cradle support. The base has linear guide bearings along the bottom that run on the top and sides of the Z tracks. The yaw turntable is a smaller slewing ring that has its inner race bolted to the top of the inner elevator base, and its outer race bolted into the bottom plate adjoining the cradle support pillars. The support pillars house the pitch pivot trunnions of the fuelling machine cradle and the cradle moves laterally within the support pillars to provide the fine X motion. Spring centering mechanisms between the fuelling machine cradle and the cradle support pillars, and between the cradle support and the inner elevator base provide the pitch and yaw freedom necessary to allow a fuelling machine to center over an end fitting.

4. SEISMIC DESIGN FEATURES

Undesirable motion of the CANDU 9 480NU carriage under seismic conditions is prevented through several means. The first restraint mechanism is an arrangement of wedge and pin style seismic locks (Figure 4). When activated, these mechanical devices prevent rotation and translation by locking the carriage sub–assemblies to one another or to fixed supports. The seismic locks are engaged during every channel refuelling sequence, prior to the fuelling machine being advanced over the end fitting. Both types of locks employ electric motors and Acme screw drives, and readily accessible manual drives for recovery operations are available.

The wedge style seismic locks accommodate thermal expansion between interfacing sub-assemblies and/or fixed supports. The locks employ force control feedback through the Acme screw drives to limit the depth of lock engagement. The force control establishes a positive contact between the locking wedge face and the locking striker plate but it limits the force in order to prevent any tendency of the lock system to move the carriage. The lower arrangement consists of two locks bolted back to back and mounted at the reactor side of the carriage base. These two locks then engage locking holes in a floor embedment. At the top of the carriage, one lock is bolted to the back of each column and it engages a corresponding hole in the upper guide beam. The locking holes in the upper guide beam and floor plate are spaced one lattice pitch apart. Due to the various angles at which the carriage must operate in the fuelling machine maintenance lock, a radial pattern of locking holes exists in that section of the upper guide beam. Because the base does not rotate, only a single embedment with two locking holes is sufficient to lock the base to the floor when the carriage is in the maintenance lock.

The locking of the turntable to the carriage base is accomplished with a straight pin style seismic lock. The straight pin style is used only in this location because the two sub-assemblies, given their physical proximity, will be subject to the same thermal transients, thus leading to minimal misalignment. One lock is located at the base of each column. The pin extends through the bottom of the turntable and engages a mating hole in a locking ring welded to the base structure within the slewing ring.

In addition to the wedge and pin type seismic locks, all electric drive motors have integral brakes which are automatically applied when the drive is de-energized, making them fail-safe. For further safety, the ballscrews are equipped with seismically qualified fail-safe brakes and are pre-tensioned with spring packs that prevent compression of the ballscrew under seismically induced lifting loads.

In order to protect the fuelling machine carriage assembly from an earthquake occurring while the carriage is in transit, a seismically activated and qualified switch will cut the power to all drives, thereby activating their brakes. As a backup to the motor brakes, seismically qualified snubbers and mechanical stops prevent overtravel and provide for controlled deceleration of any component should the motor brakes slip.

5. DESIGN BENEFITS

The benefits of the carriage design range from simplified construction to improved reliability and reduced time and radiation dose expenditures during operation and maintenance activities. The new design incorporates feedback from operators and suppliers. The CANDU 9 480NU design is also expected to be adaptable for CANDU sites with seismic levels higher than the 0.2 g DBE currently being considered.

Manufacturing tolerances are eased by the use of shims between sub-assemblies because customized shims bring critical sub-assemblies into alignment. Along with these shims, simple bolted connections between sub-assemblies allow for a shorter carriage set up time at site. Procurement, manufacturing and construction of the carriage are all simplified by maximizing the use of standard commercial parts. Additionally, the selection of proven electric motors as the prime movers has led to a significant reduction in the number of mechanical parts required for the CANDU 9 carriage in comparison to the CANDU 6 carriage, largely due to the elimination of the carriage oil hydraulic system.

The reliability of the CANDU 9 carriage is expected to be improved over that of the CANDU 6 carriage. The structure is inherently robust, with all carriage components that support the fuelling machine being designed to accommodate loads resulting from a DBE. Given that the normal operating loads are, typically, an order of magnitude lower than seismic loads, the mechanical equipment of the carriage is expected to have considerable operating lifespan. In order to further improve the system reliability, a backup X drive is provided and a single Y drive unit can safely lower the elevator and fuelling machine head assembly. As well, one seismic brake and one ballscrew can support the entire suspended mass. Other features that enhance safety and reliability include an increased Z drive range for recovery purposes; if required, the full Z drive range will allow the recovery of a fuelling machine with a guide sleeve stuck in the snout, even with a crept pressure tube.

Maintenance activities, and the anticipated radiation dose expenditures attributable to these activities, were a significant consideration in the selection of a carriage system. The CANDU 9 480NU mobile carriage system allows for carriage maintenance to be carried out inside the shielded and environmentally isolated maintenance locks, rather than in more active locations such as the fuelling machine vaults. On–power maintenance of the carriage is possible, a feature that positively contributes to the station capacity factor as carriage maintenance does not require a reactor shutdown. Other improvements are obtained through ready access to carriage components and the use of bolted component connections. Bolted connections facilitate the removal of sub-assemblies that can then be taken to low activity locations for maintenance. For field work, strategically placed shielding for the fuelling machine head will be incorporated into the carriage structure so that fuelling machine head maintenance can be carried out with less dose commitment than in the previous CANDU designs. Finally, through the use of commercially available parts, carriage maintenance can be performed by less specialized work groups, thus enabling dose equalization programs to succeed.

6. SUMMARY

The CANDU 9 480NU fuelling machine carriage design represents a departure from the bridge system utilized in the successful CANDU 6 reactor design. The carriage concept is not new, however, as its origins can be traced to the earlier CANDU reactor designs such as that at Douglas Point Nuclear Generating Station.

The two wheel fuelling machine carriage design, described in the foregoing sections, meets the *functional, performance, safety and constructability* requirements of the CANDU 9 480NU project. The combination of a robust CANDU 9 carriage, fully integrated with the proven CANDU 6 fuelling machine head, has the capacity to support the station lifetime target capacity factor of 90%.

The carriage design has the potential to satisfy the requirements of future CANDU sites with higher than 0.2 g DBE levels. Other significant benefits include on-power maintenance access to both the carriage and the fuelling machine head, and the use of commercially available, standardized components. These features are intended to minimize dose expenditures and the costs associated with fuel handling system maintenance. In addition, a high degree of pre-assembly is planned in order to shorten in-situ construction and the commissioning time requirements.

The carriage system, as a pressure boundary component support, meets the requirements of the applicable CSA standards and ASME Code requirements. It is designed to be licensable in Canada, as well as other comparable international jurisdictions.

ACRONYMS

CANDU – Canada Deuterium Uranium

ASME – the American Society of Mechanical Engineers

CAN/CSA – Canada / Canadian Standards Association

DBE – Design Basis Earthquake

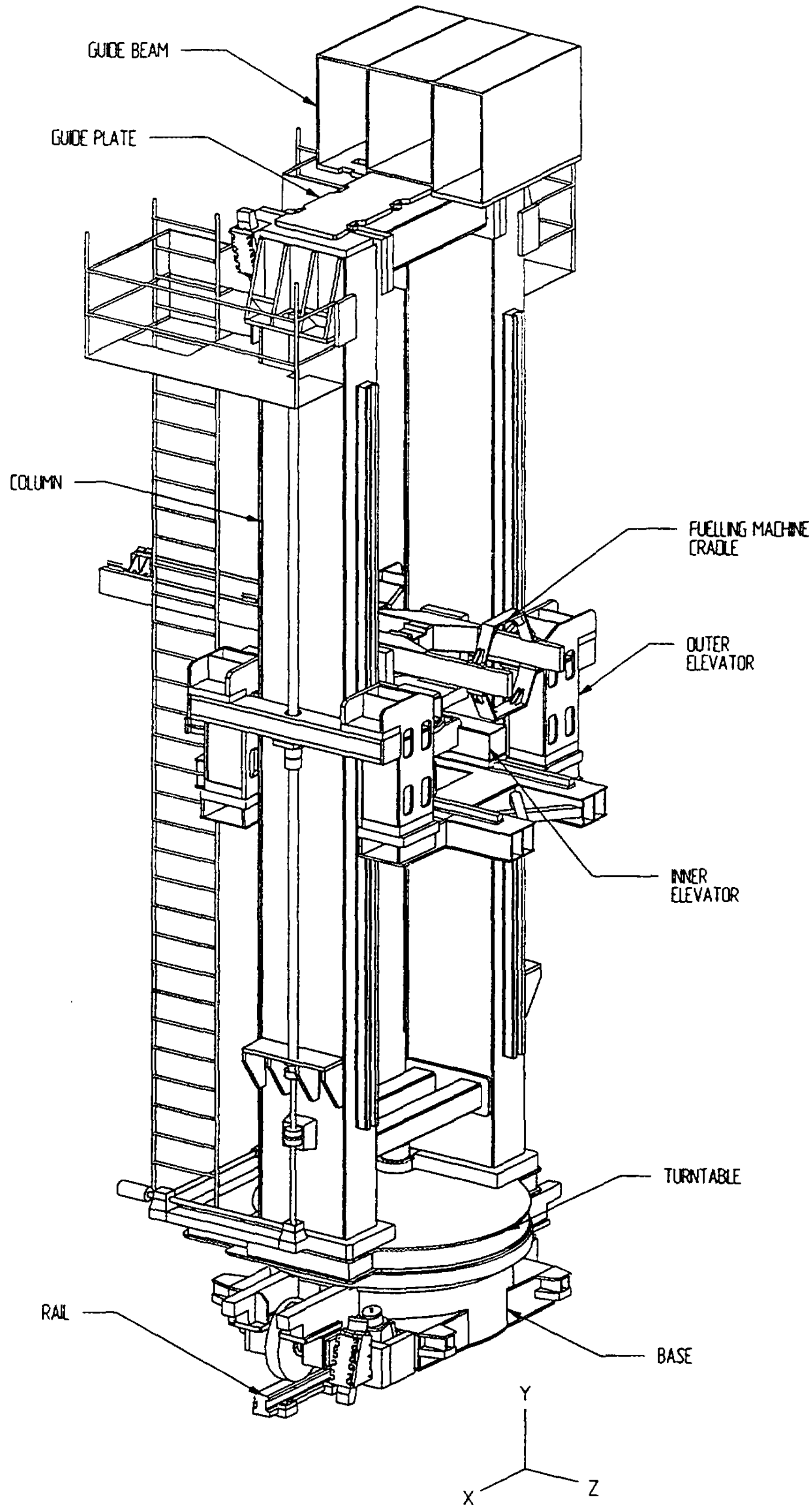


FIGURE 1 CARRIAGE GENERAL ARRANGEMENT

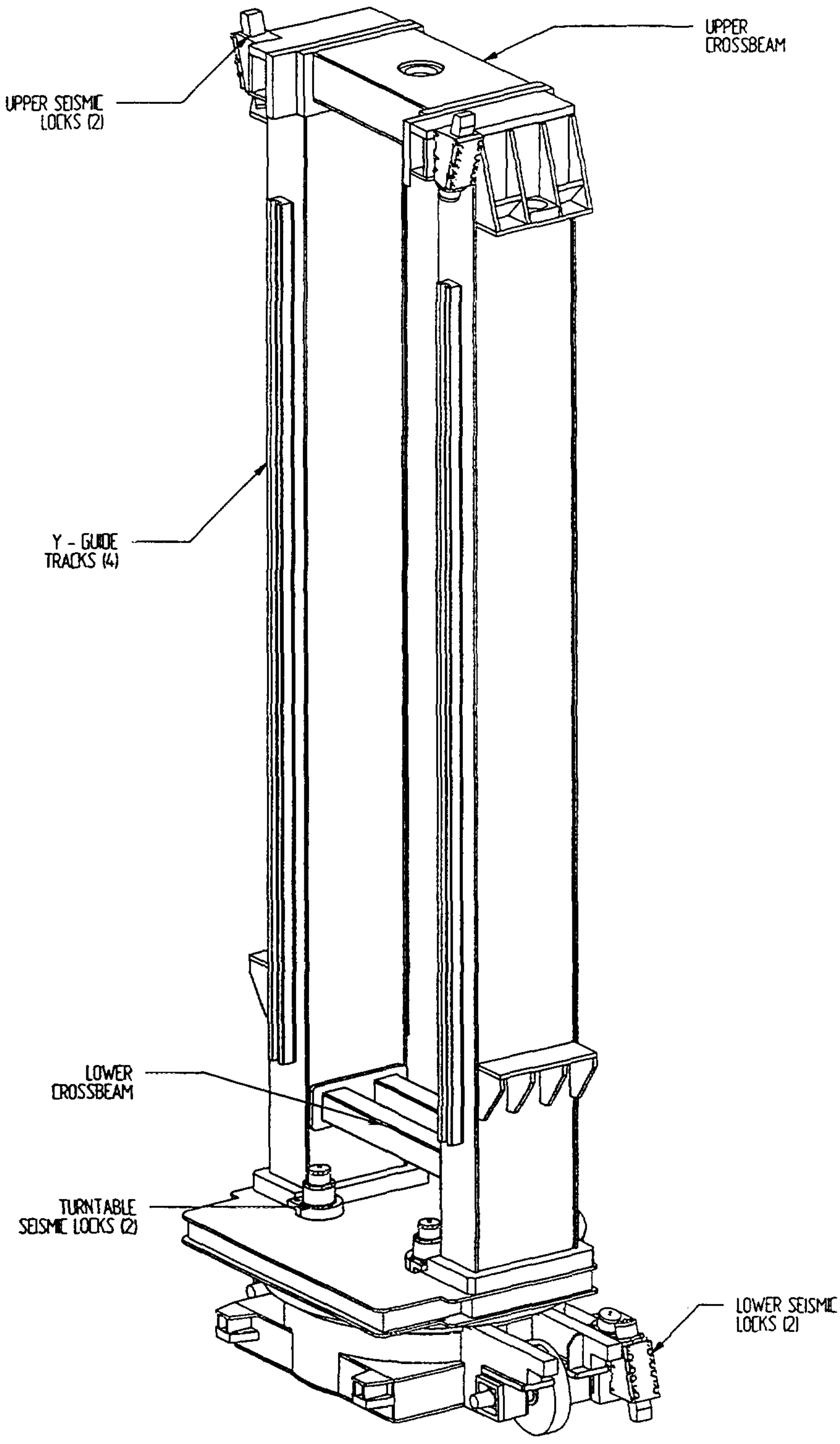


FIGURE 2 CARRIAGE COMPONENTS

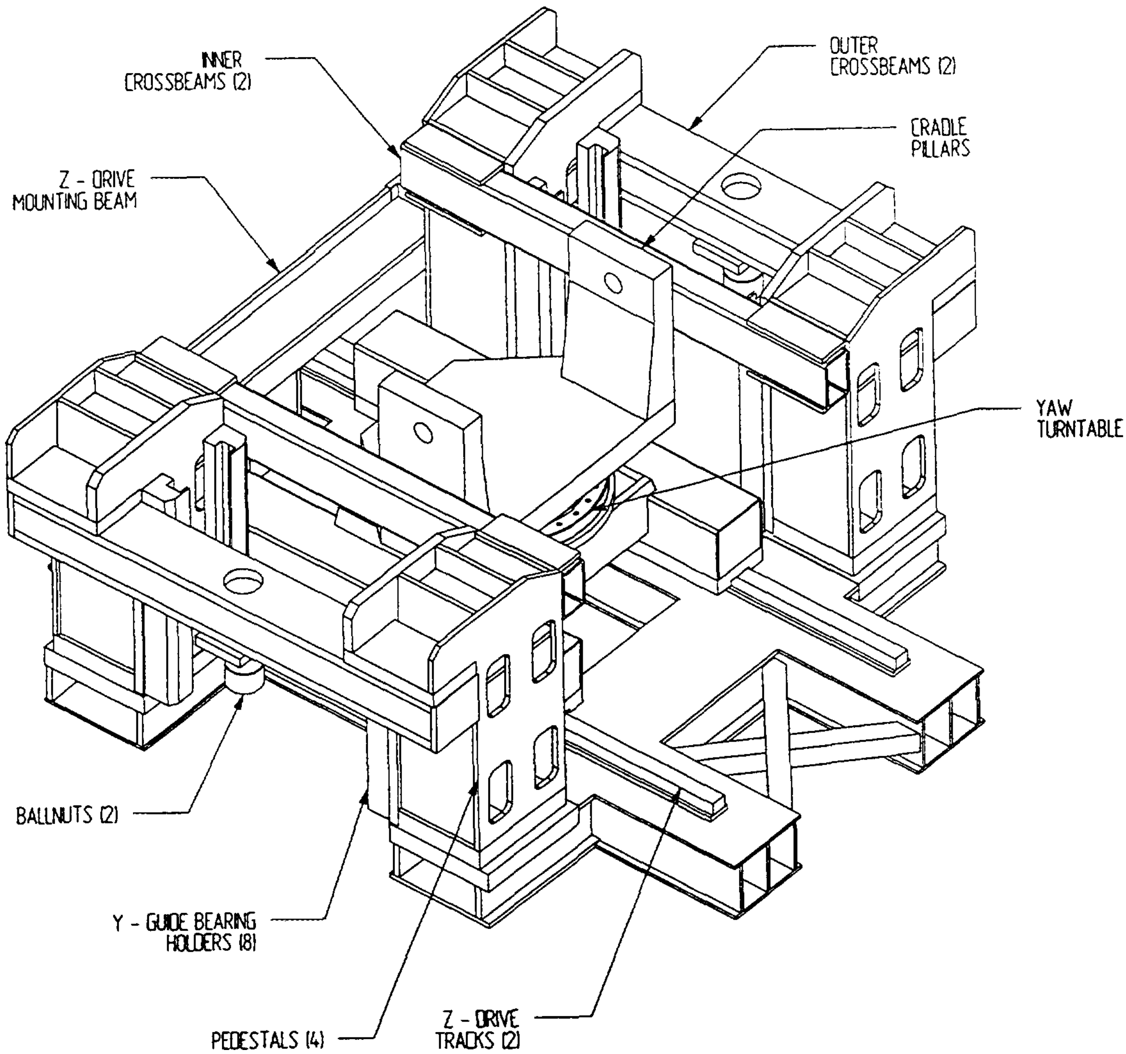
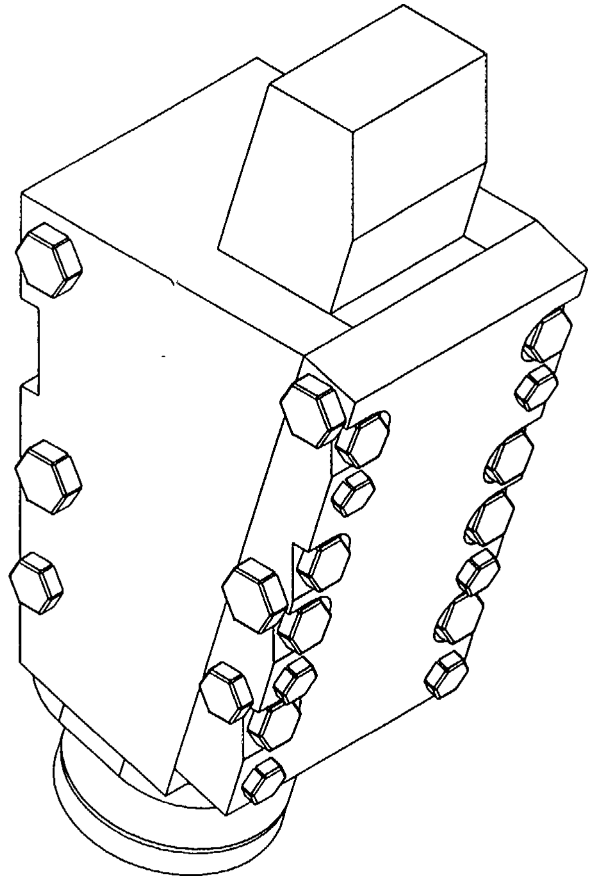
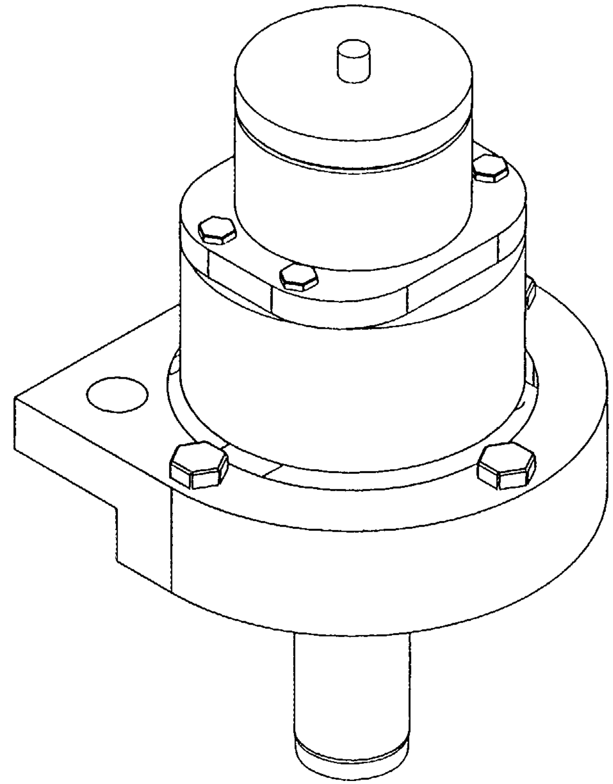


FIGURE 3 ELEVATOR COMPONENTS



WEDGE - STYLE
SEISMIC LOCK



PIN - STYLE
SEISMIC LOCK

FIGURE 4 SEISMIC LOCKING MECHANISMS

THE CANDU 9 FUEL TRANSFER SYSTEM

CA9700731

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ABSTRACT

The CANDU 9 fuel transfer system is based on the CANDU 6 and the Ontario Hydro Darlington NGD designs, modified to suit the CANDU 9 requirements. The CANDU 9 new fuel transfer system is very similar to the CANDU 6, with modifications to allow new fuel loading from outside containment, similar to Darlington. The CANDU 9 irradiated fuel transfer system is based on the Darlington irradiated fuel transfer system, with modifications to meet the more stringent containment requirements, improve performance, and match station layout.

1. INTRODUCTION

The CANDU system is based on on-power fuelling. An important aspect of the system is 'fuel transfer' which includes all the equipment and operations required to load new fuel into a fuelling machine for delivery to the reactor and later to transfer the irradiated fuel between a fuelling machine and the storage bay.

This paper describes an advanced fuel transfer system being designed for the CANDU 9.

CANDU 9 is the latest version of the Pressurized Heavy Water Reactor (PHWR) system developed in Canada by AECL. It has a net electrical output in the range of 870 MW.

The seemingly simple fuel transfer system must satisfy many stringent requirements. The basic ground rules for this equipment are safety and reliability. The equipment must be designed and built, for instance, in such a way that at no time will the operator be confronted with a situation which leaves him without any means for solving a break-down problem involving a stuck, uncooled fuel bundle in the irradiated fuel port.

The paper provides a brief description and illustration of the new and irradiated fuel transfer systems at both the CANDU 6 and Darlington stations. The paper continues with a more detailed description of the proposed CANDU 9 fuel transfer system which is being designed by a joint team of AECL and GE Canada designers.

2. DESCRIPTION OF NEW FUEL TRANSFER SYSTEMS

2.1 CANDU 6

In the CANDU 6 stations, new fuel is stored outside the reactor building and delivered into the reactor building when needed. Once in the reactor building, new fuel is loaded manually into the new fuel transfer mechanism. It is then transferred into the fuelling machine head under remote control from the main control room. (See Figure 1).

Two new fuel transfer mechanisms are provided; one to supply each fuelling machine head.

The new fuel transfer mechanisms are located in the reactor building, within containment, between the two fuelling machine locks.

The new fuel transfer mechanism consists of a loading trough and ram, a magazine and indexing drive, a single-stage transfer ram, and a port/magazine transition piece.

The magazine assembly consists of a housing and rotor, with capacity for 12 fuel bundles and the new fuel port shield plug. The magazine is indexed by an electric motor and a mechanical indexing unit.

An airlock gate valve is installed between the new fuel magazine and the loading trough to seal off the magazine whenever fuel is not being loaded into the magazine. This valve reduces the spread of contamination from the fuelling machine maintenance lock into the new fuel room. Contamination is also reduced through a duct connected to the vapour recovery system which maintains a slight negative pressure in the new fuel magazine.

The loading ram is an oil/air driven ram. The transfer ram is driven by an electric motor and chain drive.

The port/magazine adapter connects the magazine and the new fuel port. This assembly locks the new fuel port shield plug in place and releases the shield plug from the transfer ram when needed.

The new fuel ports are mounted in embedments in the maintenance lock walls. A shield plug is normally latched in the new fuel port to provide shielding for the new fuel room.

2.2 DARLINGTON

There are four new fuel transfer mechanisms in each of the two loading areas. These mechanisms are located directly above the irradiated fuel reception bay in the fuelling facilities auxiliary areas. (See Figure 2).

The new fuel port penetrates the containment wall between the new fuel loading area and the fuelling machine room. The containment integrity is maintained by two gate valves between the new fuel port and the magazine assembly. A third gate valve is located between the loading trough and the magazine and is interlocked with the other gate valves to prevent simultaneous opening. The magazine is maintained at a slightly negative pressure through an active vent to prevent any flow of tritium into the new fuel loading area.

The fuel bundles are transferred from the loading trough into the magazine by a ram, the force of which is restricted to limit the force applied to the bundles.

The fuel bundles are transferred from the magazine into a transporter located in the new fuel port and then into the fuelling machine head by a two-stage ram. The transporter is moved from the new fuel port into the head by one ram while the second ram transfers the bundles from the transporter into the fuel carriers in the fuelling machine head magazine.

2.3 CANDU 9

The CANDU 9 new fuel transfer mechanism is based on both the CANDU 6 and Darlington designs. The design permits new fuel to be loaded into the fuelling machine head from outside containment.

The CANDU 9 design uses the standard 37-element fuel bundle, but will also be able to accommodate the 43-element CANFLEX fuel bundles in the future. The fuel handling sequence for CANDU 9 is illustrated in Figure 4.

2.3.1 Description

Two new fuel transfer mechanisms are provided; one to supply each fuelling machine head.

The mechanisms, shown in Figure 5, are located outside containment in the reactor auxiliary building on either side of the airlock. New fuel is therefore loaded into the new fuel transfer mechanisms from outside the reactor building.

The new fuel transfer mechanism consists of a loading trough and ram, a magazine and indexing drive, a single-stage transfer ram, a port and two port containment valves.

The new fuel magazine, similar to the CANDU 6 magazine, consists of a completely enclosed steel housing with a rotor. As in CANDU 6, there are six fuel stations and a station for the port shield plug. The magazine rotor is driven from an electric motor through a pinion and bevel gear. Indexing is controlled by a mechanical cam and roller drive indexing unit.

The fuel loading ram consists of a stainless steel rack with a ram head assembly. The ram is driven by a pinion and brushless servo motor.

An airlock valve, installed between the new fuel magazine and the loading trough, seals off the magazine whenever fuel is not being loaded into the magazine. This valve reduces the spread of tritium through the new fuel magazine into the new fuel room and enables the transfer mechanism and fuelling machine head to be pressurized during fuel transfer to the fuelling machine head.

The transfer ram is similar to the new fuel loading ram and consists of a ram tube with a latching ram head and a stainless steel rack. The entire ram is contained in a housing with a seal at the magazine housing. The ram is driven by a pinion with a sealed shaft.

The new fuel port is similar to the CANDU 6 component. This assembly locks the new fuel port shield plug in place and releases the transfer ram from the shield plug when needed.

The shield plug is normally latched in the new fuel port and performs the same function as on CANDU 6.

The new fuel port is mounted in an embedment in the reactor building wall between the new fuel transfer room and the fuelling machine lock. Because the new fuel transfer mechanism is fixed to the reactor auxiliary building, and the ports to the reactor building, the transfer mechanism is designed to accommodate differential settlement and building shifts due to seismic events.

The port containment valves are located between the port and the magazine housing. These valves are normally closed to maintain the containment boundary.

2.3.2 Operation

New fuel pallets are delivered from the new fuel storage room to the airlock level, and from there to the new fuel transfer room.

The new fuel bundles are inspected before loading into the transfer mechanism. The fuel bundles are lifted from the pallet using a fuel bundle lifting tool, suspended from an air balance hoist and a monorail. The bundles are placed on an inspection table and checked for interlocked element spacers, using a gauge, and vacuumed/air blown if needed.

Two bundles are loaded into the loading trough and the trough lid is closed. The airlock valve is opened and the bundles are pushed into the magazine by the new fuel loading ram. The ram is retracted and the magazine is indexed to the next channel position and two more bundles are loaded. This is repeated until the magazine contains the required number of bundles.

During new fuel loading, the magazine is opened to the ventilation system to maintain air flow from the new fuel loading room into the magazine.

To transfer fuel from the new fuel transfer mechanism into the fuelling machine, the fuelling machine clamps onto the new fuel port in the fuelling machine lock, within containment in the reactor building. The water level in the fuelling machine is lowered and the fuelling machine prepares the port for fuel transfer by installing the guide sleeve.

The new fuel transfer mechanism and the fuelling machine head are pressurized and the two port containment valves are opened. During the remainder of the transfer sequence, the pressure in the transfer mechanism and port is controlled by the new fuel transfer mechanism auxiliaries.

Until the containment valves are opened, the port and containment valves are the containment boundary and rest of the new fuel transfer mechanism is outside containment. After the containment valves are opened, the fuelling machine head becomes the containment boundary. The pressure in the transfer mechanism and head allow containment integrity to be confirmed and monitored.

To meet the regulatory requirements, the new fuel port and containment valves are Class 2, and the remainder of the transfer mechanism is Class 6.

Fuel bundles are transferred from the transfer mechanism into the fuelling machine head by the transfer ram. The transfer ram first removes the port shield plug and stores it in the new fuel magazine. The ram then pushes the fuel bundles in pairs from the new fuel magazine into the fuelling machine head. This operation is controlled remotely from the fuel handling console in the main control room.

3. DESCRIPTION OF IRRADIATED FUEL TRANSFER SYSTEMS

A paper presented at an earlier conference (1) defined the major design requirements for irradiated fuel transfer systems.

3.1 CANDU 6

The CANDU 6 irradiated fuel handling system consists of discharge and transfer equipment in the reactor building, and irradiated fuel reception and storage bays and equipment in the service building. (See Figure 1).

Irradiated fuel is transferred from the fuelling machine head to the discharge bay in air. The fuelling machine clamps onto the irradiated fuel port, prepares the port for transfer, and lowers the water level in the head. The two ball valves on the irradiated fuel port are opened and bundle pairs are pushed by the fuelling machine ram through the port and onto an elevator.

The elevator lowers the bundles into the discharge bay onto a conveyor cart, then returns to the port to receive the next bundle pair. This is repeated until all fuel has been discharged from the fuelling machine. There is one discharge bay conveyor serving both fuelling machines. The discharge conveyor transfers the cart to the reception bay conveyor.

During irradiated fuel transfer from the fuelling machine head, the containment boundary is at a containment gate at the end of the discharge bay canal. Earlier CANDU 6 units were not equipped with a containment gate and rely on the fuelling machine auxiliaries and the head of water in the discharge canal for the containment boundary. At all other times, the port valves are closed.

After fuel reaches the reception bay, the fuel rack is removed from the conveyor cart and placed in the reception bay. Fuel bundles are transferred to the storage bay trays using manual tools. Once the trays are full (24 bundles/tray), they are transferred to the storage bay and placed on storage tray supports.

The storage bay trays are stacked 19 high and sets of four stacks are secured with a covering frame. This assembly is sealed with an IAEA device.

For interim dry storage, bundles are transferred from the trays to cylindrical baskets. The baskets are then placed in dry storage concrete canisters.

3.2 DARLINGTON

The major differences between the CANDU 6 and Darlington systems are as follows:

The CANDU 6 irradiated fuel transfer system is 'above water' meaning that the irradiated fuel port is located above the water level in the bay. Darlington uses an 'underwater' concept wherein the port is located below the bay water elevation. An air-filled transfer chamber ('air chamber') is used. The bundles are transferred from D_2O in the fuelling machine head, to the air chamber, and finally into H_2O in the bay. In a CANDU 6 storage bay, irradiated fuel is stored in single-layer trays. In Darlington, it is stored in multi-layer modules. This introduces an additional indexing operation.

A description of the Darlington system follows. Figure 3 illustrates the system features.

The irradiated fuel port penetrates the containment wall 4.4 m below the surface of the bay water. An underwater air chamber is attached to the port from the bay side. Flooding the air chamber with bay water provides emergency cooling in case a bundle becomes stuck in the port. Defected fuel in the underwater chamber can be identified by drawing air from the chamber through a detector ('dry sniffing').

Irradiated fuel is discharged from the fuelling machine onto a shuttle in the irradiated fuel port. The shuttle is retracted through the port into the air chamber where it is supported on a ladle. The ladle and shuttle are lowered to transfer the shuttle onto a vertically travelling elevator. The elevator is lowered to align the shuttle with a chosen tube of the module, and the fuel is pushed out of the shuttle into the module. Horizontal indexing of the module combined with vertical indexing of the elevator make it possible to fill all tubes in the module.

Accurate indexing of the module conveyor is obtained by a Geneva mechanism. A telescopic water hydraulic cylinder provides smooth motion of the shuttle through the port. Shuttle position detection is by ferromagnetic sensors. Underwater equipment is driven, as far as possible, from above water by motorized drives. All motorized equipment has the capability of being driven manually.

The initial concept for handling defected fuel at Darlington was to separate defected fuel from sound fuel and, after inspection in the reception bay, place it in cans. Later experience indicated that the isolation of defected fuel in cans and containers was not necessary. A decision was made to place defected fuel directly into modules, with provisions for later removal of defected fuel from the modules for inspection.

3.3 CANDU 9

The fuel handling sequence for CANDU 9 is illustrated in Figure 4. The system minimizes the fuelling machine refuelling cycle time. This is achieved by temporarily storing the irradiated fuel in the port. Loading of the fuel into the storage bay takes place later, after the fuelling machine departs from the irradiated fuel port.

3.3.1 <u>Description</u>

The irradiated fuel transfer system, shown in Figure 6, is located primarily in the reactor auxiliary building. It consists of:

a) An irradiated fuel port mounted in the reactor building containment wall.
b) Two port containment valves.
c) A flexible port/magazine transition piece.
d) A totally enclosed irradiated fuel magazine and its drive assembly which is located in the storage bay

wall within the reactor auxiliary building.

e) An irradiated fuel transfer ram and drive.
f) A bay valve.
g) Associated air and water auxiliary systems.
h) Storage bay equipment which includes the indexing tray, storage ram, bundle tilter, 'basket modules', stacking frames and handling tools.

A paper published earlier (2) describes the 'basket module' and future considerations for the dry storage of CANDU fuel.

The irradiated fuel port assembly conforms to Class 2, Class 4 and Class 6 as appropriate. It has the flexibility and strength requirements to permit the homing and locking of the fuelling machine and to withstand high seismic loads. The port is sloped down towards the port magazine. D_2O carried from the head on the fuel bundles is collected to prevent its loss to the bay.

The irradiated fuel transfer mechanism spans two separate buildings; the reactor building and the reactor auxiliary building. These two buildings can move independently of one another under settlement and seismic actions.

The fuelling machine unloads irradiated fuel through its associated port which leads to one of the irradiated fuel storage bays located outside the reactor building containment wall. Each face of the reactor has one storage bay dedicated to it. The two storage bays are not interconnected.

The port/magazine transition piece provides the required flexibility between the buildings to allow these movements to be accommodated without loss of containment. This is achieved using two specially designed ball joints to accommodate the vertical, horizontal and axial displacements. The transition piece has a D_2O recovery circuit to recover D_2O vapour that evaporates or boils off the fuel bundle surface during fuel transfer in air. This D_2O recovery loop also contains provisions for 'dry sniffing' to detect defected fuel as it passes through the transition piece.

The irradiated fuel magazine (or port magazine) is located in a sleeve in the storage bay wall within the reactor auxiliary building. The end plate of the magazine housing is removable to enable maintenance to be carried out on the rotor assembly. Mounted on the end plate are the transition piece, the magazine drive assembly, the irradiated fuel transfer ram, and cooling water connections. The heat generated by the irradiated fuel bundles within the magazine is removed through a heat exchanger fitted internally in the magazine housing.

The magazine rotor consists of a stationary shaft, two end plates and six magazine channels. There are seven channel positions with one position used during maintenance only. This position is fitted with a hinged blanking plate to allow space for special tools (e.g. cleaning tools). The capacity of the magazine rotor is therefore six pairs of fuel bundles.

The CANDU 9 irradiated fuel transfer system is being designed to accommodate emergency recovery for all foreseeable failure scenarios in a manner that minimizes radiation and conventional risk to station staff, the public and the environment.

3.3.2 <u>Operation</u>

To prepare for fuel transfer, a pump is started to establish water circulation flow through the port magazine. The water flows over a weir for level control in the magazine. The irradiated fuel magazine and transition piece are pressurized with air. The fuelling machine homes and locks onto the irradiated fuel port and is pressurized. The two port containment valves are then opened and the port air pressure system regulates the pressure in both the port and the head.

The fuelling machine magazine channel, containing the fuel bundles to be transferred, is aligned with the port. The rams of the fuelling machine are used to push the irradiated fuel from the fuelling machine magazine into the port magazine, two bundles at a time. The rams are then retracted into the fuelling machine and both magazines index to the next position. This procedure is repeated until all the fuel has been transferred.

The irradiated fuel transfer takes place in air from the time that the fuelling machine magazine is indexed to align the fuel for transfer until the port magazine is indexed from the load position and the fuel is submerged below the water level in the port magazine. Fuel is then cooled in the circulating water in the magazine.

The control systems associated with the design ensure that the proper sequencing of the operation is achieved. Once the fuel transfer is complete, the two port containment valves are closed. The port is then flooded. In an emergency, the port and magazine can be flooded from a dump tank by gravity.

Irradiated fuel is now transferred from the port magazine to the storage bay by opening the bay valve. This operation equalizes the pressure in the port with the bay pressure. The bundles are pushed by the transfer ram, two at a time, from the magazine onto an indexing tray as it is moved automatically across the face of the irradiated fuel port. The tray has six positions for fuel bundles. Upon completion of transfer, the bay valve is closed.

A storage ram is then used to push the bundles, one at a time, into a bundle tilter designed to rotate the bundle into the vertical position. The tilter restrains the bundle during this operation and is fitted with a shock absorber to reduce impact on the bundle during tilting. A tool supported from the manbridge is used to transfer the irradiated fuel bundles from the tilter to the 'basket module'.

4. CONCLUSION

Major features of the CANDU 9 irradiated fuel transfer system design include commonality of equipment between the new fuel and irradiated fuel mechanisms and reuse or adaptation of technology used and proven at other CANDU plants. The transfer system design provides adaptability to a wide range of site conditions, low capital cost, short construction schedule, minimum system complexity, high system availability, low operating and maintenance cost, and a generic approach where many of the features can be applied to future CANDU designs.

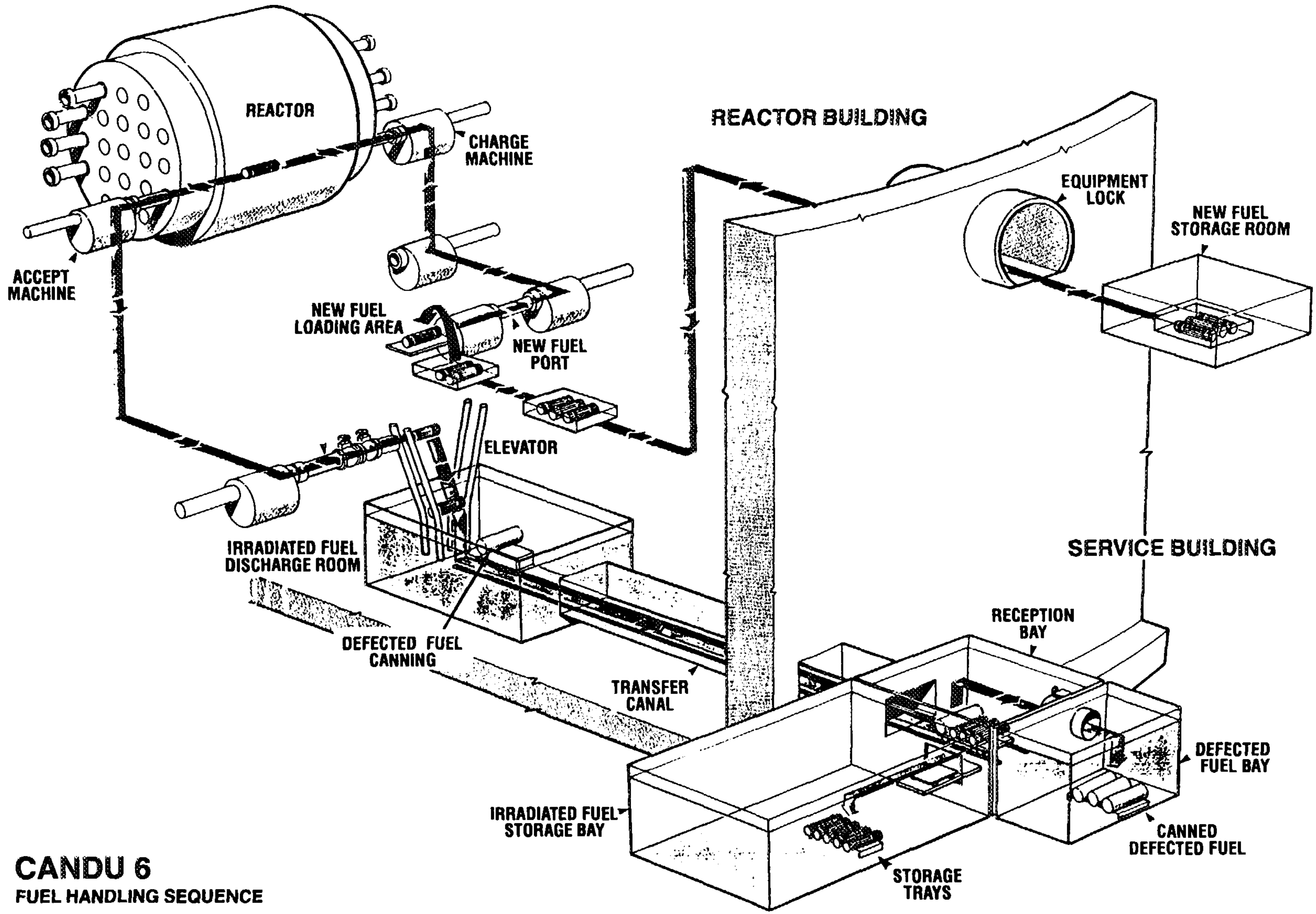
ACKNOWLEDGMENTS

The authors would like to thank Ontario Hydro, AECL and GE Canada for the use of their data.

REFERENCES

(1) KESZTHELYI Z.G., DIDOMIZIO F., GAERTNER B.R., "The Evolution of CANDU Irradiated Fuel Transfer Systems", A Paper Presented at the Third International Conference on CANDU Fuel, Chalk River, Ontario, Canada, 1992 October 4-8.

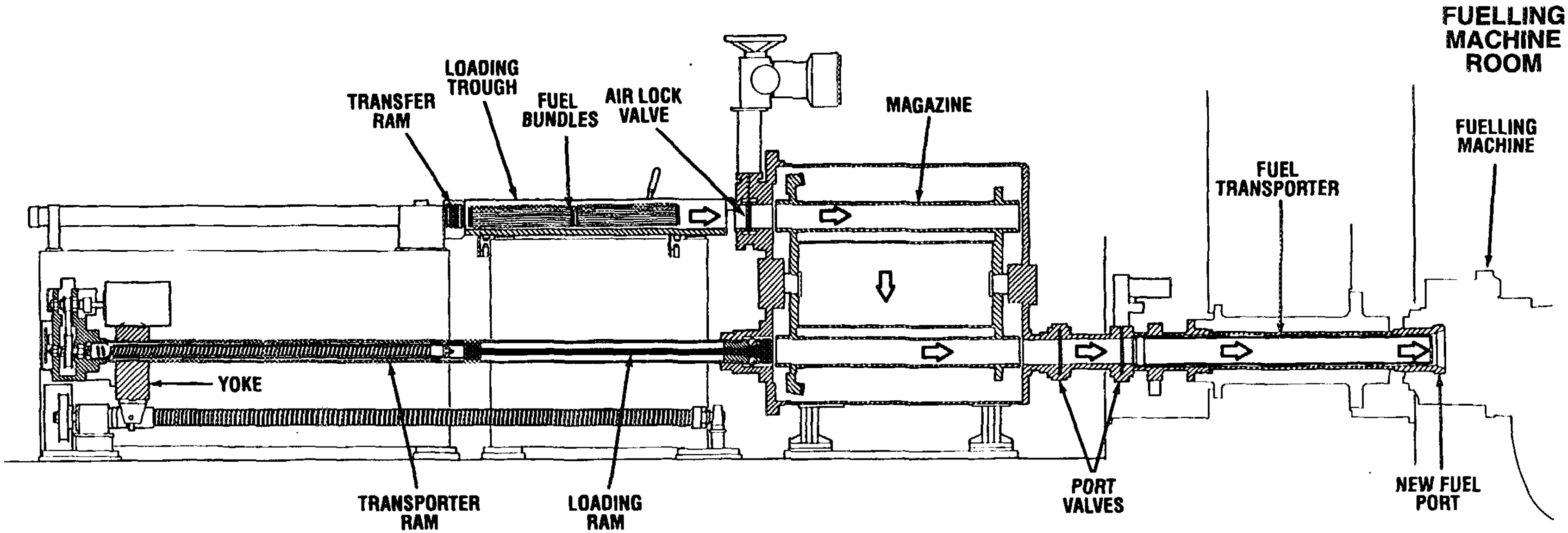
(2) NAKAGAWA R.K., "Standardized Approach to the Design of CANDU Used Fuel Receptacles", A Paper Presented at the Third International Conference on CANDU Fuel, Chalk River, Ontario, Canada, 1992 October 4-8.



CANDU 6
FUEL HANDLING SEQUENCE

CG

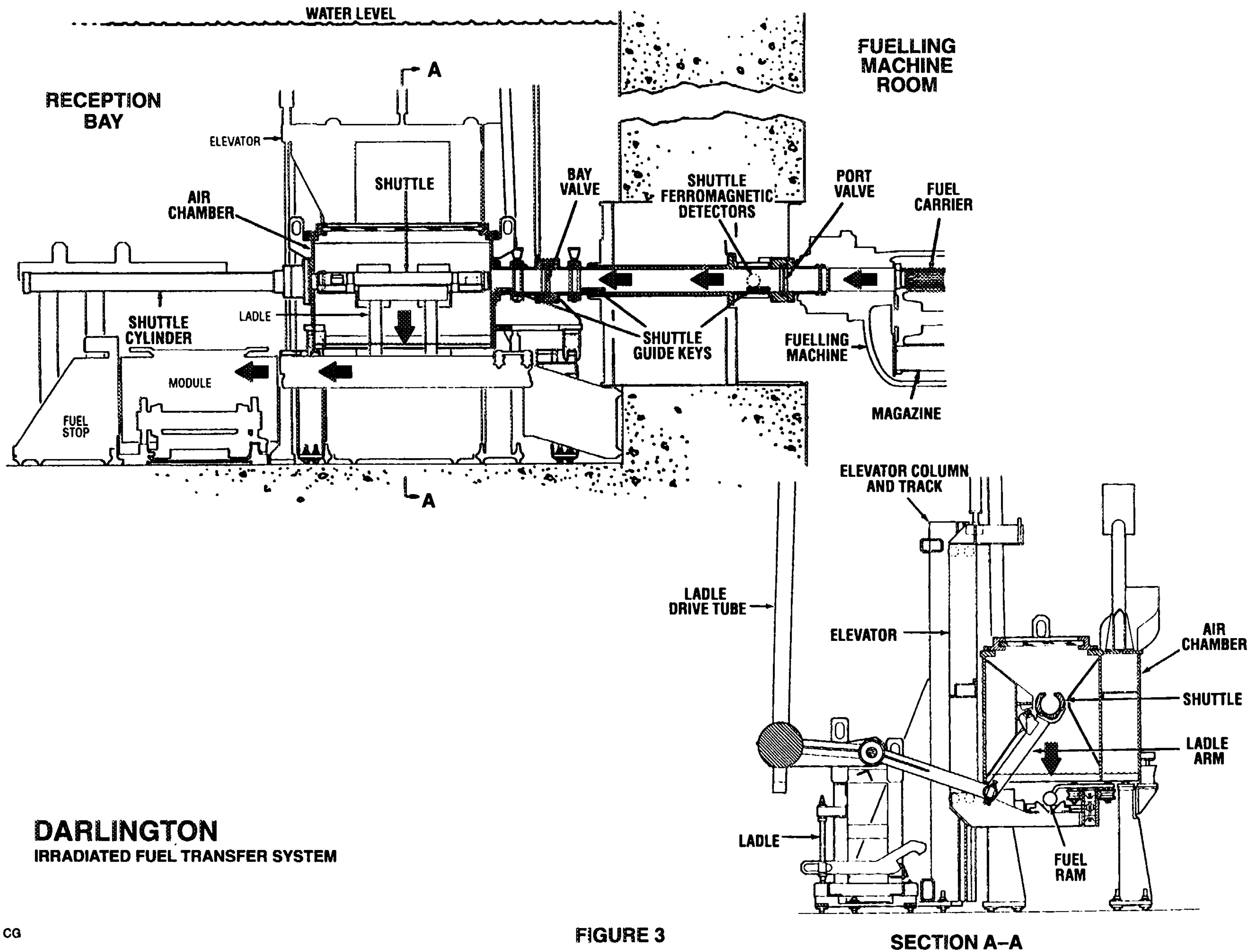
FIGURE 1



DARLINGTON
NEW FUEL TRANSFER SYSTEM

CG

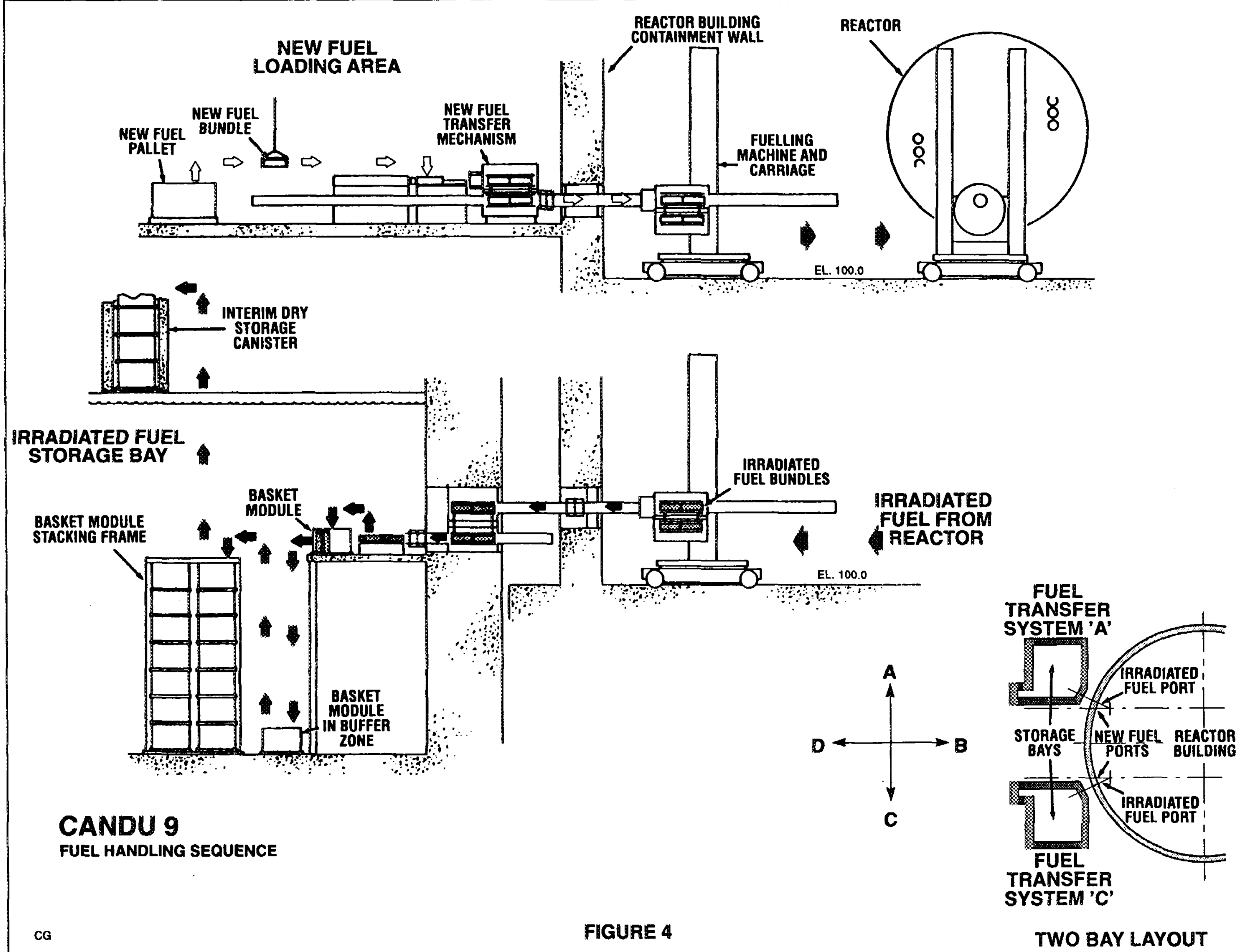
FIGURE 2



DARLINGTON
IRRADIATED FUEL TRANSFER SYSTEM

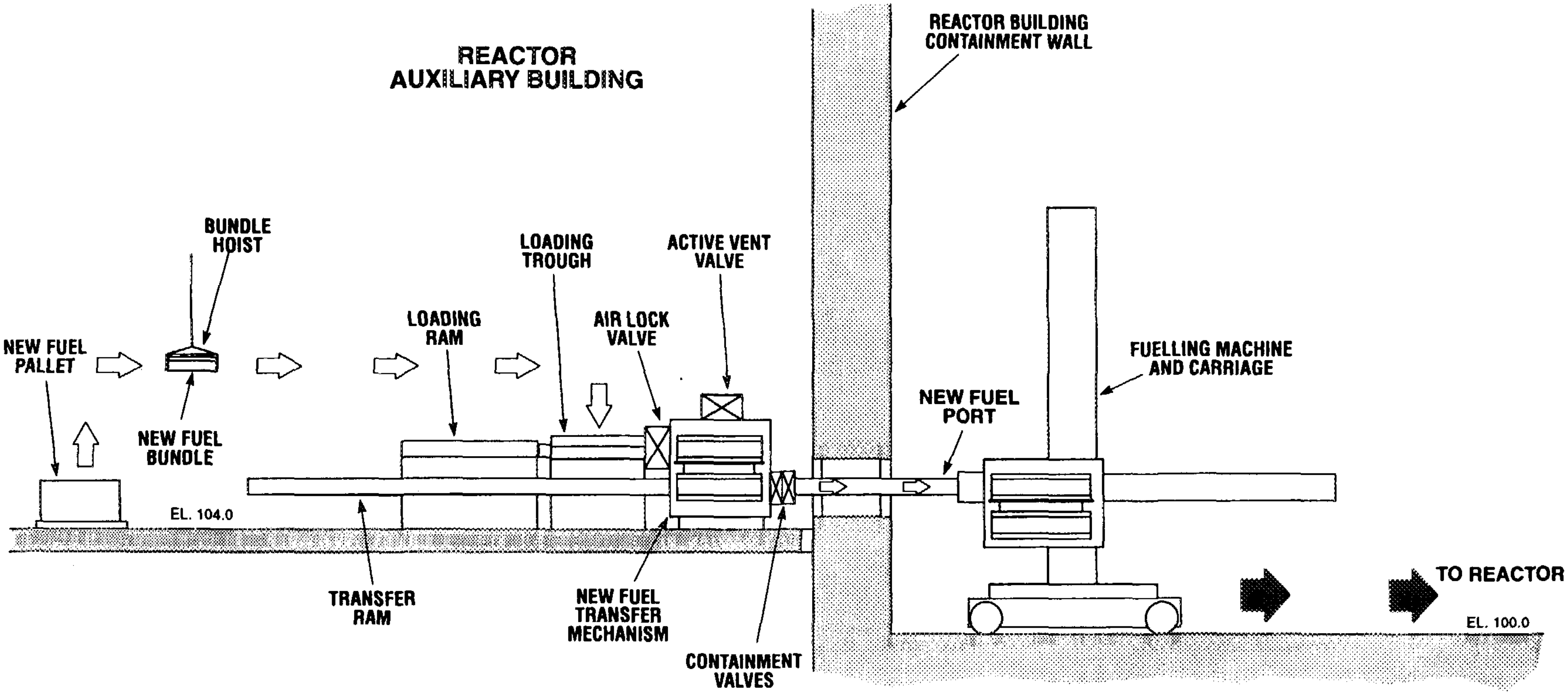
CG

FIGURE 3


NEW FUEL
LOADING AREA
REACTOR BUILDING
CONTAINMENT WALL
REACTOR
NEW FUEL
BUNDLE
NEW FUEL
PALLET
NEW FUEL
TRANSFER
MECHANISM
FUELLING
MACHINE AND
CARRIAGE
EL. 100.0
INTERIM DRY
STORAGE
CANISTER
IRRADIATED FUEL
STORAGE BAY
IRRADIATED
FUEL BUNDLES
IRRADIATED
FUEL FROM
REACTOR
BASKET
MODULE
BASKET MODULE
STACKING FRAME
EL. 100.0
BASKET
MODULE
IN BUFFER
ZONE
FUEL
TRANSFER
SYSTEM 'A'
IRRADIATED
FUEL PORT
A
D
B
C
STORAGE
BAYS
NEW FUEL
PORTS
REACTOR
BUILDING
IRRADIATED
FUEL PORT
FUEL
TRANSFER
SYSTEM 'C'
CANDU 9
FUEL HANDLING SEQUENCE
TWO BAY LAYOUT

CG

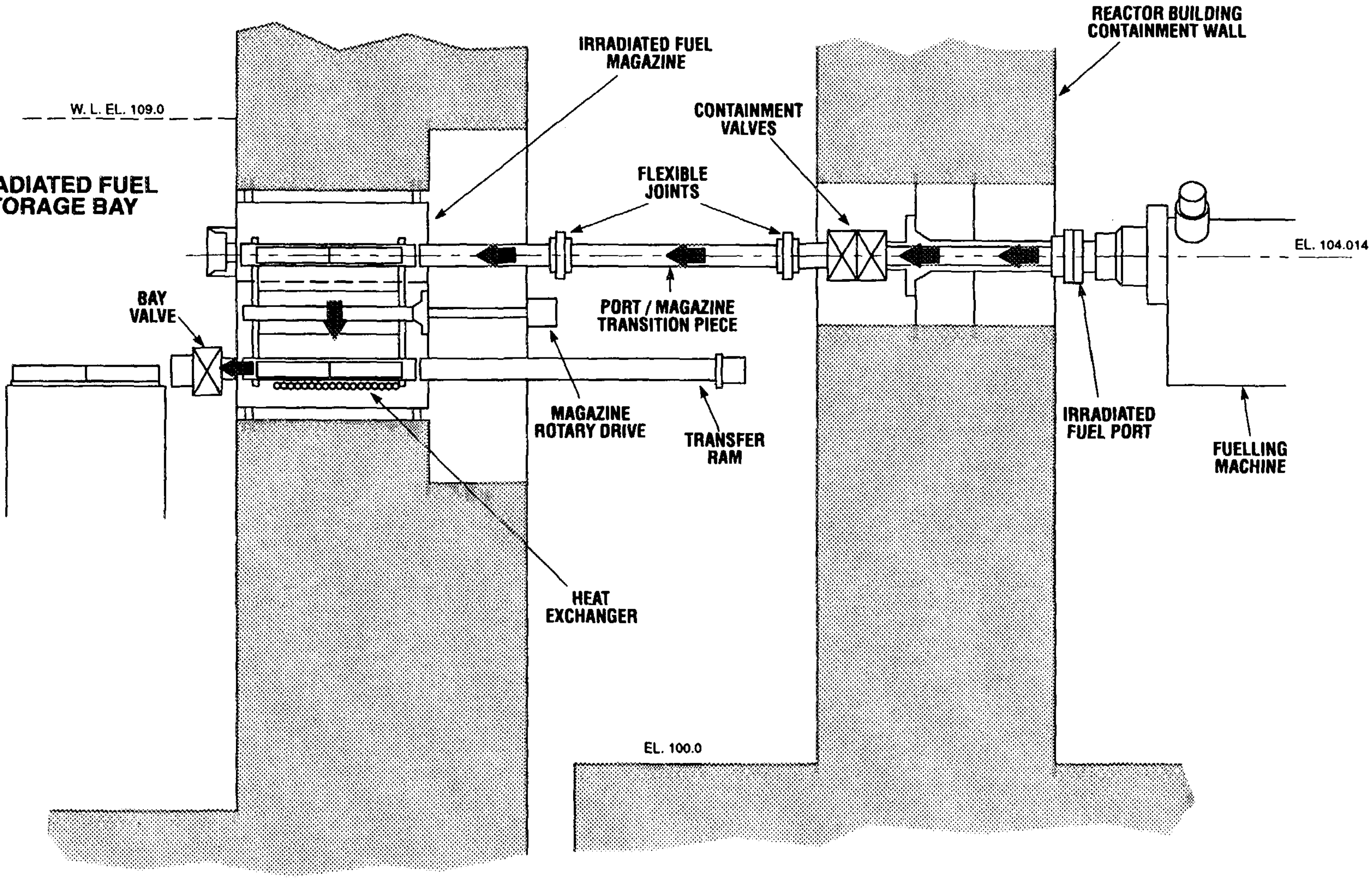
FIGURE 4



CANDU 9
NEW FUEL TRANSFER SYSTEM

CG

FIGURE 5



CANDU 9
IRRADIATED FUEL TRANSFER SYSTEM

CG

FIGURE 6

OVERVIEW OF THE CANDU FUEL HANDLING SYSTEM FOR ADVANCED FUEL CYCLES

by

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ABSTRACT

There has been much interest around the world in the development of advanced fuel cycles for the purposes of resource utilisation and the disposition of reactor wastes or weapons material. The CANDU[1] system is an ideal system for implementing advanced fuel cycles because it can be adapted to burn these alternative fuels without major changes to the reactor.

The fuel handling system is adaptable to implement advanced fuel cycles with some minor changes. Each individual advanced fuel cycle imposes some new set of special requirements on the fuel handling system that is different from the requirements usually encountered in handling the traditional natural uranium fuel. These changes are minor from an overall plant point of view but will require some interesting design and operating changes to the fuel handling system.

Preliminary conceptual design work has been done on the fuel handling system in support of some of these fuel cycles. Some fuel handling details were studied in depth for some of the advanced fuel cycles. This paper provides an overview of the concepts and design challenges.

[1] CANDU® - Canada Deuterium Uranium Registered trademark of AECL

DEMONSTRATING THE COMPATIBILITY OF CANFLEX FUEL BUNDLES WITH A CANDU 6 FUELLING MACHINE

by

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ABSTRACT

CANFLEX[1] is a new 43-element fuel bundle, designed for high operating margins. It has many small-diameter elements in its two outer rings, and large-diameter elements in its centre rings. By this means, the linear heat ratings are lower than those of standard 37-element bundles for similar power outputs. A necessary part of the out-reactor qualification program for the CANFLEX fuel bundle design, is a demonstration of the bundle's compatibility with the mechanical components in a CANDU[2] 6 Fuelling Machine (FM) under typical conditions of pressure, flow and temperature.

The diameter of the CANFLEX bundle is the same as that of a 37-element bundle, but the smaller-diameter elements in the outer ring result in a slightly larger end-plate diameter. Therefore, to minimize any risk of unanticipated damage to the CANDU 6 FM sidestops, a series of measurements and static laboratory tests were undertaken prior to the fuelling machine tests.

The tests and measurements showed that; a) the CANFLEX bundle end plate is compatible with the FM sidestops, b) all the dimensions of the CANFLEX fuel bundle are within the specified limits.

INTRODUCTION

A demonstration of the CANFLEX fuel bundle in an actual CANDU 6 FM is necessary, as a final step, to qualify the bundle for use in CANDU 6 reactors. To minimize the risk of unexpected damage to the FM, the dimensional compatibility of CANFLEX fuel bundles with the components of the CANDU 6 FM head was studied prior to the out-reactor testing. The study included a series of bench-type tests and measurements to check if the bundles met the CANFLEX bundle design dimensional requirements.

First, the CANFLEX fuel bundles were passed through a "bent tube gauge" to check the bundles' ability to pass through the rolled joint sections of the fuel channels.

[1] CANFLEX - CANDU FLEXible fuelling

[2] CANDU - CANada Deuterium Uranium is a registered trademark of AECL

The maximum diameters of the CANFLEX bundles were measured. These measurements were compared with the maximum diameter specified for the CANFLEX bundle. The diametral measurements were repeated while the bundles were squeezed inwards at the mid-plane.

Other dimensions of the CANFLEX bundles were measured using a co-ordinate measurement machine (CMM) to ensure that all the bundles' dimensions were within the specified limits.

As part of this investigation, measurements were taken in a special CANDU 6 "sidestop fixture" to determine the clearances between the end plate of the CANFLEX bundle and the sidestops. The sidestop fixture is a simple test fixture designed and fabricated with two sidestop pieces that can be positioned so as to simulate different 'penetrations' of the CANDU 6 FM sidestops.

The sidestops are part of the separators assemblies to prevent the column of fuel from moving into the downstream FM head under the influence of the hydraulic drag force created by the coolant flow.

The CANFLEX bundle end elevation measurements were also conducted in the CANDU 6 sidestop fixture.

Finally, in this study, grappling tests of the CANFLEX fuel bundles were carried out manually on a bench to ensure that in an emergency situation, the new design of fuel bundle can be withdrawn from a fuel channel using a standard CANDU 6 grappling tool.

Good quality assurance practice was followed for the tests and measurements presented in this paper, to ensure that they met the quality assurance program for design verification of nuclear components according to Canadian Standard N286.2 and complied with the practice adopted for the joint AECL/ KAERI[3] CANFLEX program.

FM SEPARATOR ASSEMBLIES

There are two separator assemblies mounted symmetrically on either side of the FM magazine end cover. Both separator units are identical in all respects, except that they are handed left and right. Each separator assembly consists of a casing that houses hydraulic pistons actuated by heavy water. The pistons actuate the sensor, the pusher and the sidestop through connecting linkages (see Figure 1).

The sensor is lightly loaded in a downward direction to ride on the circumference of the fuel bundles. This downward loading allows the sensor finger to sense a gap between two fuel bundles. When the sensor finger drops into the gap, it generates a signal to indicate that the gap is in the proper position to allow for the insertion of the sidestops. During the insertion operation the sidestop is restrained to stay in alignment with the sensor finger.

For the tests described here, the sidestop fixture, shown in cross-section in Figure 2, was fitted with two "sidestops". A gap at the centre of each sidestop piece represents the sensor finger.

BENT TUBE GAUGE MEASUREMENTS

The "bent tube gauge" is a simple tube with a very precise diameter and with an accurate bend angle, which is used by the fuel manufacturers to monitor the maximum bundle diameter and its ability to pass through the rolled joint sections of the fuel channels.

The bent tube gauge measurements were carried out to investigate if the CANFLEX bundles could pass under their own weight through the existing design of the bent tube gauge. Each bundle was placed inside the bent tube gauge horizontally. Then, one end of the bundle/tube arrangement was lifted so that the bundle passed through the tube under its own weight.

A total of eight CANFLEX bundles were passed through the bent tube gauge, and all met this acceptance criterion.

DIAMETER MEASUREMENTS

The diameters of the bundles were measured to ensure that they met the requirements for the maximum diameter specified for the CANFLEX bundle. These measurements were repeated while the bundles were under compression at the mid-plane.

[3] KAERI - Korea Atomic Energy Research Institute

Using a granite surface plate for a base, V-blocks were placed for support at either end of the bundles. The circumferences of each bundle over the mid-plane and end-planes at the sheath and at the bearing pad locations were measured. Then, the bundle was compressed at the bundle mid-plane, and similar measurements were taken.

BUNDLE DIMENSIONAL MEASUREMENTS

Dimensions of the CANFLEX fuel bundles were also measured using a Co-ordinate Measurement Machine (CMM).

The CMM is a programmable machine consisting of a bridge and a carriage assembly that may be driven in the x-y-z axes. The bridge and carriage assembly are attached to a granite table. In this system, all three axes are driven and controlled by dc friction drive motors and drive bars. The system is mounted on a series of air pads that serve as "anti vibration" stabilizers.

For the purpose of these measurements, the CMM was programmed to measure the Pitch Circle Radius (PCR), end plate profile (waviness), bearing pad heights, element bow, bundle radii, and bundle droop (i.e., end-cap-to-pressure tube clearance) of the CANFLEX fuel bundle.

SIDESTOP MEASUREMENTS

The sidestop fixture is a simple test fixture with two sidestop pieces that can be positioned so as to simulate different penetrations of the CANDU 6 FM sidestops (see Figure 2). This fixture was made to allow easy comparison between bundle types, for typical sidestop insertion distances. The sidestop measurements were carried out in two stages: clearance measurements, and bundle end measurements.

Although this fixture was designed as closely as possible to reproduce the sidestop configuration, it is recognized that there may be differences in dimensioning an actual FM because of tolerancing considerations. Nevertheless, we compared the CANFLEX measurements also with typical CANDU 6, 37-element bundle measurements, and we concluded that this is a valid method for testing for bundle compatibility in a typical FM, prior to actual hot tests in a machine.

Clearance With Sidestops

Measurements were taken to determine the clearances between the bundle end plate and the sidestops set at the maximum penetration or insertion distance into the FM liner tube. The clearances were measured for CANFLEX fuel bundles, and were compared with those of standard CANDU 6 production bundles.

To set up the sidestops with the maximum penetrations, first the sidestop fixture was set in V-blocks on a granite surface plate. Then, one of the sidestop pieces was accurately located at the so that the flat edges of the sidestop piece (see Figure 2) were perpendicular to the surface plate. The bottom corner of the sidestop piece was set flush with the tube inside diameter, the height gauge was set at zero on the sidestop centre line. Then the sidestop piece was moved down to the desired penetration. The setup was repeated for the second sidestop piece.

The clearance measurements were repeated for the standard CANDU 6 bundles for comparison. Figure 2 depicts the overall clearances between the inserted sidestops and the end plates on the CANFLEX and standard CANDU 6, 37-element bundles. The actual measured clearances are compared in Table 1.

Table 1: Typical Clearance Measurements

Sidestop Corner	Clearance (mm)			
	Bundle			
	1	2	3	4*
Top Left	0.84	0.73	0.73	1.02
Top Right	0.84	0.79	0.73	1.14
Bottom Left	0.38	0.20	0.38	0.48
Bottom Right	0.45	0.40	0.30	0.79

* CANDU 6, 37-element fuel bundle

Bundle End Measurements

Very accurate bundle end measurements were conducted on CANFLEX bundles for direct comparison with CANDU 6, 37-element bundles. These include: dimension "B" which is the perpendicular distance from the bottom of the FM liner, or sidestop tube, to the top surface of the element end cap at the 12 o'clock position, and dimension "E" which is the perpendicular distance from the bottom of the liner to the top of the end plate". These are shown in Figure 3.

To prepare for bundle end measurements, the jig was placed on two precision parallel blocks, which were set on a granite surface plate. This was done to enable an indicator on the height gauge to be used to measure the top elements and end plate. The indicator was set at zero representing the bottom of the radius surface of the sidestop fixture. Using the height set at the appropriate zero, measurements "B" and "E" were taken on both ends of the bundles. Precise measurements were obtained and recorded as the indicator reached the top of the end cap and end plate. Table 2 tabulates typical bundle end measurement.

Table 2: Typical Bundle End Measurements

Bundle	Average Bundle End Measurements (mm)	
	B	E
1	100.52	96.76
2	100.49	96.71
3	100.55	97.17
4	100.47	96.97

6. GRAPPLING TESTS

The fuel grappling tool is designed to defuel a fuel channel using a single fuelling machine when the other fuelling machine is disabled or when a channel is to be de-fuelled for routine maintenance.

When the CANDU 6 grappling tool approaches a fuel bundle, a plunger contacts the end plate. The plunger will depress, releasing the ten extracting fingers behind the end plate to latch onto the fuel-bundle end plate.

The objective of these grappling tests on the CANFLEX fuel bundles was to determine how effectively the CANDU 6 grappling tool could grapple the CANFLEX bundle. To achieve this, the CANFLEX bundle was placed horizontally inside a pressure tube of sufficient length with a cut out at its top portion for observation. Each CANFLEX bundle was grappled a number of times by the grappling tool. Two bundle orientations were tried. In the first one the lowermost element occupied the 6 o'clock position, whereas in the second the midpoint between the two lowermost elements was at that position. A minimum of 10 trials for each orientation was made. In each of these orientations the grappling tool was oriented randomly and the number of fingers grappling the endplate was recorded. Typical results from these tests are tabulated in Table 3.

Table 3: Typical Grappling Test Results

Trial no.	Number of Fingers Engaged			
	outer element at 6 o'clock		gap between outer elements at 6 o'clock	
	Bundle1	Bundle2	Bundle1	Bundle2
1	7	3	4	4
2	5	5	4	4
3	4	4	5	5
4	3	3	3	4
5	3	4	5	5
6	5	4	4	6
7	5	5	5	6
8	5	6	5	5
9	5	4	3	4
10	5	4	3	5

7. DISCUSSION

The CANFLEX bundles passed through the bent tube gauge under their own weight. This simple test showed that the CANFLEX bundle diameter is compatible with the rolled joint sections of the fuel channels.

The diameter measurements of the CANFLEX bundles showed that the maximum measured diameter of 102.28 mm was less than the maximum diameter specified for the CANFLEX bundle. Although this dimension was less than that of the standard 37-element fuel bundle, it was consistent with the smaller diameter of the CANFLEX bundle outer elements. The measurements were also consistent with the results of the bent tube-gauge measurements.

The diametrical measurements on the CANFLEX bundles were repeated while the bundles were under compression at the mid-plane. These measurements showed that the flexibility of the bundles was such that when bundles were clamped, the maximum diameter, measured at the end bearing pads, was reduced slightly. The outside diameter of the end plate at the end cap welds and between the welds was also measured.

The CANFLEX bundles' dimensions were measured before the compatibility test in CANDU 6 FM using the CMM. These measurements showed that all bundle dimensional specifications, including end plate waviness, met the relevant requirements. The measurements will be repeated after the FM test to ensure that the bundles' dimensions will stay within the specified limits.

When considering how the CANFLEX bundles interface with the CANDU 6 FM, there are several features of particular interest. During refuelling, it is desirable that the sidestops engage as fully as possible with the endcaps of the fuel elements. If the bundle is low in the liner tube, the engagement will be too small, particularly at the top of the bundle, and the sidestops at the bottom corners might touch the endplates. If the bundle is high, the sidestops at the top corners might touch the endplate. This will tend to push down on the bundle. Also, when the sidestops enter the FM liner tube, it is equally important that there be no interference across the horizontal diameter. In this case, if the end plate is too large in diameter, the sidestops may squeeze and damage the end plate.

The sidestop measurement test was designed to investigate these effects and to use the results to evaluate the adequacy of the interface between the bundles and the FM sidestops.

These sidestop "clearance" measurements with the CANFLEX bundles show that, even for the maximum sidestop penetration, there are clearances between the end plate and the bottom corners of the sidestop: a range from 0.20 mm to 0.45 mm. Although these clearances are less than those we measured for a typical standard CANDU 6 bundle i.e., minimum of 0.48 mm, the measurements showed that there will be no interference at the bottom corners of the sidestops. Thus the CANFLEX bundle's end plate will not be squeezed as the sidestops enter the liner tube.

The bundle end measurements showed that the minimum "B" and maximum "E" dimensions were within specified limits for the operational requirements of the CANDU 6 FMs. The measurements showed that there will be an adequate engagement with the fuel endcaps, especially at the top corners of the sidestops. The bundle end measurements also showed that there are approximately 20% more elements engaged for the CANFLEX fuel bundle than that for a typical 37-element bundle (see Figure 2). This will ensure that the fuel will always be held well against the flow, hydraulic drag force; and there will be a minimal chance that a bundle will be pushed by the coolant flow past the sidestop, particularly in the case of failure of a single sidestop.

The grappling tests of the CANFLEX bundles, with a CANDU 6 grappling tool head, showed that a minimum of three fingers always engaged on the end plate, which is sufficient to pull out the bundle in emergency situations. This was more than the minimum criterion of two tool fingers that was required to engage the bundle endplate at each trial.

8. CONCLUSIONS

The measurements taken with the CANFLEX fuel bundles confirmed the following:

a) Each CANFLEX bundle passed through the bent tube gauge under its own weight.

b) The bundles' dimensions were all within the specified limits.

e) There were always clearances between the FM sidestops and the CANFLEX bundle end plate which ensured no interference.

d) The bundle end dimensions "B" and "E" were within the specified limits.

e) The CANDU 6 grappling tool engaged on the bundle with a minimum of three fingers.

Therefore, it was concluded that the CANFLEX fuel bundles are compatible with the CANDU 6 FM and are not expected to interfere with the separator/sidestop assemblies. This will be verified in the actual CANDU 6 FM compatibility test which is scheduled for April 1996.

9. ACKNOWLEDGMENTS

The authors would like to thank the CANFLEX project for financial support and for permission to publish this paper. We would also like to thank A.M. Burns and T. White and their co-workers, involved in the bundle measurements used in this paper, and D.J. Koivisto and G.A. Corbett from the Fuel Handling Section for consultations during the work. Finally, we thank A.D. Lane and P.G. Boczar for their comments.

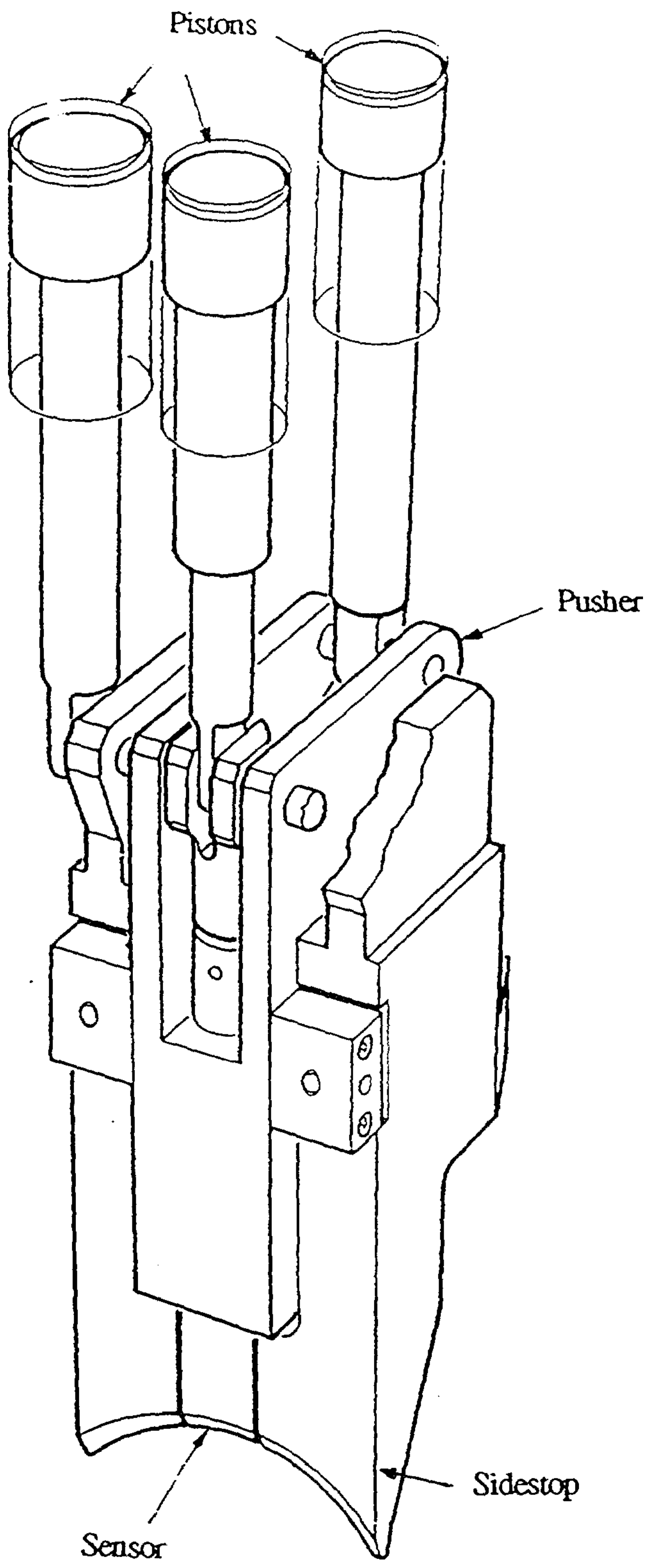
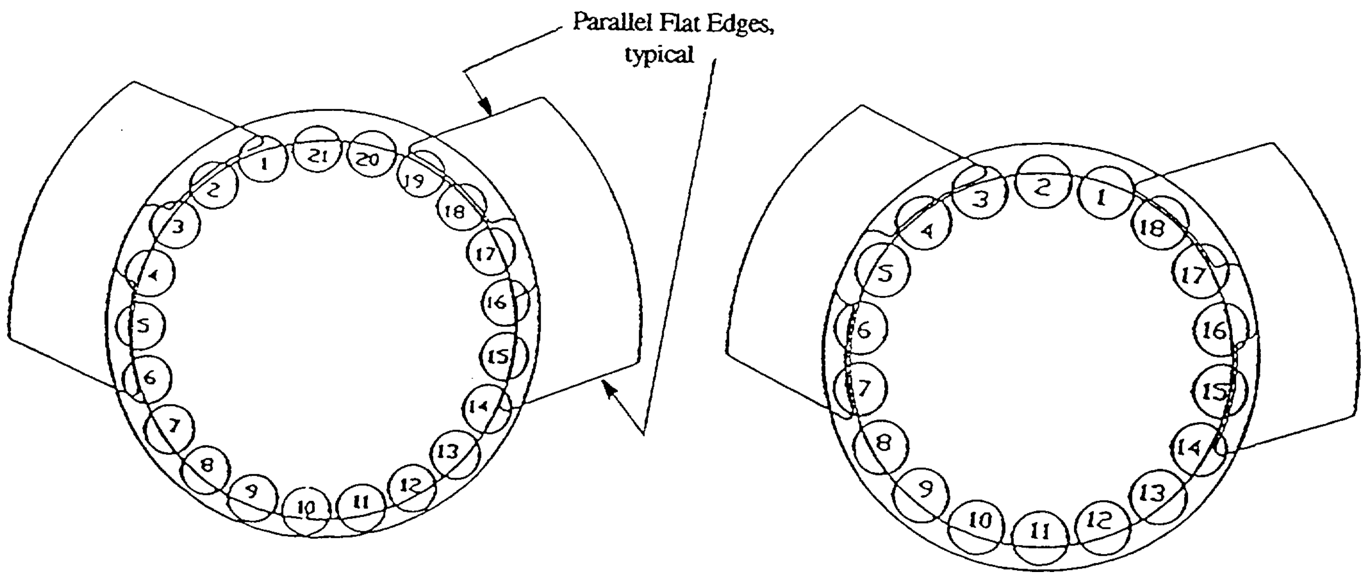


Figure 1: Fuelling Machine Separator Assembly



a. CANFLEX

b. CANDU 6, 37-Element

Figure 2: Clearance Between Sidestops and End Plate

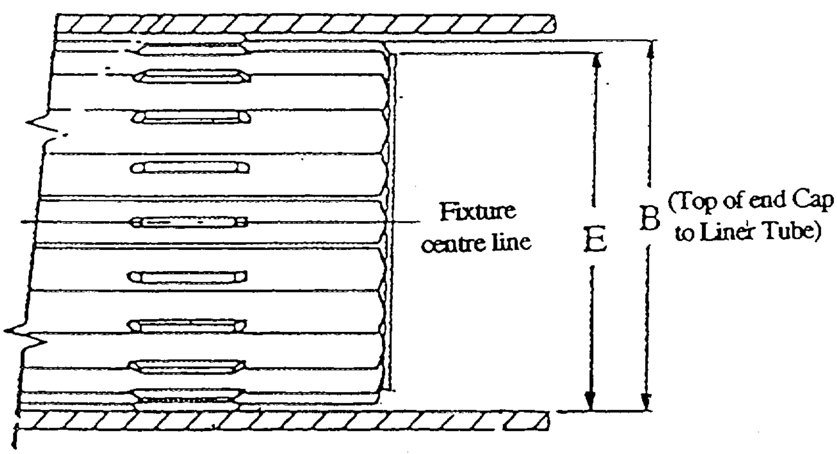


Figure 3: Bundle End Dimensions

DESIGN OF FUELLING MACHINE BRIDGE AND CARRIAGE TO MEET SEISMIC QUALIFICATION REQUIREMENTS

CA9700734

by

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ABSTRACT

During each refuelling operation, the boundary of primary heat transport system is extended up to Fuelling Machines. A breach in the pressure boundary of Fuelling Machine in this condition would cause a loss of coolant accident. Fuelling Machines are also used for transit storage of spent fuel bundles till discharged to fuel transfer system. Therefore fuelling machine including its support structures is required to be seismically qualified for both on-reactor (coupled) mode and off-reactor (uncoupled) mode.

The fuelling machine carriage used in the first generation of Indian PHWRs is a mobile equipment on wheels moving over fixed rails. As this configuration was found unsuitable for withstanding strong seismic disturbances, bridge type design with fixed columns was evolved for the next generation of reactors.

Initially, the seismic analysis of the fuelling machine bridge and carriage was done using static structural analysis and values of natural frequencies for various structures were computed. The structures were suitably modified based on the results of this analysis. Subsequently, a detailed dynamic seismic analysis using finite element model has been completed for both coupled and uncoupled conditions. The qualification of the structure has been carried out as per ASME section III Division 1, sub section NF.
Details of the significant design features, static and dynamic analysis, results and conclusions are given in the presentation.

1.0 INTRODUCTION

Fuelling machines of Indian PHWRs are designed to facilitate on power loading and unloading of fuel into/from the reactor core. Two identical fuelling machines are clamped on to a coolant channel and the sealing and shielding plugs are removed prior to movement of the fuel. The boundary of the primary heat transport system is thus extended to include the fuelling machines during every refuelling operation. A breach in the pressure boundary of the fuelling machines in this condition would cause a loss of cooling accident. Therefore, fuelling machine including the support structures are required to be seismically qualified, for on-reactor (or coupled) mode. The fuelling machines, after receiving the spent fuel are moved to fuel transfer port, to discharge the spent fuel to the fuel transfer system. Fuelling machines thus act as a transit storage for spent fuel and provide cooling to the bundles during this period. The fuelling machines including the support structures, thus are also required to be seismically qualified for off- reactor (or uncoupled) mode.

The fuelling machines used in the first generation of Indian PHWR's (RAPS and MAPS) consisted of two columns mounted a small carriage base structure fitted with wheels and it moves horizontally on rails embedded in the floor (see fig.1). As this configuration was found to be unsuitable for withstanding strong seismic disturbances anticipated at the being considered sites for nuclear reactors, the fuelling machine carriage was redesigned as bridge type with fixed columns for future generation of standardised design of 220 MWe PHWRs, beginning with Narora (see fig.2). The same bridge concept has been adopted for the 500 MWe PHWR reactors.

In this presentation, first the significant design features, considerations for design and methodology for static seismic analysis of fuelling machine bridge and carriage are discussed. In the latter part of the presentation the dynamic seismic analysis of fuelling machines and its support structures in coupled and uncoupled modes is discussed.

2.0 DESCRIPTION

The fuelling machine (FM) (See Fig.2 and 3) consists of FM head which contains complicated mechanisms for manipulating the fuel, plugs and other accessories and FM bridge and carriage which supports and positions the FM head to any desired reactor channel or FT port.

The head consists of front snout, pressure housing which houses magazine and ram housing at the rear. The head is supported on the support frame. The support frame, in turn is supported in the lower gimbals through horizontal trunnion pins and is free to tilt in a vertical plane. A spring loaded levelling mechanism keeps FM head horizontal and also limits the tilt. The lower gimbals is bolted with upper gimbals, to facilitate the removal of FM head for servicing. The upper gimbals is supported on 4 linear ball bearings fitted on the top beam. The relative motion between upper gimbals with respect to top beam provide Z-motion for FM head. The top beam is suspended from the drive plate by a vertical spindle. The vertical spindle through a set of gears permits 90 deg. rotation of FM head to facilitate its passage through hatchway into the service area for connection to FT port or for maintenance of fuelling machine. A small amount of tilt about the vertical spindle in horizontal plane is allowed at the extreme positions of 90 deg. rotation. A spring loaded centralising mechanism keeps the head in centralised position and also limits the tilt in the horizontal plane.

The drive plate is bolted to the trolley by 4 support studs. The trolley moves horizontally over the guides fixed to the underside of the bridge structure. The trolley is moved by a rack and pinion drive.

The bridge structure is a long welded beam, which is supported on the bridge supports. Bridge supports are guided vertically on guides fitted to two columns. One end of the bridge is bolted to the bridge support, the other end is freely supported on linear bearings to facilitate thermal expansion of the long bridge structure. The bridge supports are supported on 4 ball screws which are suspended from the top of two columns. Each column is a built up I-section which is supported on a base plate embedded in concrete and is tied with the nearest wall with help of the tie members.

All structural members like the column, the bridge structure, the bridge supports, the trolley, the drive plate, the gimbals and the support frame are designed to be fabricated as welded structures from ASME A 515 gr. 70 material.

3.0 CONSIDERATIONS FOR DESIGN AND METHODOLOGY FOR STATIC ANALYSIS

Fuelling machine bridge and carriage has been designed as per ASME section III subsection NF as applicable to class I components. However, the primary considerations of rigidity of structures has been incorporated to limit the deflections due to normal operating loads. For example, the cross-section of bridge structure has been so sized that the deflection of bridge for the worst combination of operating loads should not exceed 1.5 mm.

Initially, the effect of seismic loads due to postulated seismic event was considered in the design on the basis of static structural analysis. The analysis was done using the moment distribution method by calculating relative stiffness of various members. The end fixity conditions for various fixed bolted joints were conservatively fixed between 0.5 and 0.75. The calculations for bridge and column were done to determine the worst position of bridge on the column and FM head on the bridge, and to determine maximum deflections at such a position. The values of natural frequencies of vibration in different directions were computed. Similar calculations were done for all other structures including gimbals and vertical spindle. These calculations resulted in the following modifications in the design:

a) Originally an integral radiation shield on top of the bridge was contemplated for shielding of FM service area from FM vault. The concept was changed and a separate Roll-on shield was provided on Fm Vault floor.

b) The columns were tied at a number of points to the wall by tie members to provide more rigidity by reducing lengths of unsupported spans.

c) Additional bracings were provided in the column, bridge and gimbals to strengthen the structures.

d) The rating of X and Y drive brakes were augmented considering the forces generated by seismic disturbance.

4.0 DETAILED DYNAMIC SEISMIC ANALYSIS

Detailed dynamic seismic analysis using finite element technique for both coupled mode and uncoupled mode has been carried out for the new design of fuelling machine bridge and carriage by using SAP-IV computer code (Ref.1)

4.1 FINITE ELEMENT MODELLING

The finite element model using 3-D beam and 3-D stiffness element of SAP-IV computer code has been developed for both coupled and uncoupled mode of fuelling machines. Fig.4 shows the model for the coupled mode. Fig.5 shows closer details of model for FM head and carriage. The model has been evolved so as to depict the structures and the joints as realistic as feasible.

The fuelling machine bridge and carriage has a number of connecting node points where the connecting members are having multiple 'FREE' degrees of freedom. Suitable 'end release code' facility of SAP IV computer code is used at such nodes to simulate the exact end conditions and relative motions of various elements in the above models. The connection between ball screws and columns, ball screws and bridge support, bridge support and bridge, drive plate and vertical spindle, top beam and upper gimbals etc. are some of the points where end conditions and relative element motions have been simulated in detail.

4.2 QUALIFICATION METHODOLOGY

Fuelling Machine Bridge and Carriage has been analysed for dead load and two levels of earthquake viz. operating basis earthquake (OBE) and safe shutdown earthquake (SSE). The dynamic analysis using response spectra method has been done using SAP- IV code. SSE analysis has been carried out using 4 % damped SSE response spectra (Fig. 6) for horizontal direction.(Ref.2) The SSE response spectra used is envelope of response spectra for different floors/wall elevations supporting the FM bridge and carriage structures. Multiplication factor of 0.67 has been used for vertical direction spectra accelerations. The value of damping was chosen as per the ASME code (Ref.3) OBE analysis has been done using 70 % of 2 % damped SSE response spectra (see fig.7) .

Qualification of the structures has been done as per requirements of ASME III division 1 sub section NF (Ref. 4) . The load combination for supports for different loading categories are taken as follows :

Design condition -- Dead wt.+OBE loads
Level C -- Dead wt.+SSE loads.

4.3 RESULTS

The dynamic analysis of fuelling machine bridge and carriage has been conducted to extract 32 modal frequencies for coupled mode and 16 modal frequencies for uncoupled mode (see table-1). The stresses and deflection experienced by various structures in different elements were calculated and checked. Table 2 and 3 show the maximum stressed structural members. The deflections for various modal frequencies were combined by SRSS method. Table 4 shows the maximum deflection for coupled and uncoupled mode.

5.0 CONCLUSION

The results of the detailed dynamic seismic analysis shows that the structures are relatively rigid for the seismic excitations. The results have also given strength to the view that initial conservative static analysis calculations remain a useful tool in the process of designing such structures.

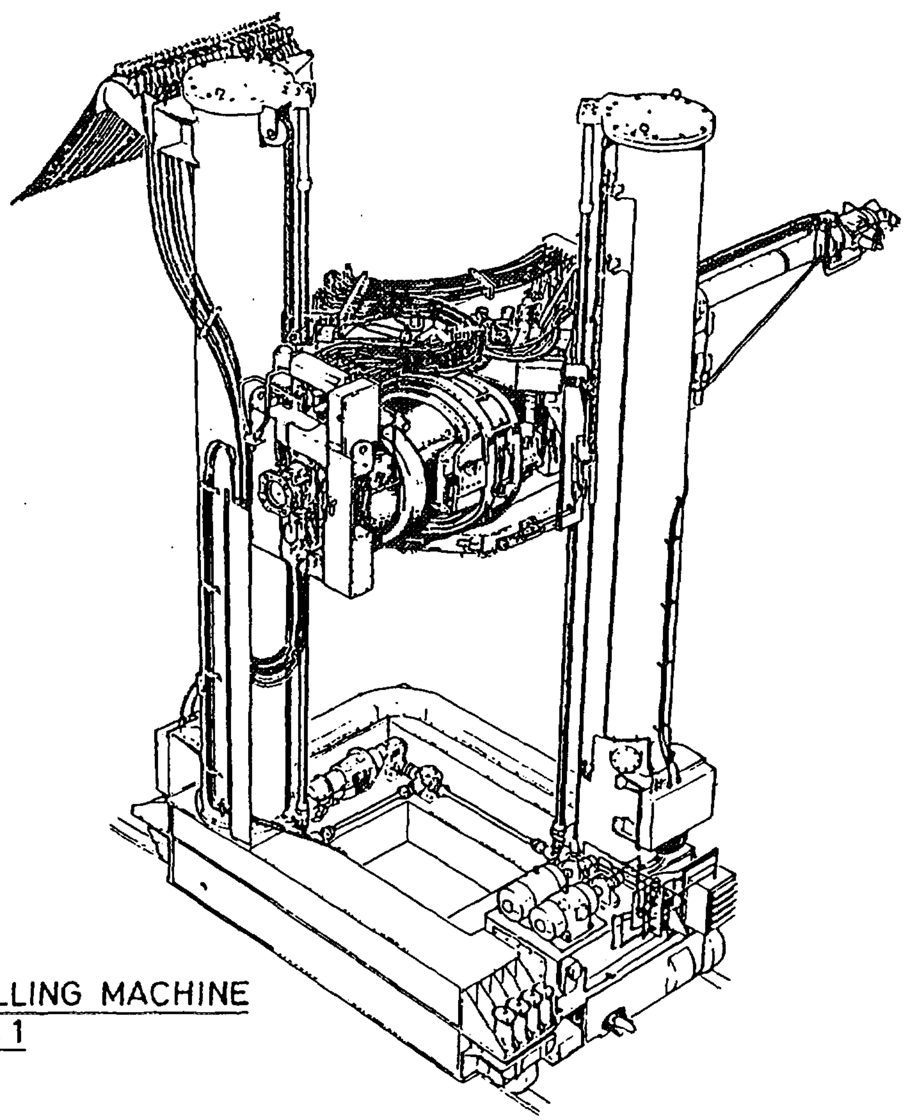
REFERENCES :

1. Computer Code SAP- IV (A *structural analysis* programme for static and dynamic response of linear systems by K.J. Bathe et.al.,Engineering Analysis Corporation, Berkeley, USA)

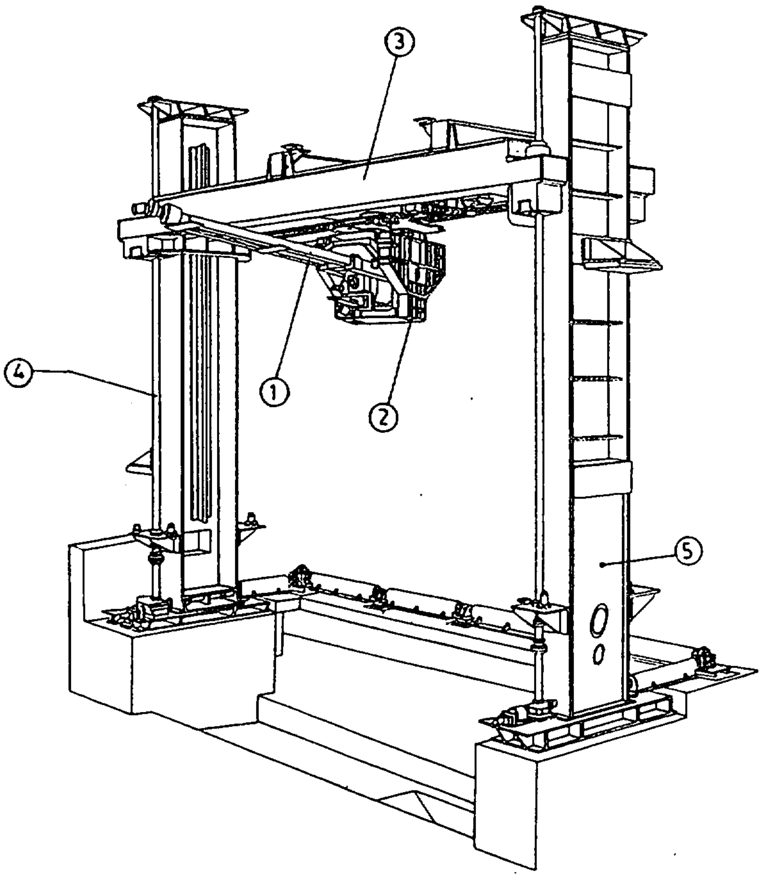
2. Earthquake Engineering studies Eq-7771, Seismic Analysis of RB of NAPP under horizontal ground motion including flexibility of internal structures, A.S.Arya et. al. University of Roorkee.

3. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Appendices, Para N-1230.

4. ASME Boiler and Pressure Vessel Code Section III Division 1, subsection NF, Components Supports, para NF-3320.

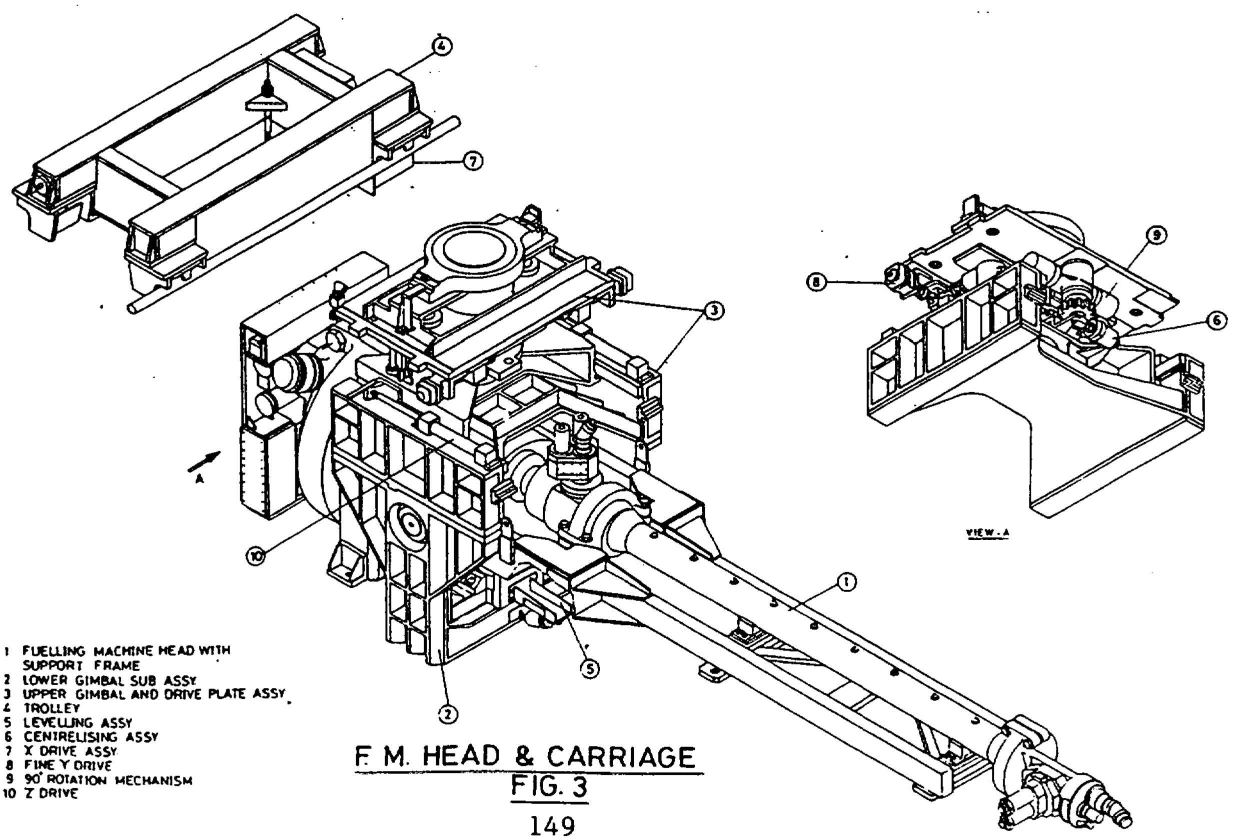


RAPS / MAPS FUELLING MACHINE
FIG. 1



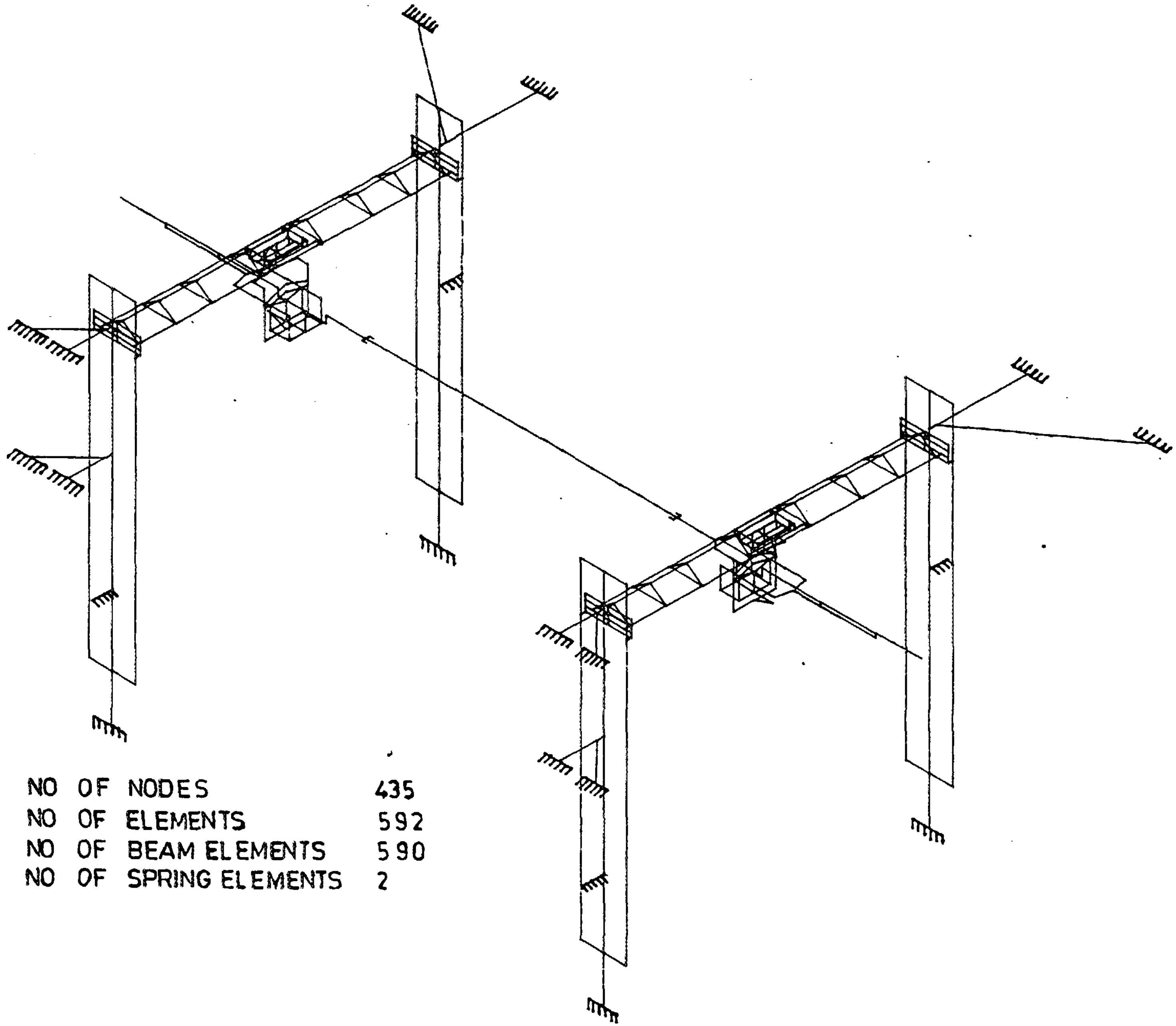
1 F M HEAD
2 GIMBALS
3 BRIDGE
4 BALL SCREW
5 COLUMN

NAPS FUELLING MACHINE
FIG. 2

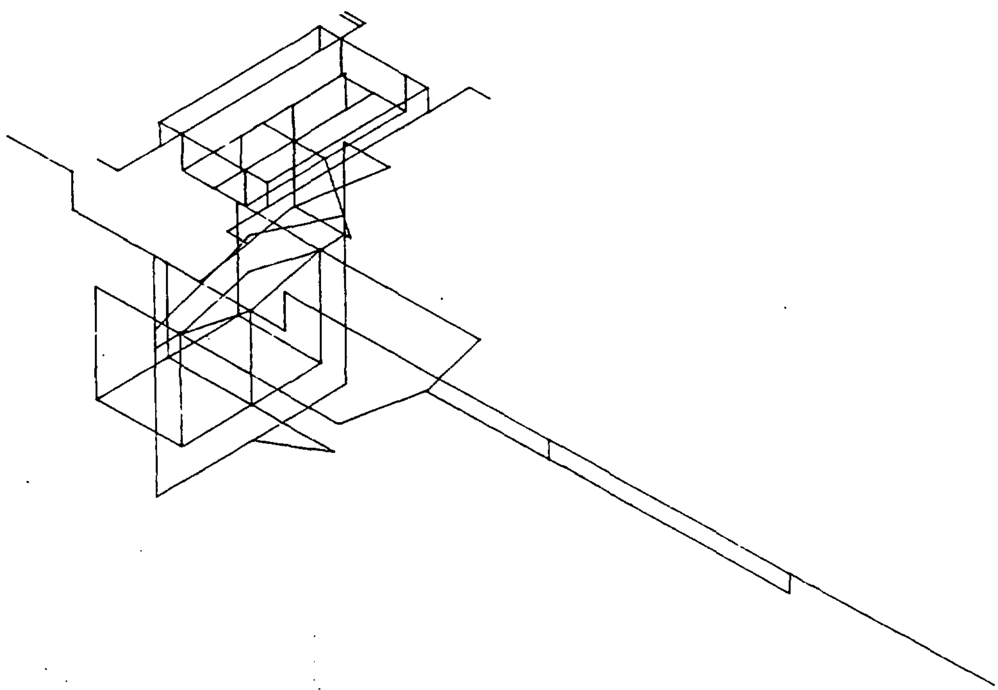


1 FUELLING MACHINE HEAD WITH SUPPORT FRAME
2 LOWER GIMBAL SUB ASSY
3 UPPER GIMBAL AND DRIVE PLATE ASSY
4 TROLLEY
5 LEVELLING ASSY
6 CENTRELISING ASSY
7 X DRIVE ASSY
8 FINE Y DRIVE
9 90° ROTATION MECHANISM
10 Z DRIVE

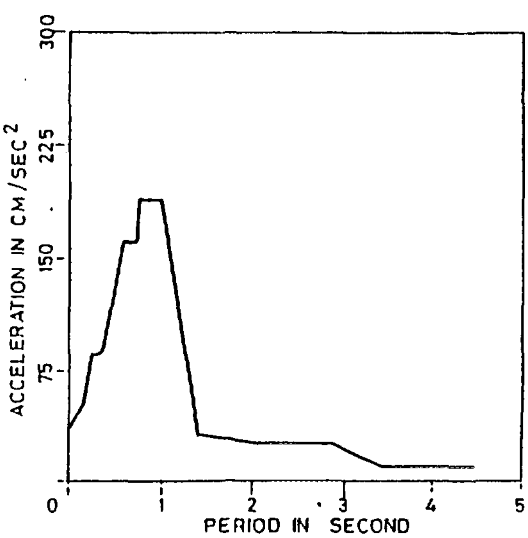
F. M. HEAD & CARRIAGE
FIG. 3



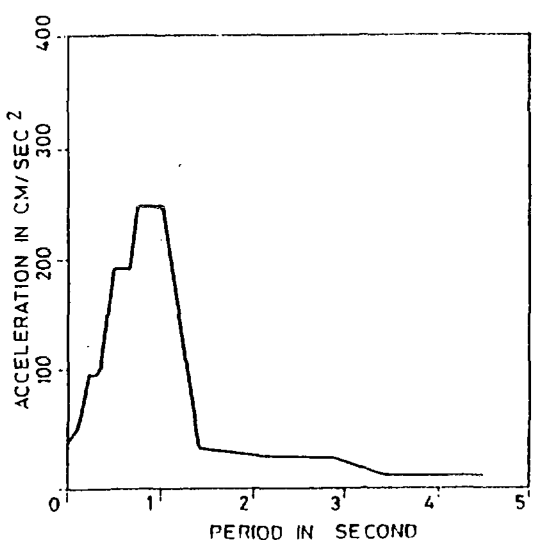
FINITE ELEMENT MODEL OF
FUELLING MACHINES IN COUPLED MODE
FIG. 4



FINITE ELEMENT MODEL OF
F. M. HEAD & CARRIAGE
FIG. 5



SSE ENVELOPE RESPONSE
SPECTRUM WITH 4% DAMPING
FIG. 6



OBE ENVELOPE RESPONSE
SPECTRUM WITH 2 % DAMPING
FIG.7

TABLE -1

MODAL FREQUENCY DATA

COUPLED MODE			UNCOUPLED MODE		
MODE NUMBER	**FREQUENCY (CYC/SEC)**	**PERIOD (SEC)**	**MODE NUMBER**	**FREQUENCY (CYC/SEC)**	**PERIOD (SEC)**
1,2	3.637	0.2750	1	3.261	0.3066
3,4	3.640	0.2747	2	3.637	0.2750
5-16	3.663	0.2730	3	3.640	0.2747
17	5.933	0.1686	4	3.663	0.2730
18	6.018	0.1662	5	3.663	0.2730
19	6.139	0.1629	6	3.872	0.2582
20	6.191	0.1615	7	3.873	0.2582
21	9.369	0.1067	8	3.874	0.2581
22	10.23	0.09778	9	3.874	0.2581
23	10.67	0.09370	10	6.326	0.1581
24	15.42	0.06484	11	9.042	0.1106
25	15.49	0.06458	12	10.280	0.09724
26,27	18.23	0.05484	13	17.230	0.05803
28	22.22	0.04501	14	23.940	0.04176
29	22.75	0.04395	15	31.210	0.03204
30	31.34	0.03190	16	40.920	0.02444
31	31.35	0.03190			
32	34.48	0.02900			

TABLE-2

MAXIMUM STRESS FACTORS
(UNCOUPLED MODE)

STRESS TYPE	DESIGN CONDITION	LEVEL C CONDITION	STRUCTURAL MEMBER
SHEAR STRESS	0.6053	0.4484	RAM HOUSING
AXIAL COMPRESSIVE STRESS	0.1428	0.1200	SUPPORT STUD
AXIAL TENSION STRESS	0.3399	0.2974	SUPPORT STUD
COMBINED AXIAL COMPRESSION AND BENDING STRESS	0.9668	0.6953	RAM HOUSING
COMBINED AXIAL TENSION AND BENDING STRESS	0.9654	0.6941	RAM HOUSING

TABLE-3

MAXIMUM STRESS FACTORS (COUPLED MODE)

STRESS TYPE	DESIGN CONDITION	LEVEL C CONDITION	STRUCTURAL MEMBER
SHEAR STRESS	0.0798	0.06317	UPPER GIMBAL
AXIAL COMPRESSIVE STRESS	0.7679	0.7176	COOLANT TUBE
AXIAL TENSION STRESS	0.1499	0.1102	SUPPORT STUD
COMBINED AXIAL COMPRESSION AND BENDING STRESS FACTOR	0.9300	0.8230	COOLANT TUBE
COMBINED AXIAL TENSION AND BENDING STRESS FACTOR	0.7080	0.5734	END FITTING

TABLE-4

MAXIMUM DEFLECTIONS

STRUCTURAL MEMBER	UNCOUPLED MODE (CM.)	COUPLED MODE (CM.)
1.RAM HOUSING TAIL END(Y-DIR)	3.3	0.8
2.BALL SCREW (X-DIR)	1.8	1.8

STRUCTURAL ANALYSIS OF FUEL HANDLING SYSTEMS

by

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ABSTRACT
The purpose of this paper has three aspects: (i) to review "why" and "what" types of structural analysis, testing and report are required for the fuel handling systems according to the codes, or needed for design of a product, (ii) to review the input requirements for analysis and the analysis procedures, and (iii) to improve the communication between the analysis and other elements of the product cycle.

The required or needed types of analysis and report may be categorized into three major groups: (i) Certified Stress Reports for design by analysis, (ii) Design Reports not required for certification and registration, but are still required by codes, and (iii) Design Calculations required by codes or needed for design.

Input requirements for structural analysis include: design, code classification, loadings, and jurisdictionary boundary. Examples of structural analysis for the fueling machine head and support structure are given.

For improving communication between the structural analysis and the other elements of the product cycle, some areas in the specification of design requirements and load rating are discussed.

1. INTRODUCTION

Quite often the analysis part of the product cycle is hidden in the underworld or undervalued if not ignored, or is taken for granted. This paper reviews and highlights the main features involved in the structural analysis and hopefully promotes mutual understanding and appreciation among the analysis and other elements of the design-analysis-fabrication-installation-operation-and-maintenance product cycle of the fuel handling systems in a CANDU nuclear power plant.

2. WHY AND WHAT TYPES OF ANALYSIS AND REPORT

Why and what types of analysis and report are required or needed? As part of the Component Design Document (per CSA N285 series, References 1 to 4) or the Design Output Documents (per ASME Boiler and Pressure Vessel Code, Article NCA-3550, Reference 5), types of analysis, testing and report required for structural integrity achievement for fuel handling components and supports are summarized below:

(1) Certified Stress Reports for design by analysis (together with Certified Review of Design Report by the owner or his designee) are required for design registration for the following components and supports:

(a) Class 1, 1C and 4 components and their supports.

(b) Class 2 and 2C vessels designed to ASME NC-3200.

(c) Class 2 or Class 3 components designed to Service Loading greater than design loading.

Due to the early contract award requirements for some of the major fuel handling equipment a Provisional (Preliminary) Design Registration is usually applied and obtained when, at the time of application, the final Design Report or the final drawings are not completed. This allows for component fabrication or system installation to commence before a final design registration is obtained. In such a case, a Certified Provisional Report (or Base Design Report) may be substituted for a final design report. The Provisional Report only has to include the Design and Testing Conditions, not any Service (A, B, C and D) Conditions (see Section 4 for the design requirements). The purpose of the stress calculations is to show that primary stresses are within allowable limits to prove adequacy of material thickness.

(2) Certified Design Report Summary may be furnished in lieu of a Certified Design Report for standard supports (Class 1, 2, 3 or 4) designed by analysis (provided by the manufacturer).

(3) Design Reports not requiring certification and registration are required by the codes for:

(a) Class 2 and 2C non-standard support.

(b) Class 3 and 3C pressure-retaining systems and components.

(4) Load Capacity Data Sheet and catalogue for Class 1, 2, 3, 4 and 6 standard supports to be qualified by load rating method. The Load Capacity Data Sheet shall identify the tests and calculations used to establish the load capacity (per Article NCA-3551.2). This is important in order that the user (designer) can interpret correctly the load ratings used in the design and analysis (see Section 6 below). Also, it shall be certified by a Registered Professional Engineer for supports for Class 1 components, Class 4 vessels, and Class 2 vessels designed to NC-3200.

(5) Design calculations are needed for design and upon regulatory request are required for:

(a) Class 3 and 3C component supports.

(b) Class 6 pressure-retaining components and their supports.

(c) Design deviation due to fabrication errors (e.g. during post-order engineering).

(d) Modification of an existing design, e.g. change of configuration or material.

(e) Change of operating conditions, e.g. hot or ambient temperature F/M D_2O supply.

(f) Change of installation conditions, e.g. deviation of bolt pre-load torque, material substitute, layout change, etc.

(g) Item replacements or repairs due to fabrication discrepancy, actual or anticipated item failure (e.g. fatigue life shorter than the plant life).

(h) Conceptual design for firming up the member sizes.

(i) Others as needed, e.g. for intervening elements.

The Fueling Machine (F/M) Head Pressure Boundary is classified as Class 1 pressure-retaining component in CSA N285.0 and N285.1, and by reference, is designed to the requirements of the ASME Subsection NB. The Fueling Machine Support systems are usually composed of structural supporting elements and mechanisms which are classified as Class 1C supports in CSA N285.0 and N285.2, and by reference, are designed to the requirements of the ASME Code, Subsection NF Class 1 component support. The classification of 1C, instead of Class 1, component support in CSA N285.2 is because that there is no rules in the ASME code for mobile support. Portions of the F/M supports, (e.g. an elevating bridge and carriage, and a mechanism such as a ball screw and nut assembly) have a mobility not usually found in supports for pressure-retaining components. One important consequence of the CSA classification is that materials specified in CSA-N285.6.9 can be used, which otherwise may not be ASME NF material.

No matter why or what type of analysis and document is required, the essential purpose of analysis is to ensure the structural integrity and the adequacy of the product design under various service conditions.

3. CLASSIFICATION AND JURISDICTIONARY BOUNDARIES OF COMPONENTS AND SUPPORTS

Classifications of pressure-retaining systems, components, and their supports are stipulated in CSA N285.0. The jurisdictionary boundary consideration involves boundaries between components and their supports, attachments, intervening elements, and building structures. The boundaries for Class 1, 2, 3, 4 and their supports are shown in NB-1130, NC-1130, ND-1130, NE-1130 and NF-1130 respectively. The specific boundaries of jurisdiction between these structures shall be clearly defined, e.g. in the Design Specification.

For the purpose of defining the jurisdictional boundary between a component or piping support and the building structure, they should be shown in respective drawings (civil/structural drawings or support drawings). The key criterion for NF support structure is that they are installed and used for the "primary purpose" of supporting piping or components. In general, a bolted connection between the NF support structure and the building structure should be designed as part of the building structure. Whereas, if the means by which the NF structure is connected to the building structure is a weld, the weld shall fall within the jurisdiction of the NF support. One simple rule for determining whether the connection design belongs to NF can be that: the analysis of the connection design need not involve the design of the building structure, otherwise, the connection design should be part of the building structure.

An attachment is an element in contact with or connected to a component or support structure. It may have either a pressure retaining or nonpressure-retaining function and either a structural or nonstructural function. Structural attachment which has pressure retaining function or is in the support load path should be treated as part of that pressure-retaining component or the support structure.

Sometimes confusion may be caused by intermittent structural elements as to whether they should be treated as NF supports or Intervening Elements. Intervening Element by definition is a structural element in the support load path for a pressure-retaining component for which a major purpose is other than to "passively support" the component. It is noted that the "means" (bolted or welded) by which the component or piping support is connected to the intervening element shall fall within the jurisdiction of component or piping support. Intervening Elements are not registered, but when required by the regulatory authority, design criteria and supporting calculations shall be submitted.

4. DESIGN REQUIREMENTS AND ANALYSIS PROCEDURES

Input requirements for structural analysis must be provided before analysis can proceed. The design basis of plant and system operating and testing conditions is stipulated in NCA-2140 and is based on the system safety criteria and operability of components and supports. Based on the plant and system operating and test conditions, Design, Service (A, B, C and D), and Test Loadings are established; its criteria are explained in NCA-2142.4

The structural analysis is performed by a designer or analyst as an N Certificate Holder. The structural integrity and safety achievements are ensured and stipulated in the codes partly by:

(a) control of material.

(b) only certain types of design or construction are allowed, e.g., for welded and bolted flanged connections.

(c) acceptable stress analysis procedures and stress limits with safety margins which are compatible with the class of construction and specification of loadings.

On the other hand, the operability of components (whether the design works for the functional purposes), including leaking, seem not to be emphasized by the codes. The assurance of operability is up to the owner (or his designee) to define the appropriate limiting parameters. However, code rules apply to the operability of pressure relief valves.

4.1 Design Requirements

The rules for construction of Class 1, 2, 3 and 4 (MC) nuclear components and their supports are given in the ASME Code, Subsection NB, NC, ND, NE and NF respectively. The owner shall provide or cause to be provided the design requirements and a design verification report for intervening elements. Design requirements should be defined in documents, e.g. design specifications. Article NCA-3252 stipulates the required contents of Design Specification. As a minimum, the design requirements should include the following information:

(a) Design, including drawings, and material specifications including impact tests.

(b) Code classification of components.

(c) Jurisdictionary boundary.

(d) Loadings for Design, Service and Test Conditions.

The loading conditions that shall be taken into account in designing component or support are specified in Article NB-3111, NC-3111, ND-3111, NE-3111 and NF-3111 respectively for Class 1, 2, 3, 4 components and their support structures.

4.2 Analysis Procedures

The rules of analysis procedure for Class 1, 2, 3 and 4 (MC) nuclear components and their supports are given in the ASME Code, Subsection NB, NC, ND, NE and NF respectively. Requirements for acceptability of design are stipulated in Article NB-3211, NC-3211.2, ND-3300, NE-3211 and NF-3131. They can be demonstrated by analysis or experiment tests. The analysis method can be either design-by-analysis or by design rules (e.g. ND-3300). For design by analysis, the classical method and/or finite element method can be adopted depending on the complexity of the structure geometry, the load types (pressure, temperature or seismic loads, etc.) and the requirements of the code allowable stress limits (primary vs. secondary stress, allowable stress intensity vs. allowable maximum stress, etc.). Highlights of the considerations for structural analysis in accordance with Subsection NB and NF are given below.

Class 1 Components (NB)

(1) The design details shall conform to the general rules given in NB-3130, including the minimum required thickness of shells.

(2) The stress limits for Design, Service, and Test Conditions are based on the stress intensity (i.e. maximum shear stress theory). Fatigue evaluation shall be considered for Service Lever A, B and Test Conditions.

(3) Protection against nonductile fracture shall be provided.

(4) Buckling should be evaluated, e.g. under external pressures.

NF Support Structure

(1) Types of supports are given in Article NF-1212, NF-1213, NF-1214. Standard supports and catalog items are supplied by a Quality System Certificate Holder as material, including Certification of Load Capacity Data Sheets and Design Report Summary.

(2) Analysis procedure by: (i) design by analysis, (ii) experimental stress analysis, and (iii) load rating method.

(3) NF support needs not include thermal or peak stress, except for high cycle fatigue, n > 20,000 cycles, for Class 1 Linear Type support.

(4) Buckling should be evaluated, e.g. for beam type elements of F/M support bridge and columns.

(5) Protection against nonductile fracture for Class 1 component and piping support should be considered.

(6) There are three types of supports: (i) Plate-and Shell-Type support, (ii) Linear Type Support, and (iii) Load Rated Support. The stress limits for Class 1 Plate-and Shell-Type supports are defined by the design stress intensity (Sm) which is based on the maximum shear; others are defined in terms of the allowable stresses (S) which are based on the yield strength of material and the maximum stress (principal stress). For bolting, the limits are based on the yield strength and the ultimate strength.

5. ANALYSIS EXAMPLES

5.1 F/M and Support Structure Seismic Analysis

Earthquake loads are part of the loading conditions (as Level C Service loads) for the F/M head pressure boundary components and the support structure. They are also required for the interfacing systems, i.e., F/M process system, reactor structure, fuel channels, and feeders of the PHTS. Seismic analysis of the F/M and the support structure is therefore carried out to generate the seismic loads.

The seismic analysis methodology follows the requirements and procedures of the National Standard of Canada CAN3-N289.3-M81 (Reference 6). Seismic models have been constructed using beam and spring elements for various systems, e.g. the F/M, the support structure and the reactor. For examples, see Figures 1, 2, and 3. To account for various operation modes during the re-fueling process, seismic models representing various configurations have been developed:

(a) F/M attached or unattached to the reactor in the reactor vault area, with the F/M located at seven representative fuel channel locations (A11, E03, E20, K11, P02, S20, W11).

(b) F/M on the maintenance lock track, five configurations were considered: (i) unattached at centre of track, (ii) unattached at new fuel port location, (iii) unattached at spent fuel port location, (iv) attached to new fuel port, and (v) attached to spent fuel port.

For the F/M seismic analysis, the input earthquakes are the F/M support points motions, in terms of floor response spectra or acceleration time-histories, which are generated from the reactor building seismic analysis by the Civil design group. The input motions take into account the effects of the variation of soil conditions at the site and the sensitivity due to the uncertainties of the structural properties (frequencies). The seismic loads resulted from the seismic analysis are represented in terms of nodal accelerations, beam end loads and third-level floor response spectra. These seismic loads are then used in the seismic qualifications (by analysis or test method) of the affected systems.

5.2 Stress Analysis of F/M Head Pressure Boundary
The F/M head assembly consists of a number of major sub-assemblies: a snout assembly, a magazine assembly, a ram assembly and two separators. The housings for these sub-assemblies form the pressure boundary of the F/M head. Loads and load combinations for Design, Service Level A, B, C, and Test Conditions for the F/M head pressure boundary were defined in a design specification. Final detailed stress analysis and report was prepared and certified by the author and the third-party reviewer. It formed part of the submission for the final design registration in accordance with the requirements of References 1 and 5.

The methodology adopted in the stress analysis of the F/M head pressure boundary components makes use of an optimum combination of classical and finite element (FE) methods. The stress analysis is based on linear elastic static analysis except for some assemblies in which non-linear gap elements are used in the FE method to simulate the interaction behaviours under various loading conditions at the contacted face between two components.

In the finite element method, the trend is to utilize computer-aided capabilities for modeling and meshing. Ideally, the mechanical design automation tools used should able to provide direct interface between the design models, the drafting models and the analysis solid models. The analysis solid models are usually simplified to remove unnecessary details. Figure 4 shows such a FE model for the magazine housing, which was generated by using I-DEAS software package (by vendor SDRC).

5.3 Stress Analysis of F/M Support Structure
The F/M head assembly and the cradle assembly are supported from the F/M carriage. The carriage is suspended from rails on the bridge in the reactor vault area (Fig. 2) or on the track frame in the maintenance lock area. The F/M support structure is analyzed in accordance with the analysis procedures stipulated in Subsection NF (as described in Section 4.2) as:

(a) Beam (linear) type elements in the cradle assembly, the bridge and the columns.

(b) Non-beam type elements (plate and shell) in the cradle, the carriage, the bridge and the column assemblies.

(c) Load rated mechanisms of manufacturer's proprietary components in the carriage, and the bridge-elevator interface. There are 19 load rated components used in the F/M support structure as listed in Table 1.

6. INTERACTIONS BETWEEN ANALYSIS AND OTHER ASPECTS OF PRODUCT CYCLE
Various elements of the product cycle need the service of structural analysis (see Section 2) while the latter requires input from the former (see Section 3 and 4). Mutual understanding, appreciation, and efficient communication among them are important for a successful product. Experience indicates that some areas within the interfaces warrant improvements. Examples are given below:

(1) The required information for analysis should be provided "timely and adequately". This is vital in order to avoid repeated analysis. Prior to contract award, it is essential that the requirements (e.g. design specifications) be clear and available.

(2) Have a section on "Requirements for Analysis" included in the design specifications to provide specific instructions for analysis. Examples can be:

(a) The component can be defined as piping, while the analysis, for convenience, can be based on the requirements for vessel.

(b) Analysis can be optionally based on a higher class (usually means higher allowable limits) than the class of which the component is classified. In this case, the material requirements for the higher class shall be satisfied.

(3) For non-standard NF supports, do not specify (e.g. on drawings) the type of support, whether Plate- and Shell-Type or Linear-Type. The type of support to be assumed in the analysis should be determined by the analyst depending on the geometry complexity and the load distributions.

(4) Procedures for determining load ratings for Plate- and Shell-Type and Linear-Type support are provided in ASME NF-3280 and NF-3380 respectively. The

load ratings are defined for Level A, B and C (including seismic) Service Loading. The load ratings provided by the manufacturers, in terms of test reports, Load Capacity Data Sheets or catalogues, should be compatible with the NF definitions. For instance, the load ratings given in catalogue as "static" or "dynamic" can be determined by testing conditions and failure mechanisms which may be quite different from the required ASME NF testing procedures or the actual operating conditions. Similarly, the "capacity" rating given in some catalogues can be established based on the fatigue life while the load ratings stipulated in the ASME NF procedure are based on the ultimate load failure criterion. Any discrepancy in the compatibility of the definition between the ASME NF and manufacturer's can create confusion and mis-application for analysis.

(5) If a Provisional (Preliminary) Design Registration has been applied and obtained, see Section 2, therefore, component fabrication might have commenced before the results of the final structural analysis are completed. It can happen that the *Service Conditions (considered in the final Design* Report, but not in the Provisional Report), e.g. seismic loads or fatigue life, indicates that the preliminary design may not be adequate. This requires an extra effort on the structural analysis to remove any fictitious overstress that might result in order not to modify design during manufacture. Close coordination between the structural analysis, design, manufacturing and maintenance (e.g. item replacement due to short fatigue life) are vital in order to find an acceptable solution.

7. CONCLUSIONS

This paper describes the basic requirements and procedures for structural analysis in accordance with the codes. To satisfy the code requirements is necessary, however, to have the analysis work done in a cost effective manner is vital to the overall success of a project. This paper has highlighted some areas for the improvement of the work method, especially the interface between the structural analysis and the other elements of the product cycle.

8. REFERENCES

(1) National Standard of Canada CAN/CSA-N285.0, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants".

(2) National Standard of Canada CAN/CSA-N285.1, "Requirements for Class 1, 2, and 3 Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants".

(3) National Standard of Canada CAN/CSA-N285.2, "Requirements for Class 1C, 2C, and 3C Pressure-Retaining Components and Supports in CANDU Nuclear Power Plants".

(4) National Standard of Canada CAN/CSA-N285.3, "Requirements for Containment System Components in CANDU Nuclear Power Plants".

(5) ASME Boiler & Pressure Vessel Code, Section III, Subsection NCA, NB, NC, ND, NE and NF.

(6) National Standard of Canada CAN3-N289.3, "Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants".

Table 1 NF-Grade Load Rated Components in F/M Support Structure

No.	F/M Support Structure	Load Rated Component
1	Cradle Trunnion Bearing (Fixed End)	Spherical Roller Bearing
2	Cradle Trunnion Bearing (Free End)	Cylindrical Roller Bearing
3	Cradle Ram Assembly Cam Followers	Rollerway
4	Carriage Wheel	Wheel
5	Carriage Wheel Bearing	Wheel Bearing
6	Carriage 'Z' Guide Cam Followers	Camrol Bearing
7	Carriage Cam Follower	Camrol Bearing
8	Carriage Fine 'Y'-Drive Screw Jacks	Screw Jack
9	Gimbal Roundway Bearing	Roundway Bearing

Table 1 (continued)

No.	F/M Support Structure	Load Rated Component
10	2" Gimbal Roundway	Roundway
11	Upper Gimbal Turntable Bearing	Sleeving Bearing
12	Carriage Seismic Clamp	Thrust Bearing
13	Carriage 'Z' Motion Drive	Hydraulic Cylinder
14	Elevator Roundway Bearing	Roundway Bearing
15	Elevator Roundway	Roundway
16	Bridge Ball Screw	Ball Screw
17	Bridge Screw Jack	Screw Jack
18	Bridge Screw Nut	Ball Nut Thrust Bearing
19	Bridge/Elevator Camrol Bearing	Camrol Bearing

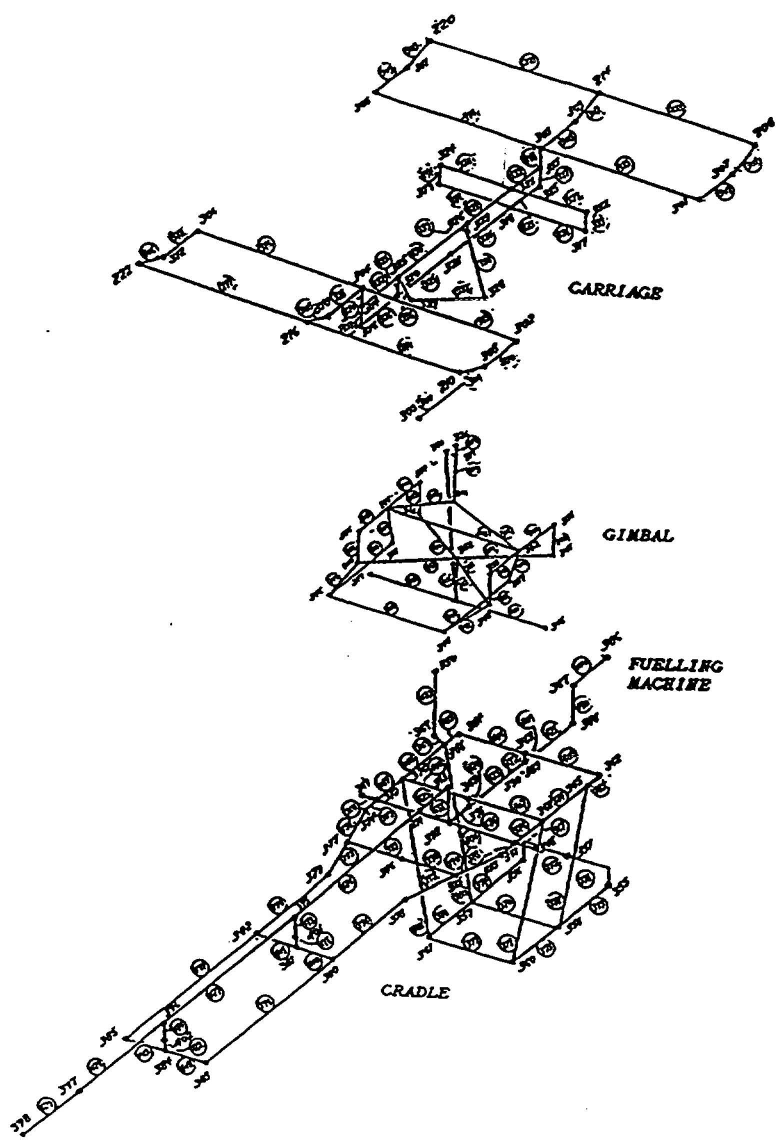


FIGURE 1 FUELING MACHINE AND CARRIAGE SEISMIC MODEL

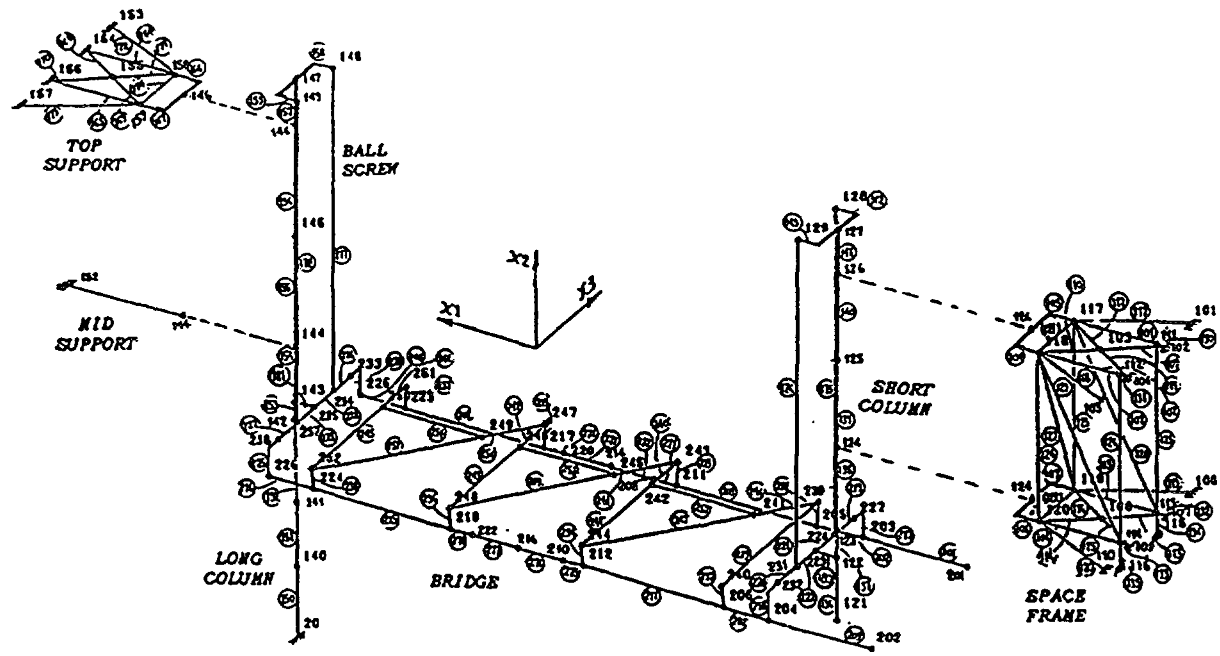


FIGURE 2 BRIDGE AND COLUMN SEISMIC MODEL (for Fuel Channel Location W11)

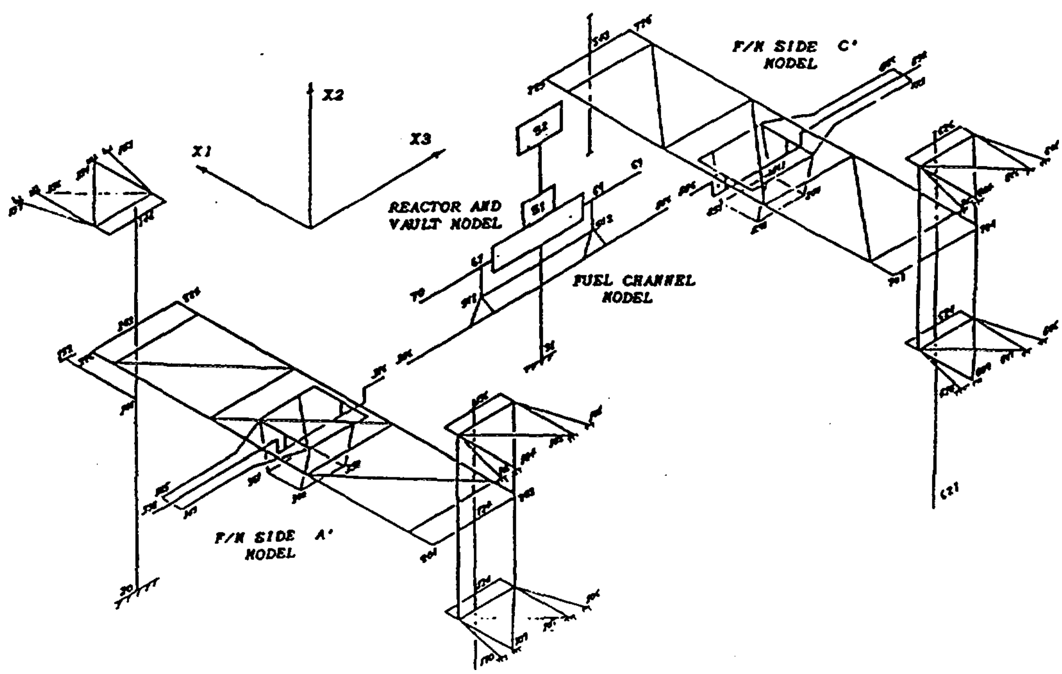
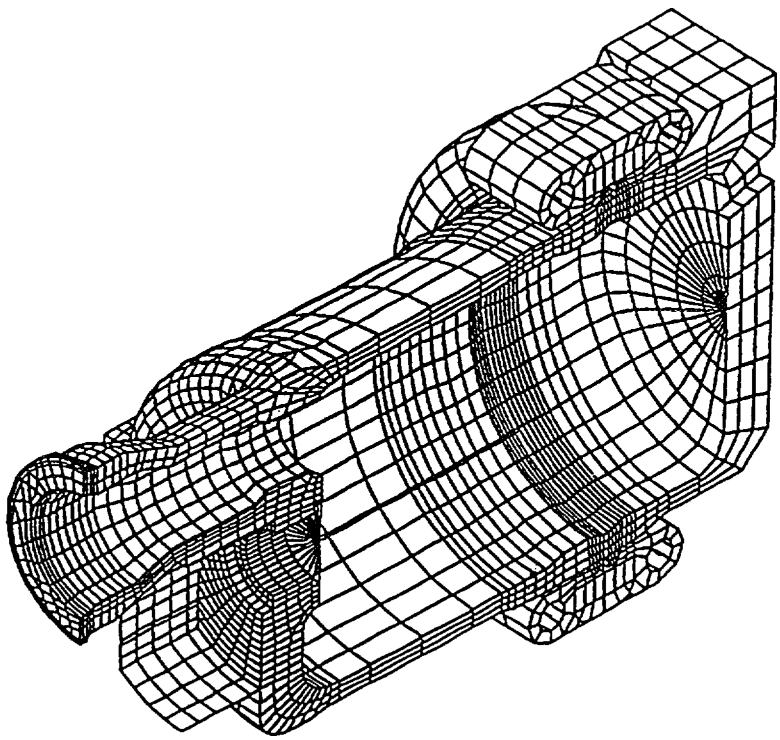
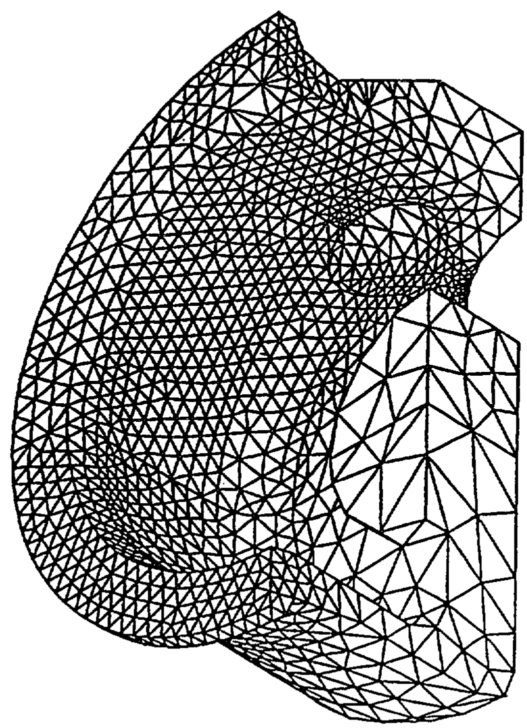


FIGURE 3 ATTACHED FUELING MACINE AND REACTOR SEISMIC MODEL



(a) Global Model



(b) Submodel of the End Cover

FIGURE 4 FINITE ELEMENT MODEL OF F/M MAGAZINE

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PICKERING DRY STORAGE

COMMISSIONING AND INITIAL OPERATION

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ABSTRACT

Having commissioned all individual conventional and nuclear systems, the first Dry Storage Container (DSC) was loaded with four modules of 17 year cooled irradiated fuel (366 bundles) in the Auxiliary Irradiated Fuel Bay (AIFB) on November 29, 1995. After decontamination of the outer surface, and draining of water, the DSC was transported to the Used Fuel Dry Storage Facility (UFDSF) workshop, where it was vacuum dried, and then the lid was welded on. Following successful radiography test of the lid weld, the DSC was vacuum dried again and backfilled with Helium to a pressure of 930 mbar(a). The Helium leak test showed zero leakage (allowable leak rate is 1×10^{-5} cc/sec). Finally, after loose contamination checks were performed and permanent safeguards seals were applied, the DSC was placed in the UFDSF storage area on January 23, 1996.

Radiation fields at contact with the DSC surface were <0.6 mrem/hr, and at the exterior surface of the storage building wall only 33 micro-rem/hr (far below the target of 250 micro-rem/hr). Therefore, the actual dose rates to general public (at the exclusion zone boundary) will be well below the design target of 1% of the regulatory limit.

1. INTRODUCTION

Implementation of the dry storage concept at Pickering Nuclear Division (PND) followed a successful demonstration program, which used two concrete containers loaded with six and ten years cooled irradiated fuel from the AIFB (1988-1992). Results of this program confirmed both the technical and economical viability of the dry storage system as an alternative to the conventional wet storage system. The current DSC design offers two main features:

- interim dry storage of irradiated fuel for up to 50 years
- a means of transportation to its final disposal destination without the need to re-package irradiated fuel

The UFDSF will provide additional on-site storage for about 576,000 irradiated fuel bundles, or approximately 1,500 DSCs by the end of station operating life (year 2025). The UFDSF currently has an operating license as well as a transportation license (for DSC off-site shipment) granted by the Atomic Energy of Canada Board (AECB).

Safeguards are applied to the UFDSF and DSCs as per the requirements of the International Atomic Energy Agency (IAEA) regulations. Surveillance cameras are installed in the UFDSF workshop and storage area, and permanent safeguards seals are applied to each DSC by installing two fibre optic cables and attaching "cobra" seals.

2. COMMISSIONING

2.1 DSC Pre-Loading Activities

New DSCs are delivered to the UFDSF workshop where they are unloaded from a truck and temporarily stored. Prior to transferring the first empty DSC (#OH-0001) to the AIFB, it was inspected for corrosion and other physical damage. A lid and transfer clamp were installed on the DSC body to verify a proper fit. Also, four empty modules were placed into the DSC to verify proper clearance between them and the DSC cavity, and that the lid would not contact the uppermost module. Each DSC has two ports, a vent at the top and a drain at the bottom. The internal threads of these two ports were inspected for any damage and repaired as necessary. With the lid taken off the DSC body, the main weld groove was carefully inspected for any rust. None was found. The prepared DSC was placed in the temporary laydown area, ready to be taken to the AIFB (ref. Fig. #1).

At the same time as the DSC inspection, two operators were recording at the AIFB serial numbers of the bundles to be loaded into the DSC. This was done using a telescope and recording equipment specially fabricated for this task. The process was quite lengthy (it took about 3 days vs. the 4 hours expected), but, at the end, a complete record of the 366 bundle serial numbers was available. This requirement came from the AECB and IAEA in mid 1995 and was agreed upon by PND. Before the loading of the DSC, IAEA staff performed the following verifications:

- 100% item count, (i.e. 366 bundles)
- Non Destructive Analysis (NDA) of irradiated fuel
- correctness of serial numbers by checking 8 out of 366 bundles

This task was very cumbersome, and took about 12 working hours vs. the 2 hours originally planned.

2.2 Loading of DSC in AIFB

The fully prepared DSC was transported from the UFDSF workshop to the AIFB along the route north of the powerhouse (Fig.#1) by the on-site transporter vehicle (Fig.#3). In the AIFB, the lid was taken off and the DSC was filled with demineralized water. On November 29, 1995 it was submersed into the AIFB, placed on an impact pad, and loaded with four modules of approximately 17 years cooled irradiated fuel; a total of 366 bundles. This operation was performed under surveillance of the IAEA and AECB staff and various station personnel. Once the lid and in-bay clamp were positioned and engaged on the DSC, it was hoisted out of the water. Total in-water time was 1 hour, 43 minutes, which is considerably less than the originally estimated 3 to 4 hours. Radiation fields on the DSC after loading were measured as 0.5 mrem/hr at contact and non-detectable at working distance. It should be noted that radiation fields at contact rose marginally after draining as the shielding from the water was lost.

The DSC was then immediately washed in the decontamination pit for about 15 minutes with water heated to about 75°C. After placing the DSC back on the handling pad, it was allowed to dry before taking smears from the surface. The smears showed zero surface contamination, which was much better than expected. It reflects the positive aspects of maintaining clean bay water, minimizing in-water time and ensuring a prompt washing with hot water after removal from the bay water.

2.3 Draining and Vacuum Drying in AIFB

On November 30, 1995 the water in the DSC was drained back into the AIFB by gravity, with both drain and vent ports open. It took 15 minutes to reach the point where no water was flowing through the drain port. The drain port was then closed, and the transfer clamp was installed on the DSC flange.

This method of draining, although it had worked well during "dry" commissioning, left about 55 litres of water in the DSC cavity versus the expected 10 litres. The draining of the water from subsequent DSCs was vastly improved by tilting the DSC in two different directions. It is believed that most of the water was trapped amongst the fuel bundles in the module tubes.

The vacuum drying system was connected to the DSC drain port and allowed to run for 14 hours. Water was continuously removed from the DSC, and internal pressure was lowered from atmospheric to 13 mbar(a). Although the target of 2.5 mbar(a) was not achieved, and the DSC had obviously not been completely dried, it was decided to transport it to the UFDSF and resume vacuum drying there.

2.4 Transport of DSC to UFDSF

Final contamination surveys found only one area on the base of the DSC with 1000 cpm of loose contamination which was easily removed. This is a sign that the sweating phenomenon that had been observed during the demonstration program is not taking place. Gamma surveys found fields still at about 0.5 mrem/hr at contact and zero at working distance.

By means of a special "O" ring seal placed in the welding groove, a sub-atmospheric pressure of less than 830 mbar(a) was established in the DSC cavity. This sub-atmospheric pressure was maintained during the DSC transport through the station yard to prevent the possible spread of loose contamination.

On December 1, 1995, having been picked up by the Transporter vehicle (fig.#3), the DSC was taken to the UFDSF workshop. Along with station and AECB personnel, IAEA staff accompanied the DSC as part of "human surveillance" on its 18 minute trip to the UFDSF (fig.#1). The Transporter worked flawlessly throughout this process. At the UFDSF, the DSC was first dropped off by the Transporter, then moved by crane to welding station #3. IAEA staff then applied a security seal to the base, which was an additional safety measure to being monitored by the permanent surveillance cameras. The DSC internal pressure rose only marginally during the trip.

2.5 Vacuum Drying in UFDSF

Upon arrival in the UFDSF, about 20 litres of water was collected through the drain port. It is believed that this water was forced out from the module tubes by vibrations during the trip through the yard. This was followed by two days of vacuum drying in an effort to completely dry the DSC internals. However, internal pressure only got down to 7 mbar(a) and the presence of water was still evident. In an attempt to attain the required dry condition that had been observed during the testing stage, vacuum drying of the DSC proceeded non-stop for two more days. On December 5, 1995 an internal pressure of 5.5 mbar(a) was achieved with no further water being removed. This was deemed to be an acceptable condition for the initial stage of drying prior to welding the lid to the DSC body.

2.6 Welding and X-ray

The flange weld, about 30 feet long, is made by two automatic welding heads, each traveling along the circumference of the lid on opposite sides (fig.#4). Prior to welding, the DSC flange must be preheated to 105°C, while a slight vacuum is drawn in the DSC cavity to prevent any spread of contaminants. During earlier tests, it was confirmed that a pre-heat period of 16 hours is required to literally "soak" the DSC lid and body with heat so it will not cool down before the first welding pass is done. The design estimate had been 8 hours.

Preparation of the DSC for welding of the flange took several days, and some of the work was done in parallel with vacuum drying. First, primer paint had to be ground off the weld groove down to bare metal. Then, the welding machine was installed and put through a series of checks. A problem with one of the trailing cameras was discovered and quickly rectified. By December 7, 1995 the preheaters and thermocouples were installed and function tested.

Having sufficiently pre-heated the DSC flange, the pre-weld vacuum system was disconnected and the first weld pass completed on December 12, 1995. Constant monitoring of the process from the control room, as well as frequent local visual inspection by the welders, was required to ensure a flawless weld. It took about two days to complete eleven passes, seven of which were done on the first day.

The welding machine was removed from the DSC, the X-ray machine installed and the weld allowed to cool down for 48 hours. In the meantime, the DSC internals were maintained slightly sub-atmospheric by connecting the drain port to the active ventilation system. The flange weld was inspected using the X-ray machine (radiography). To reduce the X-ray hazard, it was performed in the evening hours, and was completed on December 16, 1995. There were no flaws found and the weld was deemed acceptable.

2.7 Helium Backfill and Leak Test

During the final vacuum drying, which took less than 2 hours, a pressure of 1 mbar(a) was reached, indicating that practically no water remained inside the DSC cavity. It was subsequently backfilled with Helium to about 930 mbar(a).

Manual welding of the vent and drain ports took substantially more time than expected. However, the welds were all found to be good and did not require any re-work. Each weld required 8 hours vs. the 2 hours originally estimated.

The Helium leak test requires repositioning the DSC to station #4, which has the necessary equipment and instruments for this test (fig.#5). The unique test method uses a two piece "bell-jar" so that the whole DSC can be tested at once. A vacuum is drawn in the space between the DSC outer surface and the bell-jar so that any leakage from the DSC cavity can be quickly detected. Having placed the DSC into the bell-jar, it took 26 hours to pull the required vacuum of 15-20 micro-bar(a). This lengthy process was expected and it is mainly due to outgassing of the DSC outer surface with its coating of epoxy paint. The Helium leak test was performed within the next five minutes, and it showed zero leakage (acceptable leak rate is $1x10^{-5}$ cc/sec). For all practical purposes, the DSC was ready to be put into final storage on December 21, 1995.

Having painted the welded surfaces, and having installed the safeguards "cobra" seals, the actual placing of the DSC in the storage area and a ribbon cutting ceremony took place on January 23, 1996.

2.8 DSC History Records

Detailed information will be computerized, with hard copies kept in the station's Q.A. vault. Included will be each DSCs serial number with manufacturing data, the modules' serial numbers, the age of the irradiated fuel, the position of modules within a DSC, the number of bundles in each module, date of loading, as well as X-ray and Helium leak test results. This has all been recorded in a checklist and verified by station and IAEA staff. The UFDSF storage area has a simple grid system to keep track of the DSCs' locations, and each DSC has a nameplate.

All information listed above is currently kept in the form of a "DSC history docket".

3. RESOURCE REQUIREMENTS

About 314 personhours were required to complete the first DSC, from start to finish. Contrary to original estimates, mechanics were in the highest demand, followed by operators and quality control staff (see Table 1).

Once the whole process becomes routine, it is expected to take about 160 personhours per DSC. Table 2 shows individual process activities, type and number of resources required and the number of hours needed to complete the first DSC.

TABLE 1 - RESOURCE REQUIREMENTS (in person hours)

OP's	MM	SM	QC	IAEA	P&M	RC
105	125	16	35	19	12	2
33.5 %	39.8 %	5.1%	11.2 %	6%	3.8%	0.6%

4. OCCUPATIONAL RADIATION SAFETY

4.1 Station Personnel

The shielding provided by the DSC proved to be much better than anticipated, and resulted in gamma fields of less than 0.6 mrem/hr at contact with the DSC surface and zero at 1 metre distance. Also, by keeping the bay water very clean, the gamma background fields in the AIFB are very low (about 0.25 mrem/hr). As a combination of these two factors, the gamma dose received by operators was 10 mrem per DSC (5 operators were involved). Similarly, the gamma dose received by mechanical maintainers was about 10 mrem per DSC. Other work groups received no measurable gamma dose. There was no Tritium dose received while working in the AIFB.

Therefore, the total radiation dose received by all work groups was about 20 mrem per DSC, which compares favourably with the original estimate of about 75 mrem per DSC.

4.2 General Public

Radiation fields at the outside surface of the storage building wall were about 33 micro-rem/hr, which is far below the target of 250 micro-rem/hr. These radiation fields may marginally rise as the number of DSCs in the storage building increases. However, the actual dose rates to the general public at the exclusion zone boundary will be well below the design target of 1% of the regulatory limit.

Air emissions to the environment from the UFDSF workshop (zone 3 from the radiation point of view) are monitored and were found to be zero. There has been no active liquid discharge from the UFDSF workshop thus far.

5. INITIAL OPERATION

The initial production runs took place from the "B" Irradiated Fuel Bay (IFB-B) because of the proximity of its fill-in date. They yielded a transfer capability of four (4) DSCs per two months. Full production transfer capability of eleven (11) DSCs every two months will be required by early 1997, just before the AIFB becomes full as well. A total of sixty-six (66) DSCs will have to be transferred from then on every year, assuming the eight Pickering reactors are operating at an average 85% Capacity Factor.

Major difficulties encountered thus far are listed below:

- Draining of water from the DSC was inadequate causing very long vacuum drying periods. This has been overcome by tilting the DSC in two different directions. However, the DSC Lifting Beam suffered minor damage since it had not been designed for tilting. A design change is expected to take place shortly.
- Welding of the vent and drain ports took two days. It will be reduced by using modified procedures.
- Welding of the lid to the DSC body (flange weld) was not satisfactory on the first 3 DSCs. Extensive re-work has been required due to porosity and lack of fusion, predominantly in the root of the weld (the first of eleven passes). An investigation is in progress while the welding of the flanges carries on at a slow and resource-consuming pace.
- The helium leak test on its own does not take more than 5 minutes. However, it requires a vacuum in the bell-jar of 15 - 20 micro-bar(a), which needs in excess of 24 hours to achieve. It is believed that re-calibration of the leak detector to operate at a vacuum of 1 mbar(a) will reduce the total test time to less than 4 hours.
- A lack of manoeuvring space in the IFB-B is slowing down the process. Particularly, the rationale for the portable impact pad is being questioned in view of limited DSC movements. Also, the manoeuvring sequence of equipment such as the in-bay clamp stand and lifting beam stand will have to be improved upon.

All of the above, plus other minor factors, contribute to the current transfer rate being about 15 calendar days per DSC. This transfer rate must be improved to about 5 calendar days per DSC by early 1997 in order to maintain the power level of the eight Pickering reactors at an average 85% Capacity Factor.

ACKNOWLEDGMENTS

The author wishes to thank Ms. C. Walker and Mr. D. Hunter for their help in preparing this paper.

REFERENCES

1. "Pickering UFDSF Safety Report", Ontario Hydro to the Atomic Energy Board, 1994.
2. R.N.Sumar and S.Jonjev, "Dry Storage of Irradiated CANDU Fuel at Pickering NGS", Proceedings of Annual Conference, Canadian Nuclear Society, June 1994.
3. D.Hunter, Work Plan #2-95-10768-00 "Commissioning of DSC in AIFB", November 1995.

TABLE 2 - UFDSF PROCESS

STEP	OPERATION	MANPOWER TYPE	PERSONS X HOURS
1	• RECORD SERIAL #'S OF ALL BUNDLES TO BE LOADED (4 MODULES). • INSTALL 4 MODULES ON THE INSPECTION STAND. • IAEA TO VERIFY I/F IN 4 MODULES ON THE INSPECTION STAND.	• OPERATORS • IAEA STAFF	2 X 16 HRS. 2 X 8 HRS.
2	• UNLOAD NEW DSC, INSPECT FOR CORROSION OR DAMAGE. • MM TO INSPECT VENT AND DRAIN PORT THREADS AND WELD GROOVE. • PERFORM DSC TEST MODULE CLEARANCE CHECKS. INSTALL DSC LID AND TRANSFER CLAMP. • CRANE PREPARED DSC TO TEMPORARY LAYDOWN AREA. FILL OUT INDIVIDUAL DSC RECORD SHEET.	• OPERATORS • MM	2 X 3 HRS. 2 X 2 HRS.
3	• TRANSPORT PREPARED EMPTY DSC TO AIFB OR BIFB USING DSC ON-SITE TRANSPORTER. • REMOVE DSC LID AND TRANSFER CLAMP. REMOVE TRANSFER CLAMP FROM LID AND INSTALL INBAY CLAMP. • FILL DSC WITH DEMINERALIZED WATER, SPRAY ALL EXTERNAL SURFACES OF DSC WITH DEMINERALIZED WATER.	• OPERATORS	2 X 6 HRS.
4	• MOVE DSC INTO IFB, LOAD FOUR (IAEA) VERIFIED MODULES OF USED IRRADIATED FUEL. • INSTALL DSC LID, ENGAGE INBAY CLAMP. • REMOVE LOADED DSC FROM IFB. MONITOR CONTINUOUSLY AND SPRAY ALL SURFACES OF DSC WITH DEMINERALIZED WATER.	• OPERATORS	3 X 2 HRS.
5	• DECONTAMINATE DSC IN DECONTAMINATION FACILITY. REMOVE INBAY CLAMP, INSTALL TRANSFER CLAMP AND ENGAGE. • DRAIN WATER FROM DSC TO IFB VIA DRAIN CONNECTION. PULL PARTIAL VACUUM AND VERIFY ACCEPTABLE TRANSFER CLAMP LEAK RATE. • CONTAMINATION CHECK OF DSC. (LESS THAN 200 CPM ABOVE BACKGROUND)	• OPERATORS	3 X 3 HRS.
6	• COMPLETE CONTAMINATION CHECKS OF DSC & TRANSPORTER. DRIVE LOADED DSC TO UFDSF, UNDER IAEA HUMAN SURVEILLANCE. • POSITION DSC AT SELECTED WELD STATION. REMOVE TRANSFER CLAMP AND "O" RING SEAL. • PREPARE WELDING GROOVE AND INSTALL WELDING MACHINE ON DSC.	• OPERATORS • IAEA STAFF • MM • RAD CONTROL	2 X 3 HRS. 1 X 1 HRS. 2 X 4 HRS. 1 X 2 HRS.
7	• INSTALL FLANGE PRE-HEATERS AND ACTIVATE (16 HR PRE-HEAT). DRAW WELD VACUUM AT DRAIN PORT, WITH VENT PORT OPEN. • SHUT OFF WELD VACUUM BEFORE THE FIRST WELD PASS IS STARTED. • WELD LID TO DSC BODY (AUTOMATED WELD 11 PASSES).	• OPERATORS • MM • P & M	2 X 1 HRS. 3 X 24 HRS. 3 X 2 HRS.
8	• WHEN FLANGE WELDING COMPLETED, REMOVE WELDING RIG AND PRE-HEATERS. • CONNECT VACUUM LINE FROM DRAIN PORT TO HV SUCTION HOLE ON WALL.	• OPERATORS • MM • P & M	2 X 1 HRS. 3 X 3 HRS. 3 X 2 HRS.
9	• ALLOW 48 HOURS FOR ALL WELDS TO COOL. • INSTALL X-RAY MACHINE. • RADIOGRAPH FLANGE WELD.	• OP's (MOVE X-RAY MACHINE) • QC (RADIOGRAPH FLANGE WELD)	2 X 1 HRS. 2 X 16 HRS. (2 NIGHTS)
10	• MANUALLY WELD VENT PORT. • DRAW VACUUM TO <1 MBAR, AND BACKFILL DSC WITH HELIUM. • MANUALLY WELD DRAIN PORT. • QC VENT AND DRAIN WELDS.	• OPERATORS • MM • QC	2 X 4 HRS. 2 X 16 HRS. 1 X 3 HRS.
11	• POSITION DSC AT WELD STATION #4. PERFORM DSC He LEAK TEST (BELL JAR METHOD).	• OPERATORS	2 X 6 HRS. (26 HR TO PULL VACUUM)
12	• INSTALL SAFEGUARD SEALS (IAEA COBRA SEALS).	• OPERATORS • IAEA STAFF	1 X 2 HRS. 1 X 2 HRS.
13	• PREPARE DSC FOR WELD AND TOUCH UP PAINTING. • COMPLETE ALL PAINTING AND DSC IDENTIFICATION LABEL APPLICATION.	• SM (PAINTERS)	2 X 8 HRS.
14	• COMPLETE FINAL DSC CONTAMINATION CHECKS (SPOT DECONTAMINATION AS REQUIRED). • ENGAGE DSC WITH TRANSPORTER AND MOVE TO STORAGE AREA.. • COMPLETE DSC INDIVIDUAL RECORD SHEET (HISTORY DOCKET).	• OPERATORS	2 X 3 HRS.

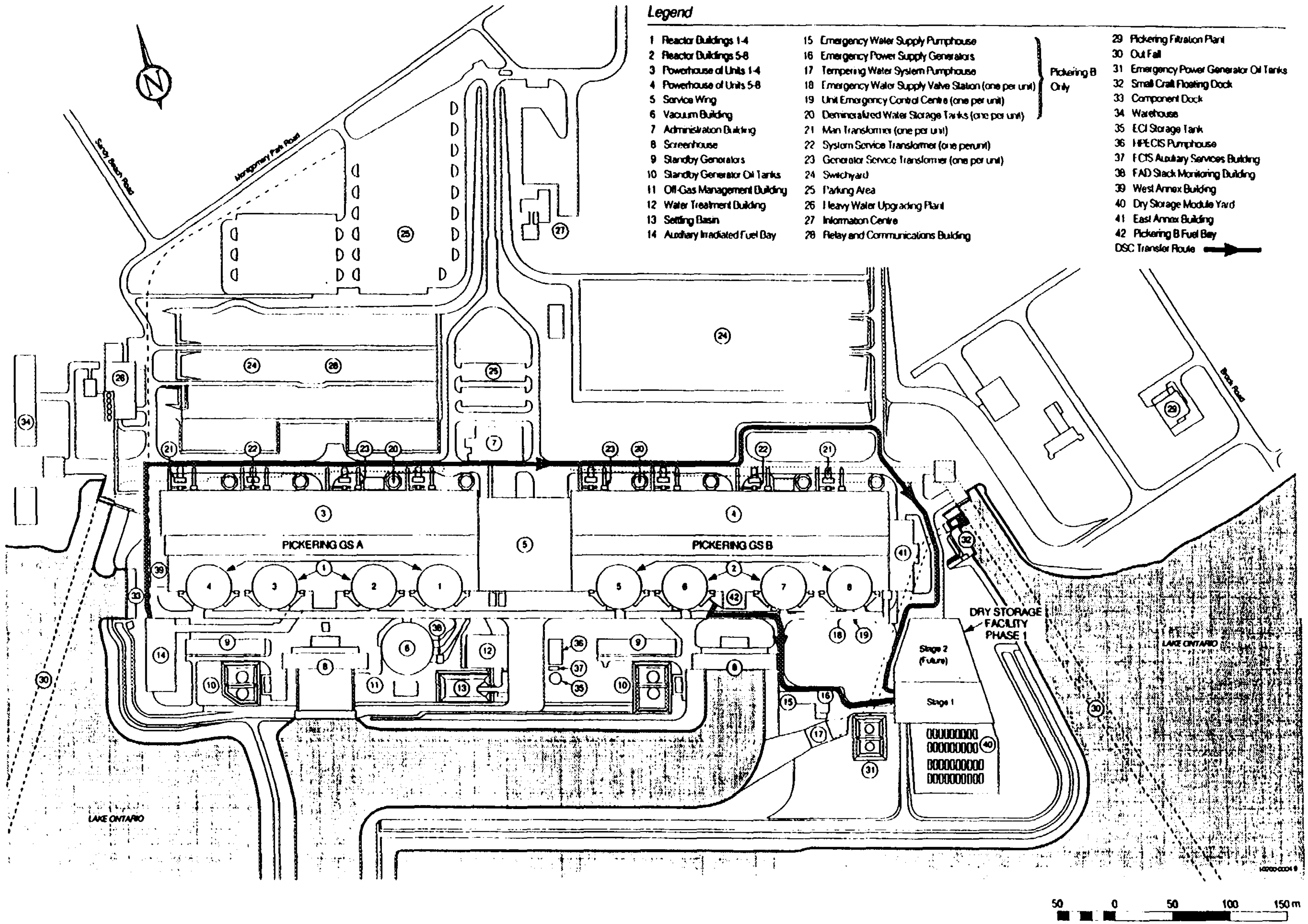
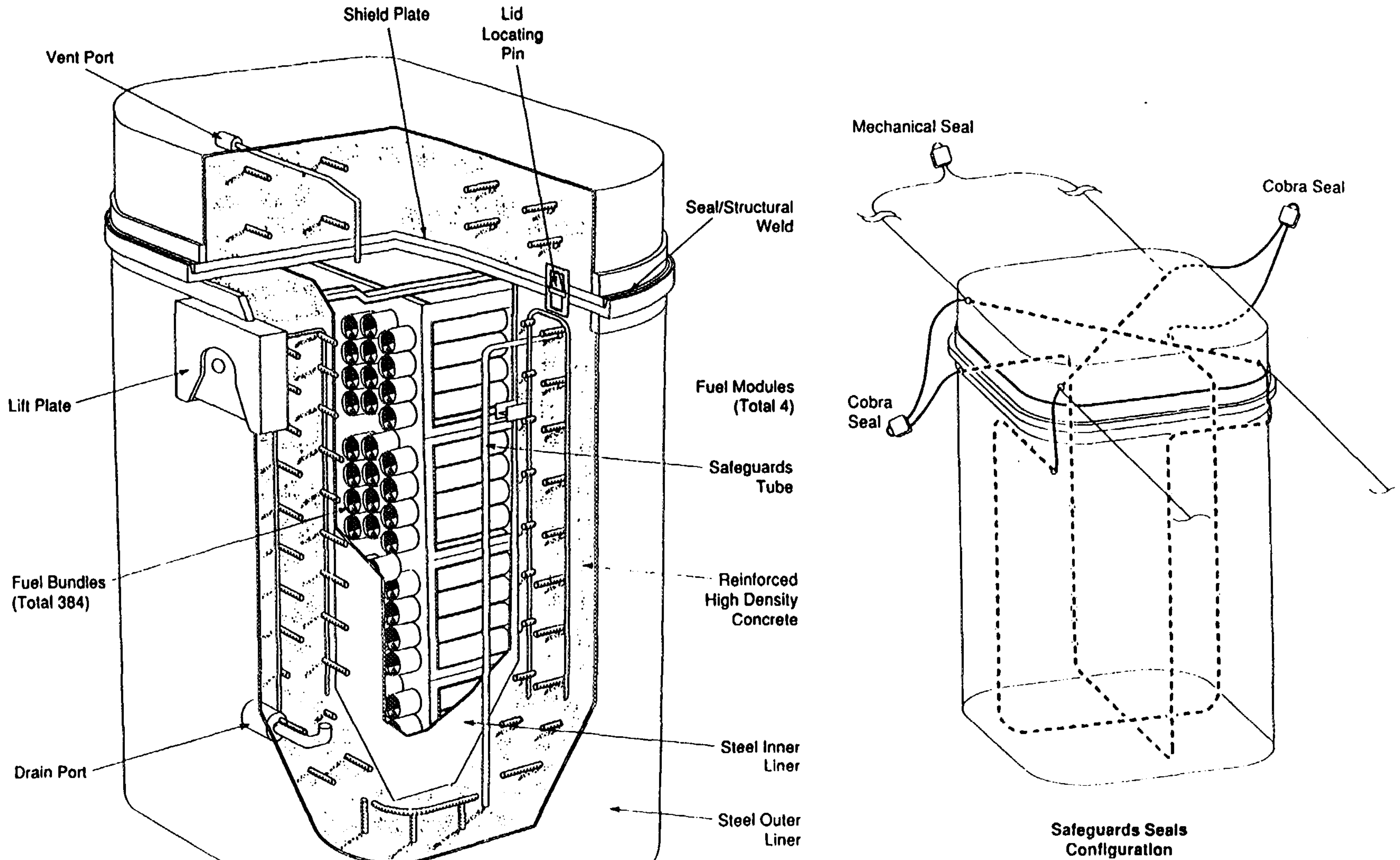


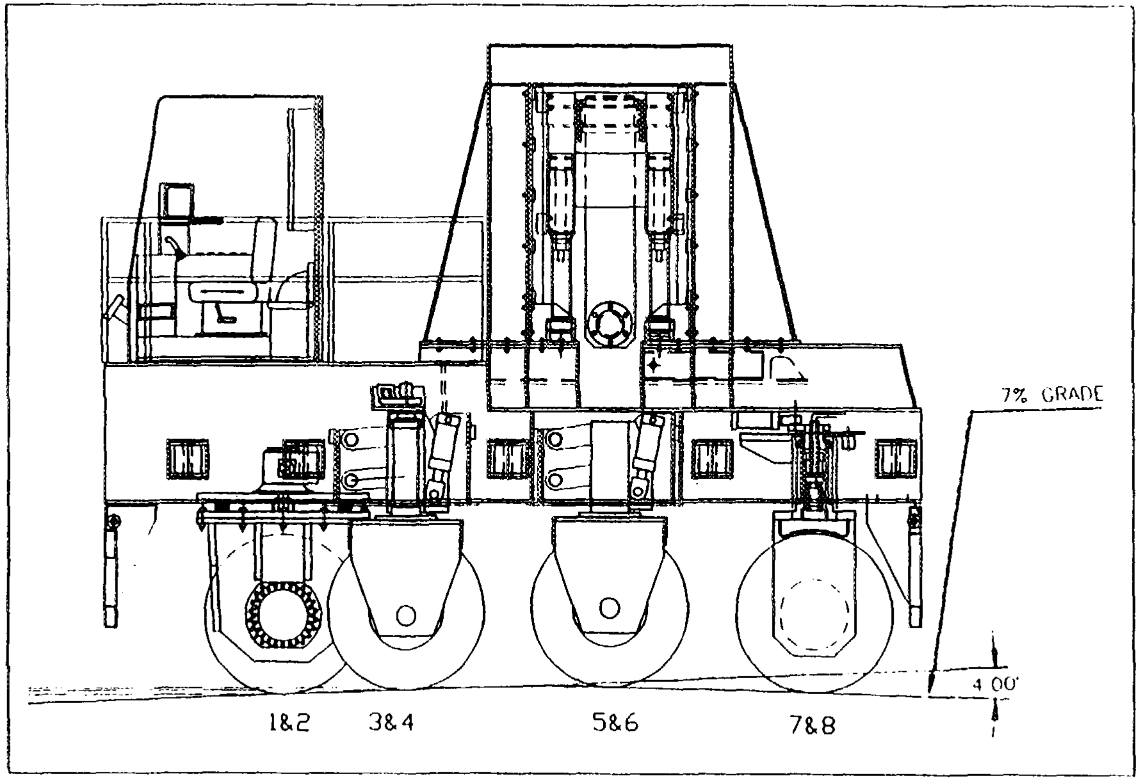
FIGURE 1
Site Layout



Data	
Height	3.55 m
Front	2.12 m
Side	2.42 m
Weight (Empty)	53 Mg
Payload	10 Mg

FIGURE 2
Dry Storage Container for Used Fuel Storage

UFDSF DSC ONSITE TRANSPORTER



7 PERCENT GRADE OPERATION
MAXIMUM

WHEEL #1&2 DRIVE/STEER
WHEEL #3&4 STEER
WHEEL #5&6 FIXED
WHEEL #7&8 STEER

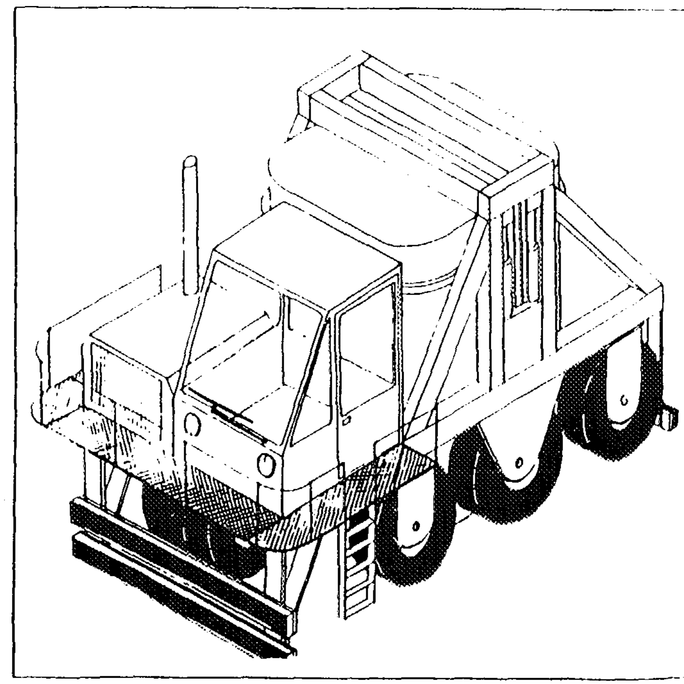
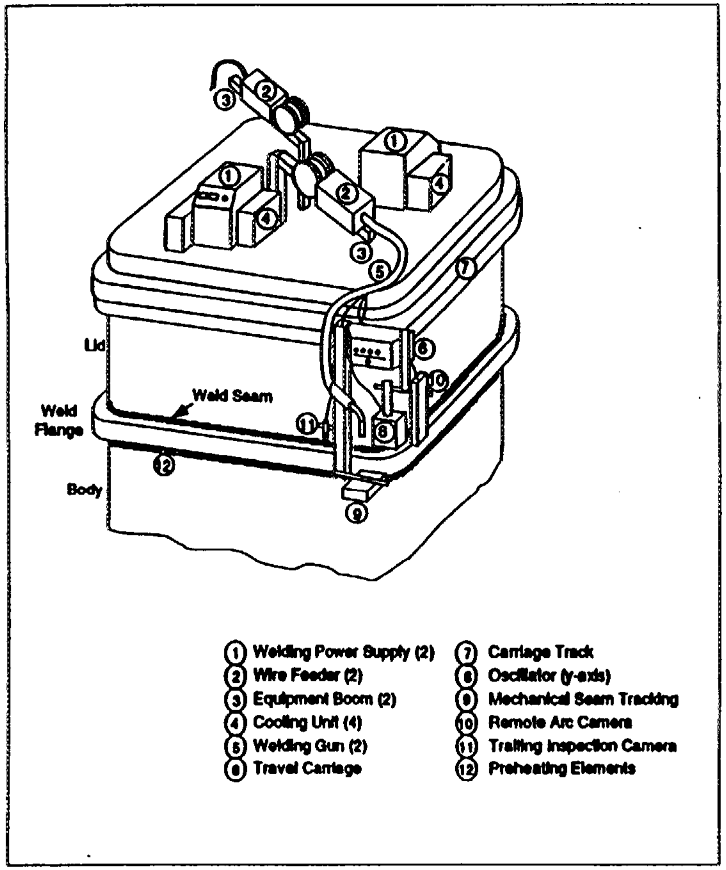
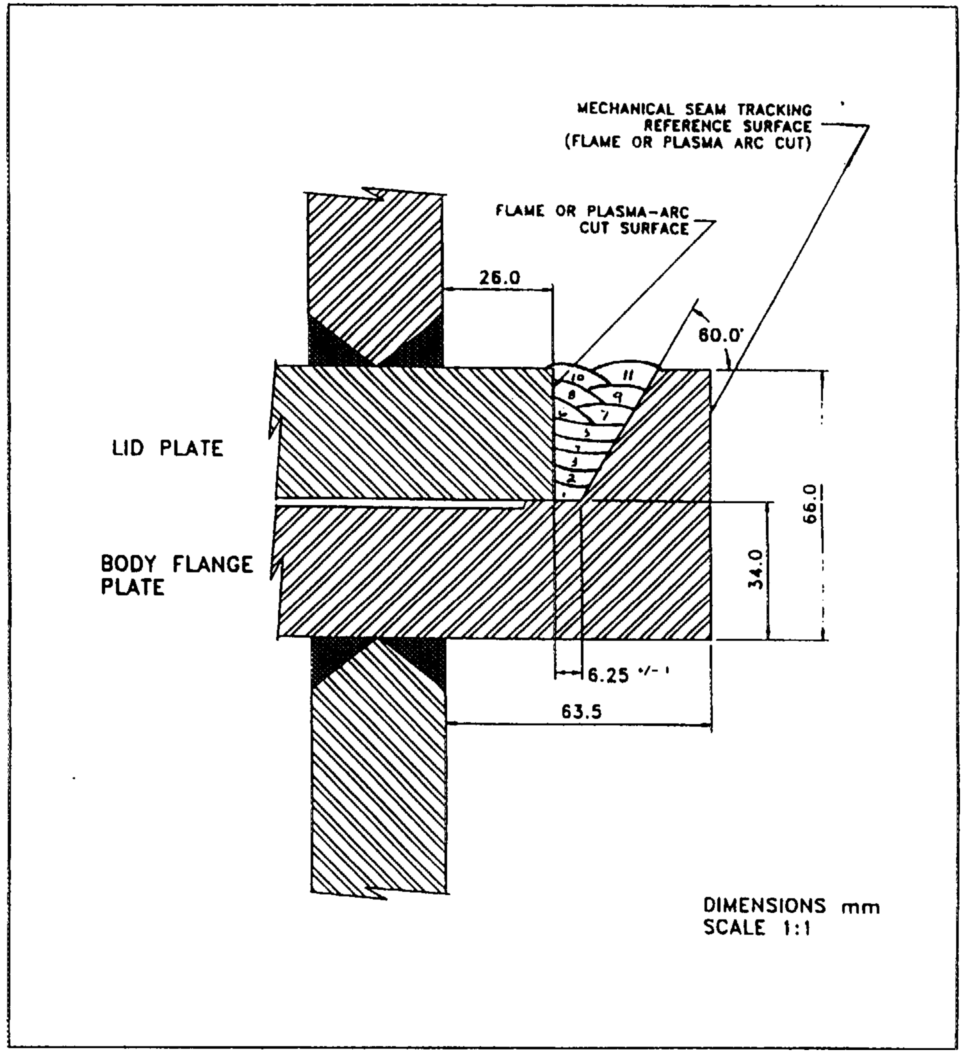


FIGURE 3
On-Site Transport Vehicle

UFDSF DSC LID TO BODY REMOTE WELDING



LID REMOTE WELD ILLUSTRATION



DSC LID-TO-BODY WELD JOINT

FIGURE 4
DSC Lid to Body Remote Welding

UFDSF DSC HELIUM LEAK DETECTION (BELL JAR)

A) He LEAK DETECTOR

B) VAC. PUMP (He DETECTION VAC.)

C) VAC. PUMP (He DETECTION VAC.)

D) VAC. PUMP (MAIN BELL JAR VAC.)

E) VAC. PUMP (MAIN BELL JAR VAC.)

F) VAC. PUMP (CAL.LEAK VAC.)

G) VACUUM GAUGE

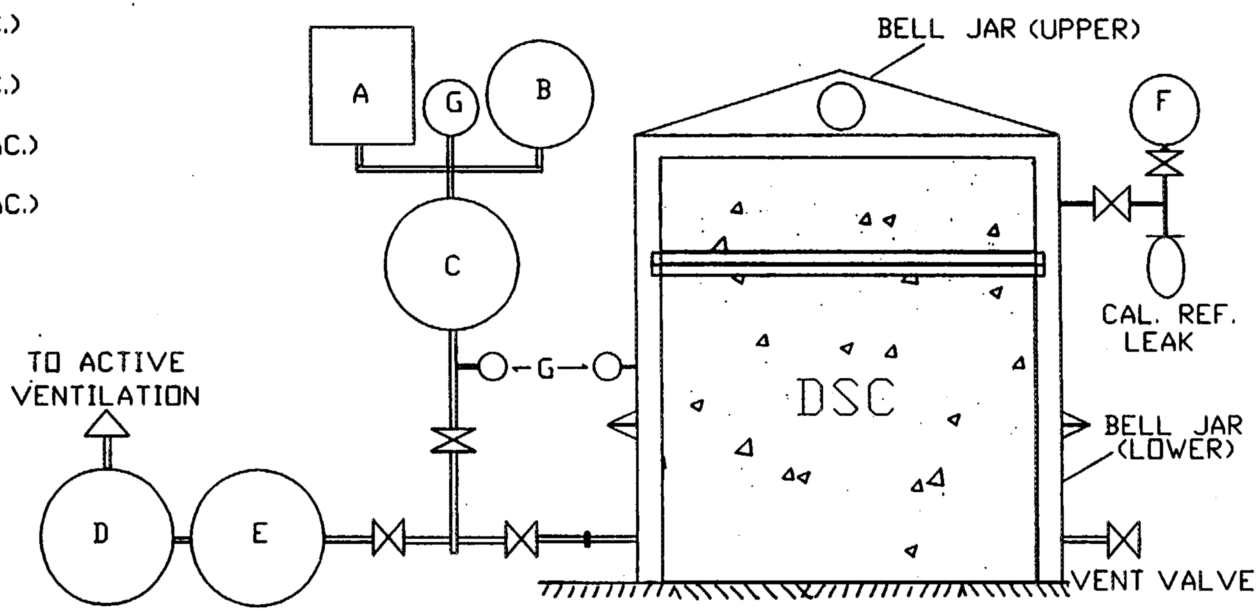


FIGURE 5
DSC Helium Leak Detection

BRUCE USED FUEL DRY STORAGE PROJECT
EVOLUTION FROM PICKERING TO BRUCE

by

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Problem Statement

By 1999, the Bruce Generating Station A will have filled all of its current wet pool used fuel storage and by 2003, the Bruce Generating Station B will have filled all of its current wet pool used fuel storage. Additional used fuel storage space is required to allow these two stations to operate beyond these two dates.

The recommended option was to utilize a dry storage container (BDSC 600) similar to that used at the Pickering Nuclear Generating Station. The changes made to the PNGS container included:

- fuel stored in trays
- container capacity increased to 600 fuel bundles
- changed the container lid to a metal lid
- changed the single concrete lid to a double metal lid system
- made the container non transportable
- the container would be dry loaded

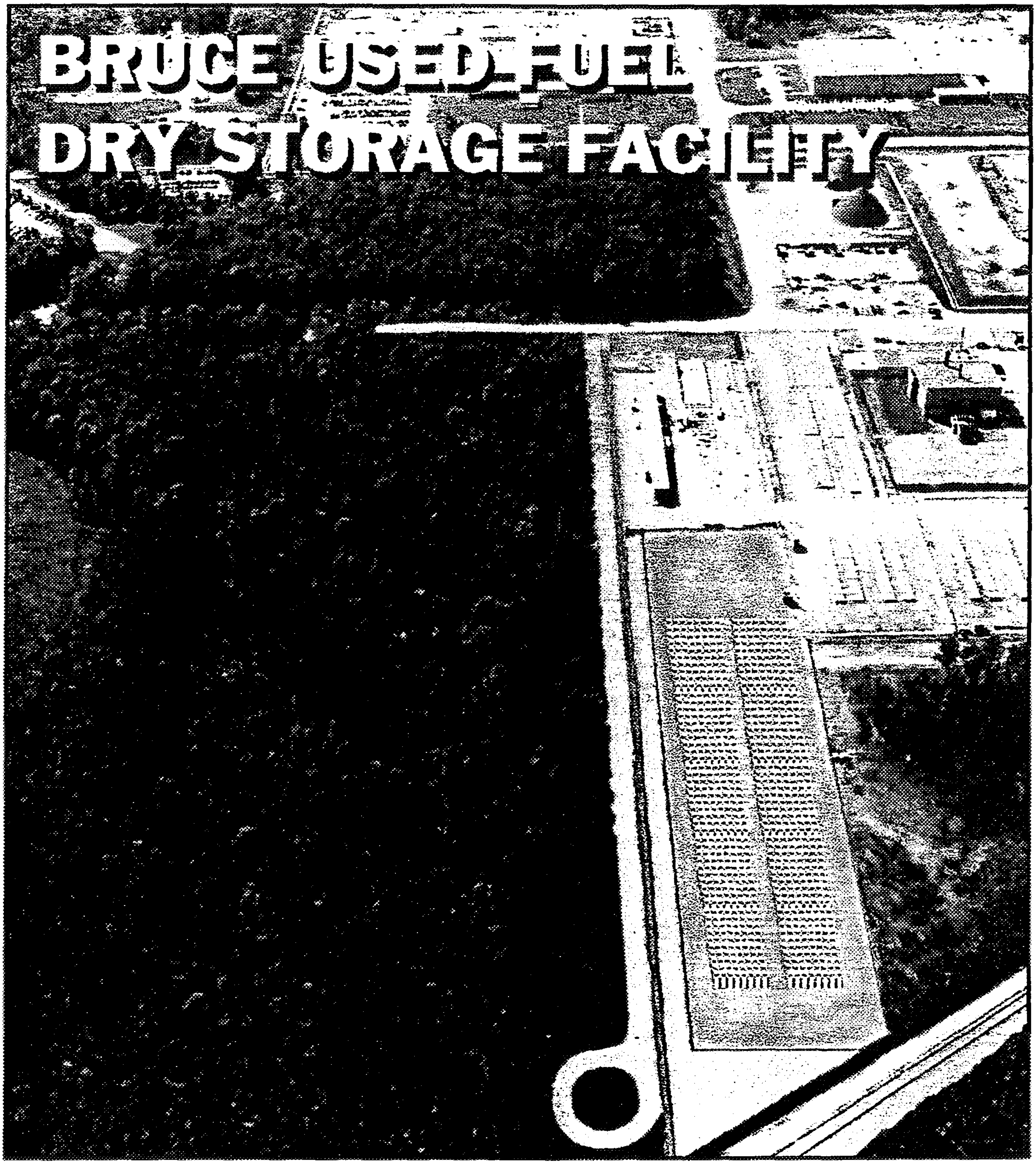
Project Description

The BDSC 600 dry storage system is an evolutionary development of Pickering Dry Storage Container (DSC) system design but is not licensable for transportation off-site under current regulations. The Bruce container is a dry loaded concrete cask similar in size to the Pickering DSC. However, it is much simpler to load, seal and test. The primary benefit of this alternative is that the fuel remains in trays for the duration of the storage period. Handling efficiencies accrue with this alternative because trays are loaded directly into the container. A dry load method has been developed for the Bruce Site, thus eliminating the need for fuel bay floor reinforcement, impact pads and crane capacity upgrading, with an associated reduction in costs.

The BDSC 600 will stand about 13 ft. tall and have a foot print of 8 ft. x 7 ft., weighing 76 tons empty and 93 tons fully loaded. The walls are 21 inch thick concrete with a ½ inch steel inner liner and outer shell. The cavity is of rectangular shape with similar dimensions to the quadrants of the stacking frames in the SIFB. All exterior surfaces are painted with 3 layers of high solids epoxy paint to reduce the possibility of contamination during loading.

The double lid arrangement of the BDSC is also different from the Pickering single lid "top hat" design. The lids are an all-steel plug type design and the closure welds will be situated on top of the container in a circular pattern. The primary lid consists of a 9" thick circular steel plate rough machined around the edge and fine machined near the weld joint at the top edge. A small vent is provided in the lid to aid vacuum drying. The primary lid is welded to the container body with a ½ inch bevel weld reinforced with a ¼ inch fillet weld. The 1" thick steel secondary lid is placed on the container body after the primary lid has been welded in place. This secondary lid is also circular and is welded to the body with a ½ inch fillet weld. The second lid facilitates leak testing of the primary containment weld and provides redundant containment. This lid has a single penetration to aid leak testing. The container is backfilled with inert helium cover gas after the fuel is loaded.

This design offers many advantages over the Pickering DSC system design. The construction of the container is simpler and this fact is reflected in a 28% reduction in the life cycle cost over a Pickering repeat. The circular lids are easy to machine simplifying fit-up with no concrete pouring, thereby reducing welding, testing, and machining cost as well as eliminating rebar and concrete itself. Using steel also reduces cask height and hence a slight weight reduction. The circular lid also permits lifting provisions to be placed on top of the container, reducing the lifting bean span and weight, as well as simplifying the connection into the container body. Another major advantage of this lid design is the simplification of the closure welding machine. The machine will simply rotate around a centre point laying in weld material from above the joint in two or three passes. The weld equipment will most likely utilize a MIG process similar to the Pickering system, however, the machine will only put down two or three passes, instead of the 11 passes required at Pickering. No preheat will be required and the total weld time (including set-up, inspection and final cleaning) could be less than one shift, compared to the Pickering 72 hour system.


BRUCE USED FUEL
DRY STORAGE FACILITY

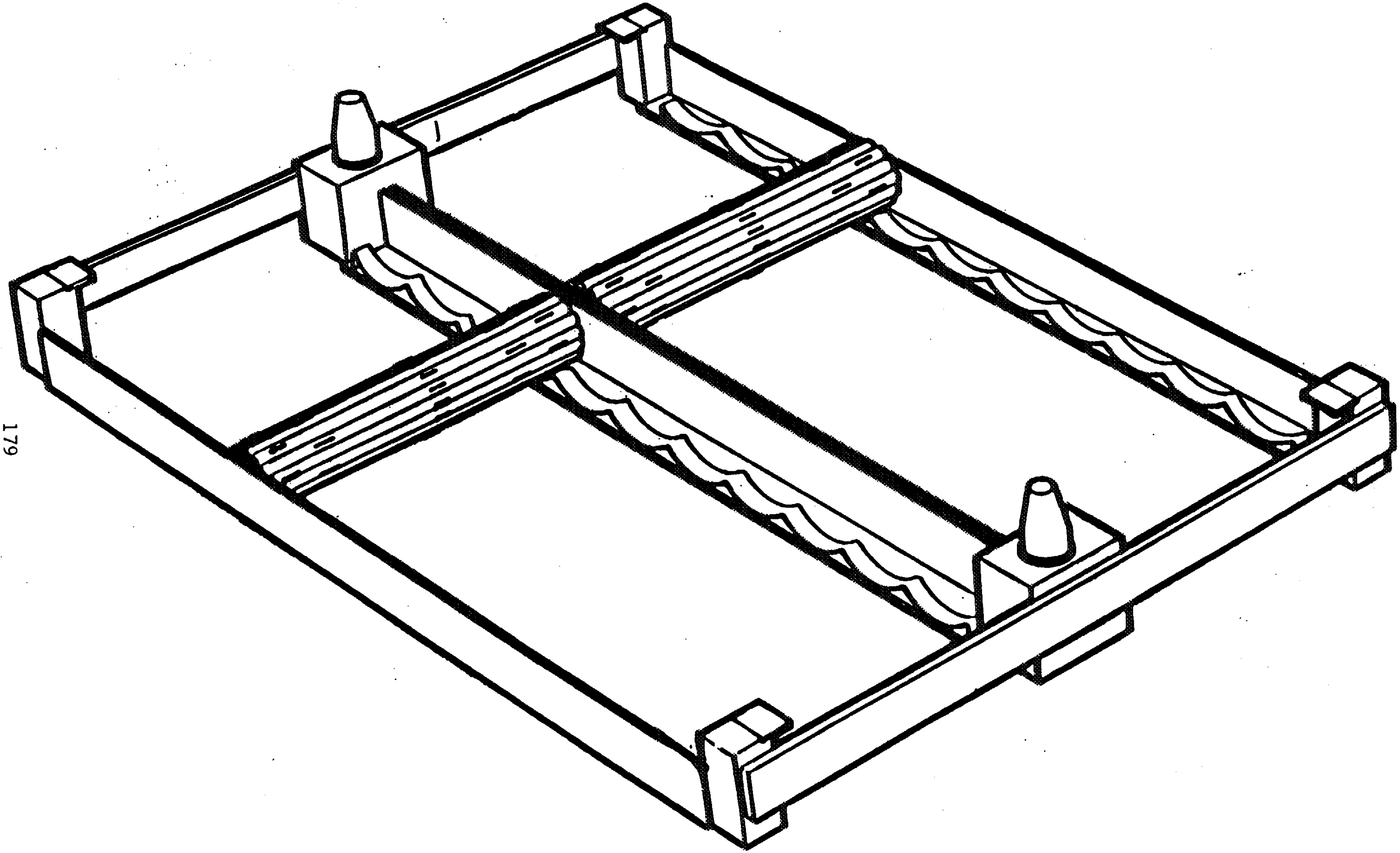
Pickering and Bruce DSC Comparison

PICKERING

- 384 bundles
- 140" high
- 62 tons
- Transportable off site
- Modules
- Slab type square lid
- Wet fuel load
- 11 pass perimeter V-groove (1 1/2")
- ASME NC
- Indoor storage

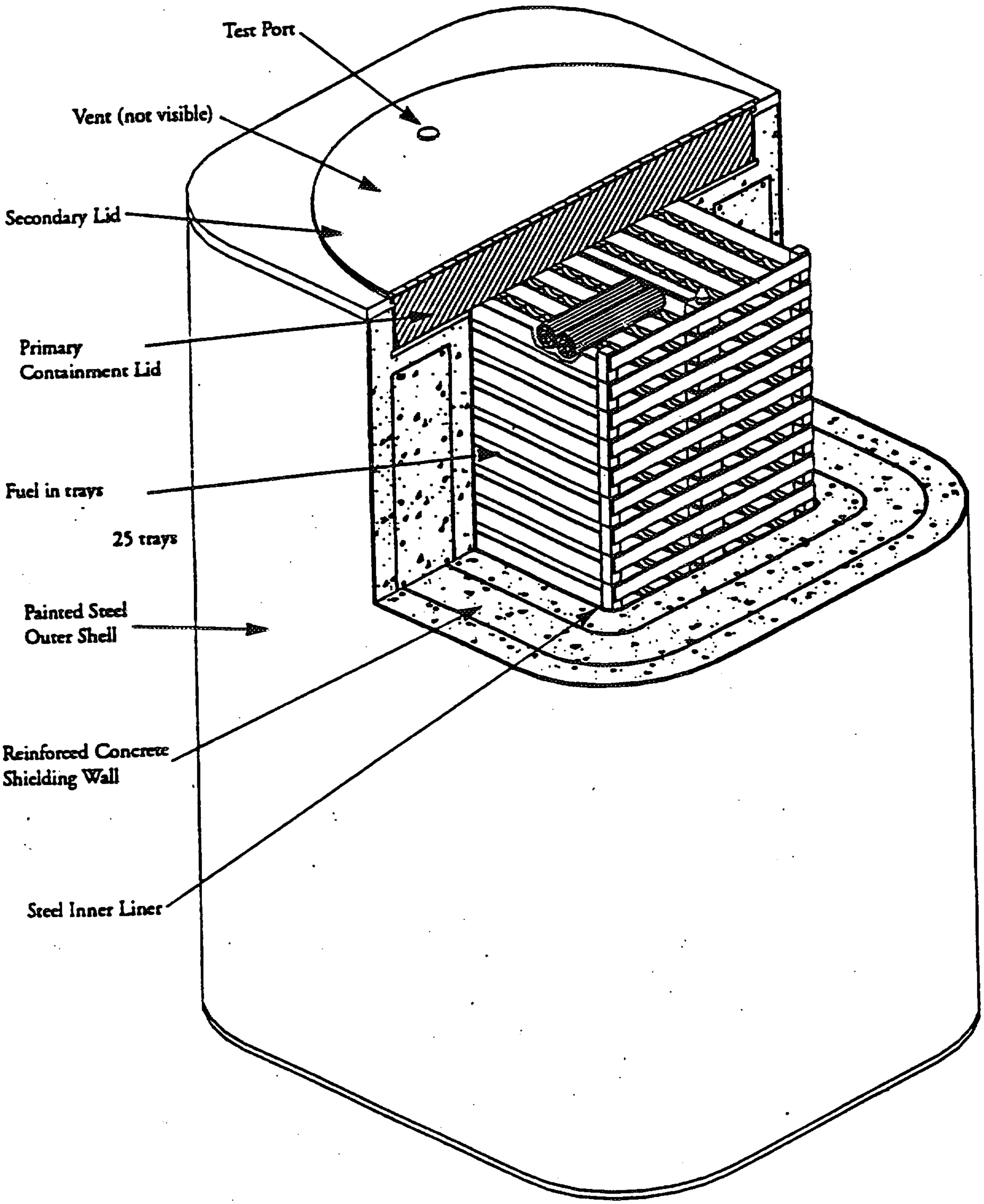
BNPD

- 600 bundles
- 159.5" high
- 93 tons
- Non-transportable off site
- Trays
- Circular plug lid
- Dry fuel load
- 4 pass in top circular V-groove (3/4")
- ASME NE
- Outdoor storage

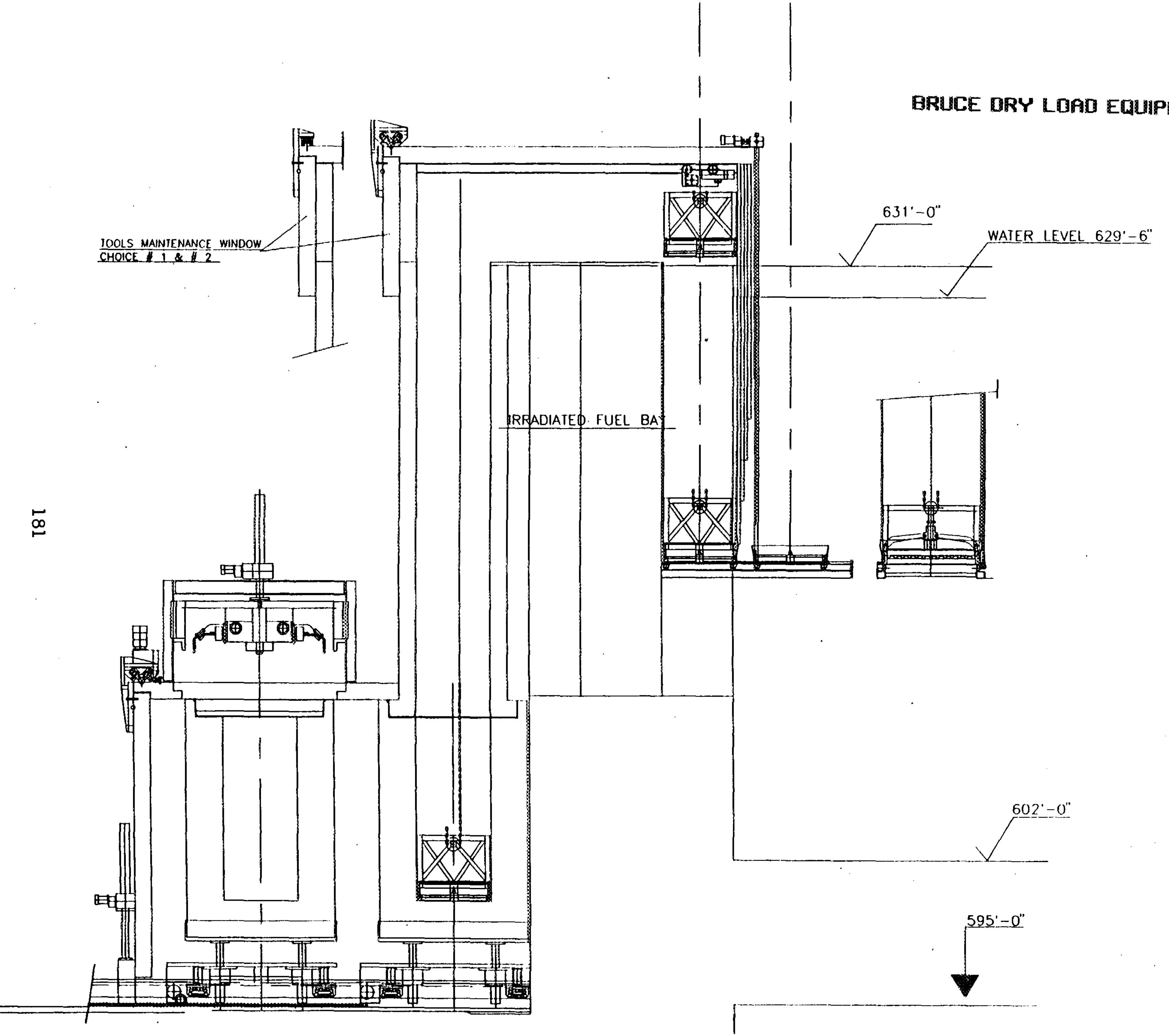


BRUCE GS
IRRADIATED FUEL STORAGE TRAY

Bruce Dry Storage Container - 600
BDSC 600

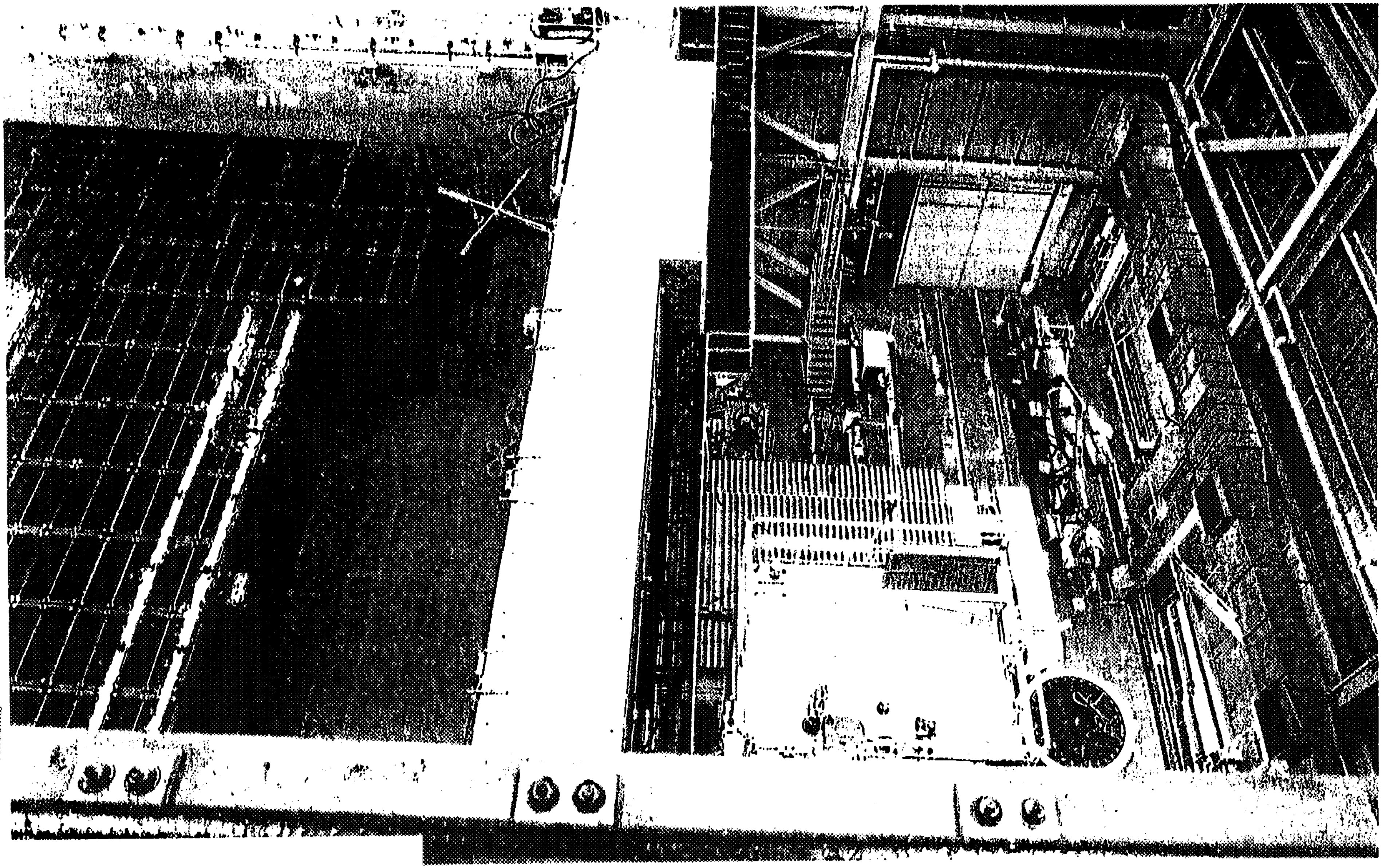


BRUCE DRY LOAD EQUIPMENT



Dry Load Summary

- Independent of other station systems
- Tested before it is brought into the Irradiated Fuel Bay
- No component is greater than the 35 ton capacity of the existing crane
- Local control centre will be independent of other station systems



BRUCE SECONDARY IRRADIATED
FUEL BAY

PND FUEL HANDLING DECONTAMINATION: FACILITIES AND TECHNIQUES

by
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CA9700738

The use of various decontamination techniques and equipment has become a critical part of Fuel Handling maintenance work at Ontario Hydro's Pickering Nuclear Division. This paper presents an overview of the set up and techniques used for decontamination in the PND Fuel Handling Maintenance Facility and the effectiveness of each.

1.0 INTRODUCTION

The current PND Fuel Handling Maintenance Facility (FM Shop) was constructed in the late 1970's as part of the Pickering 'B' expansion that increased the size of the Pickering station from four to eight units. It is located in the Unit 0 operating island just west of Unit 5.

Despite various other improvements, FM Shop personnel still experienced the same problem with inadequate decontamination equipment that existed in the original Pickering A Fuel Handling Maintenance Shop. Decontamination was still limited to manual scrubbing and dry wiping for smaller items like closure plugs, and manual scraping and vacuuming for larger items like FM rams and head internals.

A renewed effort to upgrade Fuel Handling decontamination capabilities was initiated in December 1987, after excessive levels of Co-60 were identified in the station laundry and traced back to protective clothing from the FM Shop. This effort focused on two key result areas: (1) eliminating the existing contamination hazard, and (2) developing more effective decontamination capability.

The FM Shop was thoroughly decontaminated over a four month period in order to eliminate the loose contamination hazard.

An assessment of the facility was conducted with the assistance of personnel from what is now Ontario Hydro Nuclear Technology Services (NTS) in order to facilitate development of better decontamination equipment.

1.1 Performance Criteria

A set of performance criteria was developed from this assessment and used as a general guideline for improving FH decontamination capability. They are as follows:

(a) The cleaning activities carried out in the decontamination centres must reduce loose activity to less than 1,000 cpm. Fixed activity to be dealt with by disposal.

(b) The decontamination equipment should be selected to minimize the amount of "hands on" work in order to minimize personnel radiation dose during decontamination.

(c) Decontamination equipment must have sufficient capacity to meet the anticipated work load of the FM shop in a timely manner.

(d) The layout for the FH Decontamination Facility must satisfy the following.

(i) A managed flow of personnel and equipment through the decontamination centre. This is best achieved by one-way materials flow.

(ii) Contaminated and clean equipment must be kept separate to prevent cross contamination.

(iii) Ease of inventory control, so that parts brought in for decontamination are easily retrieved and processed.

(iv) Sufficient space to work and manoeuvre materials along the transfer flow paths.

(v) Sufficient storage space for contaminated equipment awaiting decontamination. The location of contaminated equipment storage must not pose a radiation hazard to any staff working in the shop or station.

(e) Radiation monitoring of equipment and personnel entering and leaving the decontamination centre is required. Appropriate monitors and meters must be in place and operational.

(f) Procedures for facility operation, housekeeping, equipment use and maintenance, waste handling, and radiation control must be available.

(g) Proper management of the Shop is essential to ensure on-going successful decontamination facility operation. Periodic assessments of the facility operation should be carried out to assist in identifying deficiencies and required corrective action.

2.0 DECONTAMINATION TECHNIQUES

Many techniques can be used to clean contaminated items. Some of the methods looked at for use in the FM Shop are described briefly below.

2.1 Manual /Mechanical

Manual/mechanical techniques rely on physical contact between the surface of a cleaning tool, such as a metal wire brush or scouring pad, and the surface to be cleaned.

The process can be quite effective at removing loose and some fixed contamination from the surface of fuelling machine components and tooling. The degree of contamination removal depends on the aggressiveness of the mechanical method selected as well as the application technique. An aggressive manual application of a mechanical cleaning technique can be highly effective at removing fixed contaminants. However, it can also remove the upper layers from the surface of the contaminated object which is usually not desirable unless the part is to be scrapped. Less aggressive mechanical methods may not result in material loss, but may also not be effective at removing more adherent deposits.

2.2 Chemical

Chemical cleaning relies on the use of a solvent solution to dissolve the radioactive deposit or contaminant. Chemical cleaning processes are usually classified as either dilute or concentrated. In general, dilute chemical cleaning refers to processes using solutions with concentrations $<1\%$ (w/w); whereas concentrated chemical cleaning typically refers to processes that use solutions with concentrations $>1\%$ (w/w).

Chemical cleaning effectiveness is dependent on a number of factors: concentration and composition of the cleaning solution, application method (temperature, pH, flow rates, contact time), and composition of the deposit or contaminant are key among these.

2.3 High Pressure Water

High pressure water jets with discharge pressures ranging from 1 to 15 kpsi have been used to clean a variety of contaminated equipment surfaces and tools.

The decontamination effectiveness of the high pressure water stream depends on the shape, flow rate, and discharge pressure of the jet and stand off distance.

Water jets employed at the lower end of the pressure range (~2 to 3 kpsi) are effective at removing some loose contamination and will generally not damage protective coatings or painted surfaces beneath the deposit. A typical system operating in this pressure range will use a "fan" type spray nozzle (30° or 45° angle of dispersal on the spray), have discharge water flow rates of 3 to 6 USgpm, and be manually operated.

Water jets employed at the higher end of the pressure range (~9 to 15 kpsi) are effective at removing all accessible loose contamination and some fixed contamination. (For the purposes of this discussion, "fixed" refers to any contaminant that cannot be removed by manually wiping with a soft cloth.) Damage to underlying protective coatings and painted surfaces can result if care is not taken when decontaminating with water jets in this pressure range. Typically, a system operating in this pressure range will employ fan or pencil type spray nozzles (jet can be angled or straight), have discharge water flow rates of 6 to 15 USgpm, and be either automated or manually operated.

2.4 Vibratory Finishing

Vibratory finishing uses small metal or ceramic cleaning media immersed in a cleaning solution to decon small metallic components and hand tools. During cleaning using this method, contaminated component, cleaning media and solution are all held in a large tub with a system to continuously filter and recirculate the cleaning solution. The tub vibrates on its spring mounted base, creating physical contact between the cleaning media and component in the tub. A thorough inspection should also be performed after cleaning in the VFM to ensure that none of the ball cones have become lodged within the part cleaned.

The cleaning action obtained using this method is a combination of the mechanical effect created by contact between the metal cleaning media and component/ contaminant surface, and the chemical action of the cleaning solution. Each process enhances the action of the other.

Vibratory finishing is effective at removing loose and some fixed contamination from the surfaces of small fuel handling components and hand tools.

2.5 Ultrasonic Cleaning

Ultrasonic cleaning combines the chemical action of a cleaning solution with the physical cleaning action of cavitation to remove deposits/ contaminants from component/ tool surfaces. The basic system consists of a tank (which contains the cleaning solution), a heater, and an ultrasonic generator and transducer assembly. A recirculating clean-up system with in-line filter is also sometimes employed.

The effectiveness of ultrasonic cleaning on any given deposit or contaminant is dependent on the cleaning solution used, and method of application. In general, ultrasonic cleaning enhances the effectiveness of chemical cleaning solutions by producing a physical effect (cavitation) on or near the contaminated surface. This is due to the multitude of bubbles that are produced in the cleaning solution as a result of vibrations generated by the transducer assembly. These bubbles are unstable, localized low vapour pressure regions and collapse quickly. The cavitated "collapsed bubble" region produced is a temporary, localized pocket of high temperature and pressure that can enhance the chemical cleaning properties of the liquid solution by up to 10 times, according to results reported by EPRI.

(f) Strippable Coatings

Strippable coatings are used for cleaning/ contamination control in two ways: [1] as a protective coating (to prevent contamination of the material underneath) which can be easily removed when loaded with deposits, or [2] as decon technique to remove contaminants already present on a surface. Typically, a strippable coating will be applied as one thick layer (0.020-0.080 in.) directly to the clean or contaminated equipment surface. When applied as a protectant, the strippable coating may be left in place for up to a few days before removal. When applied to decon a surface, the strippable coating is removed as soon as it has dried or cured sufficiently to obtain maximum strength.

The effectiveness of strippable coatings is quite good in general. Decontamination factors as high as 1000 have been reported by EPRI using this method.

(g) Dry Ice Blasting

Dry (CO_2) ice blasting is a decontamination method that relies on physical and thermal effects to remove deposits from contaminated surfaces. Systems on the market today typically use compressed pellets of CO_2 "snow" in the cleaning process. The pellets are entrained in an air stream that is directed via a jetting tool to the contaminated surface to be cleaned. The cleaning mechanism is dependent on several factors: [1] the impact energy of the pellet, which is a product of the pellet's mass/density and velocity; [2] pellet mass flow rate/ number of impacts per unit time; and, [3] thermodynamic effects.

Factors [1] and [2] are controllable by making adjustments to the blasting machine. Increasing any one of these factors (pellet density, velocity, or pellet flow rate) will increase the effectiveness/ aggressiveness of the decontamination.

Factor [3], thermodynamic effects, is caused by the temperature differential between the dry ice pellets and the contaminated surface being impacted. An effect known as "fracking" (freeze cracking), in which the contaminant layer becomes embrittled by the cold CO_2 is one such effect. Fracking results in a break up of the active deposit layer, making it easier to dislodge.

Another thermodynamic effect occurs when the CO_2 pellet sublimates on impact. The CO_2 gas from the sublimating pellet is able to penetrate beneath the deposit layer (which has been made accessible by "fracking") and dislodge contaminants from underneath as it expands.

Dry ice blasting can be effective at removing most loose contaminants, oil and grease, and some paints and adhesives. It is not effective at removing oxides, scales/mineral or other hard deposits.

3.0 DECONTAMINATION EQUIPMENT SELECTION

Field testing indicated that high pressure water jetting was effective at removing most of the contamination normally encountered in the course of Fuel Handling maintenance. This contamination consists mainly of activated corrosion and some fission products picked up from the primary heat transport system during fuelling.

In addition to being effective, high pressure jetting also offered the advantages of being non-corrosive, and being relatively inexpensive to start up and operate. Waste disposal was also easy since no chemicals, abrasives, or other additives are ordinarily needed in order to achieve good results.

Decontamination using manually operated 10 kpsi jetting equipment commenced in 1989. Very careful control of the water jetting operation was maintained during this time in order to ensure worker safety as well as to obtain good decontamination factors.

The FM ram was the most frequently decontaminated item using this method over the next 18 months. After several repetitive manual cleans of the ram using FM Shop personnel in plastic suits, it became apparent that this physically tiring, somewhat tedious task would be ideally suited to automation. This observation eventually led to the development of the automated Ram Spray Booth (RSB), a ventilated, totally enclosed automated water spray booth.

4.0 Ram Spray Booth

The RSB was custom designed and constructed to decontaminate PND fuelling machine rams, and can accommodate other long cylindrical items with diameters < 17".

Some key design features incorporated into the RSB are as follows:

- accommodates an FM ram assembly with the drive housing gray lock attached
- use of overhead crane to load /unload individual components
- high accessibility to the ram when inside the booth
- all high pressure spray, drain, and filter systems located within the booth
- built to facilitate decon of interior surfaces and minimize potential for crud traps
- able to use robotic arm to perform the majority of the ram housing and ball screw decon with rotary water jets
- capability to manually water jet areas not accessible by the robotic spray arm or as backup
- built in safety devices to limit access to the booth interior when robotic spray arm in operation
- PLC access codes to restrict use to authorized personnel
- connectable to active ventilation

The height of the booth (92" including foot pads), required construction of a raised platform around the booth to provide operators with easy access to controls and glove ports.

The Automated Water Lancing Machine (AWLM) uses a two nozzle rotary spray jetting tool (~9kpsi) for cleaning. Separate electric motors inside the booth move the spray arm longitudinally and radially along separate tracks to provide spray coverage to most areas of the ram. An Allen Bradley PLC directs both motors to control the spraying pattern when in use.

A separate, smaller 3 kpsi water jetting pump is also built into the RSB to permit manual jetting of any areas missed. The hand-held spray lances are operated through glove ports down the length of both sides of the spray booth.

The Ram Spray Booth was placed in-service in mid-1994 at a total cost of approximately $300,000.

Decontamination results achieved using the RSB have been very good. Rams with smearable activity > 1 million cpm have been reduced to virtually zero loose. Further, time required for a full ram decontamination has been lowered from 7-10 days using manual spraying to about 3 days using the RSB.

4.1 Ram Spray Booth Modifications

A number of modifications were carried out during the RSB commissioning process to address deficiencies and enhance performance.

4.1.1 L-225 High Pressure Pump Installation

The existing FM Shop high pressure jetting pump was modified to allow it to be installed beneath the Ram Spray Booth for use as part of the Automated Water Lancing Machine, resulting in savings on equipment costs when purchasing the RSB.

4.1.2 Light Curtain

Difficulties were encountered setting up the electric light curtain. The light curtain is a device consisting of an emitter, a receiver, and two mirrors. As designed, this device was supposed to stop the AWLM in the event that one of the operator's arms was extended into the booth while in operation. Although the unit did eventually function as designed, it proved to be exceedingly difficult and time consuming to maintain the proper set up.

This was due to the fact that the light curtain model selected by the manufacturer is at the limit of its range when used as a perimeter detector for the RSB. Slight jarring of the mirrors from personnel walking by was enough to set them out of alignment. The corrective measure eventually used to resolve this problem consisted of disconnecting the light curtain system, and using shield doors over the glove ports to prevent access while the AWLM is in operation. Safety interlocks, which trip the Emergency Stop instantaneously when activated, are in place on all RSB doors and hatches to prevent all other access routes to the booth interior while the AWLM is in operation.

4.1.3 Automated Water Lancing Machine Programming

Customization of the PLC software for the AWLM was carried out in the FM Shop between April and July 1994. The number of passes and the spray pattern of the robotic arm were modified in order to maximize coverage during FM ram decontamination. The spray pattern was modified so that washing occurred from the top down (i.e. the first pass at 90^{o} to horizontal, then at increments on either side of top centre down to horizontal) as opposed to the original pattern which had sprayed from 0^{o} to 180^{o} (i.e. from horizontal to the top then back to horizontal).

4.1.4 Tooling and Accessories

Ram trolley cart modifications and numerous other mechanical changes were made to the Ram Spray Booth during commissioning to help achieve better performance. These changes included things such as:

- fabrication /installation of spray cradles and supports
- fabrication /installation of ballscrew / tube racks
- installation of safety supports (for side doors) in the RSB
- repairs to defective air valves (for side doors)
- repairs to drip trays (for side doors)
- addition of glove bar/ hanger
- repairs to the cable tray
- addition of an access port (2" dia.) to the north end loading hatch to enable water jetting of ram housing and tube internals without removing them from the RSB

5.0 Other Decontamination Equipment

Other equipment/techniques that had proven to be effective --such as ultrasonic cleaning and vibratory finishing-- were also introduced into the FM Shop at around the same time as water jetting. Figures 1 and 2 show the layout of the renovated facility.

6.0 Decontamination Equipment Effectiveness

Effectiveness of the equipment installed was assessed as part of the commissioning process. A summary is presented in Table 1 below. Regular checks since that time have shown that the equipment is still meeting performance targets (i.e. able to decontaminate components to $<$ 1000 cpm loose).

7.0 CONCLUSION

Improvements in productivity and equipment reliability have been achieved as a result of upgrading Fuel Handling decontamination capability.

Ram performance, for example, has continued to improve as decon capability has increased. This is at least partly attributable to better rebuilds during maintenance as a result of [1] being able to fully see the part, and [2] being able to spend more "quality" time working on the part.

There are also other indirect benefits of decontamination. The maintenance side of the FM Shop, for example, is no longer a rubber area, which makes for more comfortable working conditions. As well, dose pickup has dropped by a factor of 5 since the commencement of the program, an important health and safety gain.

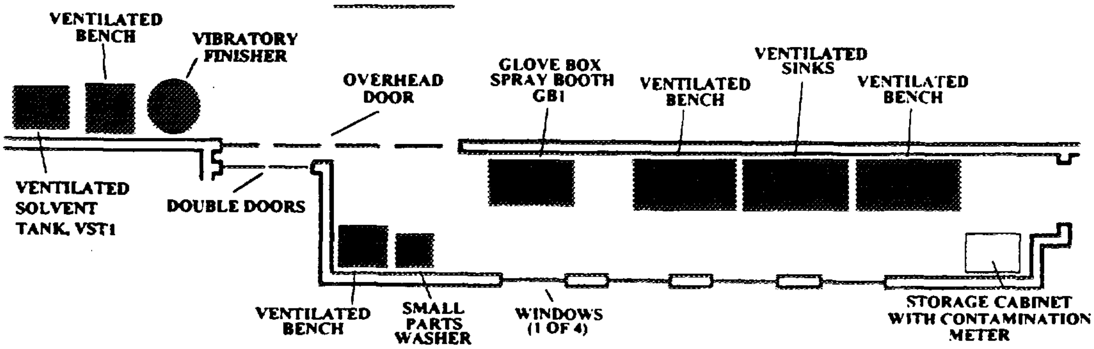
Table 1: Decontamination Equipment Effectiveness

DESCRIPTION	USED FOR	EFFECTIVENESS# (cpm)
Ram Spray Booth	rams, other cylindrical parts	>1,000,000 to<10,000
Vibratory Finishing Machine	closure plugs, small parts	>25,000 to<2,500
Glove Box Spray Booth	small parts (< 18 kg)	> 100,000 to<2,500
Ventilated Sinks/Benches	small parts, hand tools	>10,000 to<2,500
Automated Parts Washer	small parts, hand tools	>55,000 to<30,000##
Ultrasonic Cleaner	small parts, hand tools	>25,000 to<2,500
Ventilated Solvent Tank	oil, grease removal	fully degreased

Maximum smearable activity levels detected before and after decontamination. Total DFs for each piece of equipment will vary depending on the nature of the contamination on the component being cleaned. Most components can usually be decontaminated to zero loose activity levels.

Measurements after one wash cycle. Most components can usually be decontaminated to <1,000 cpm with repeated washing (two to three additional cycles).

FIGURE 1: SMALL PARTS DECON ROOM LAYOUT



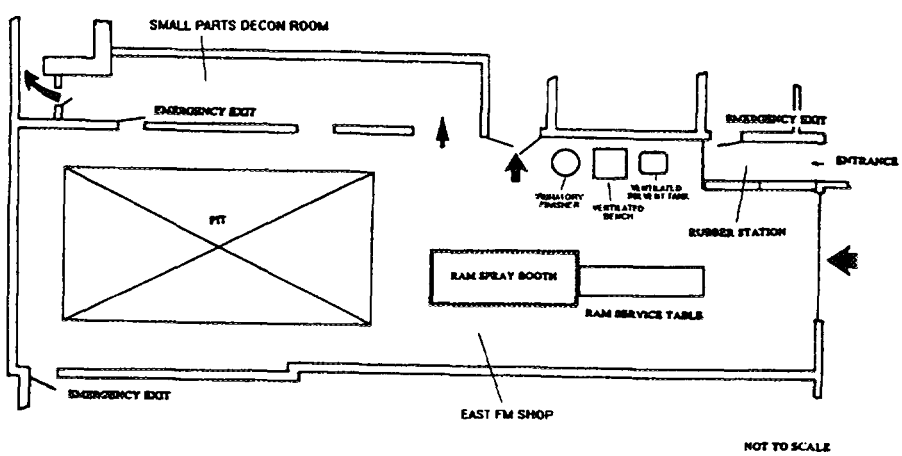


FIGURE 2: EAST FM SHOP EQUIPMENT LAYOUT

FIGURE 3:
RAM SPRAY BOOTH DETAILS

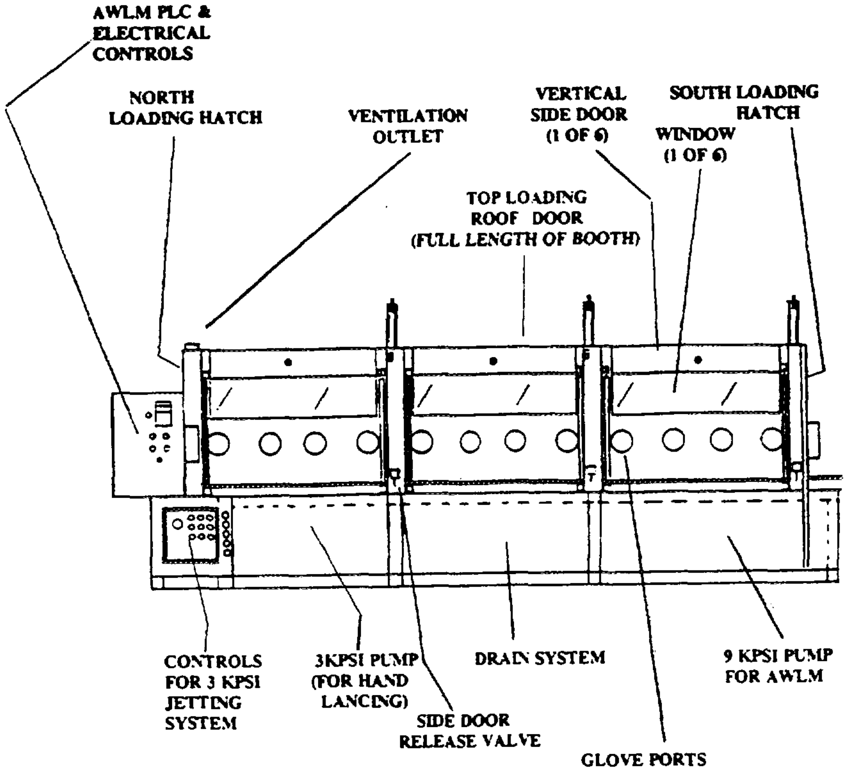
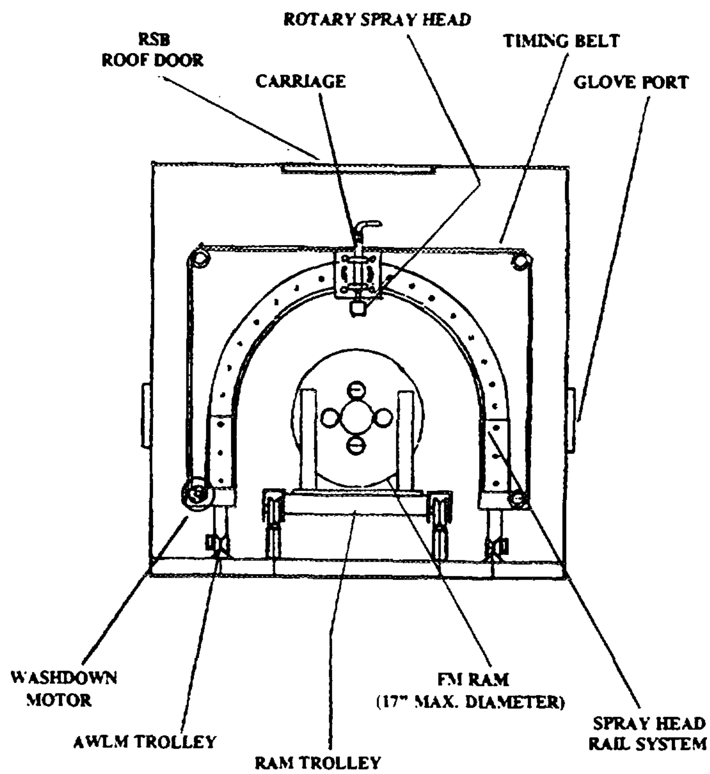


FIGURE 4:

AUTOMATED WATER LANCING MACHINE DETAILS

(END VIEW)



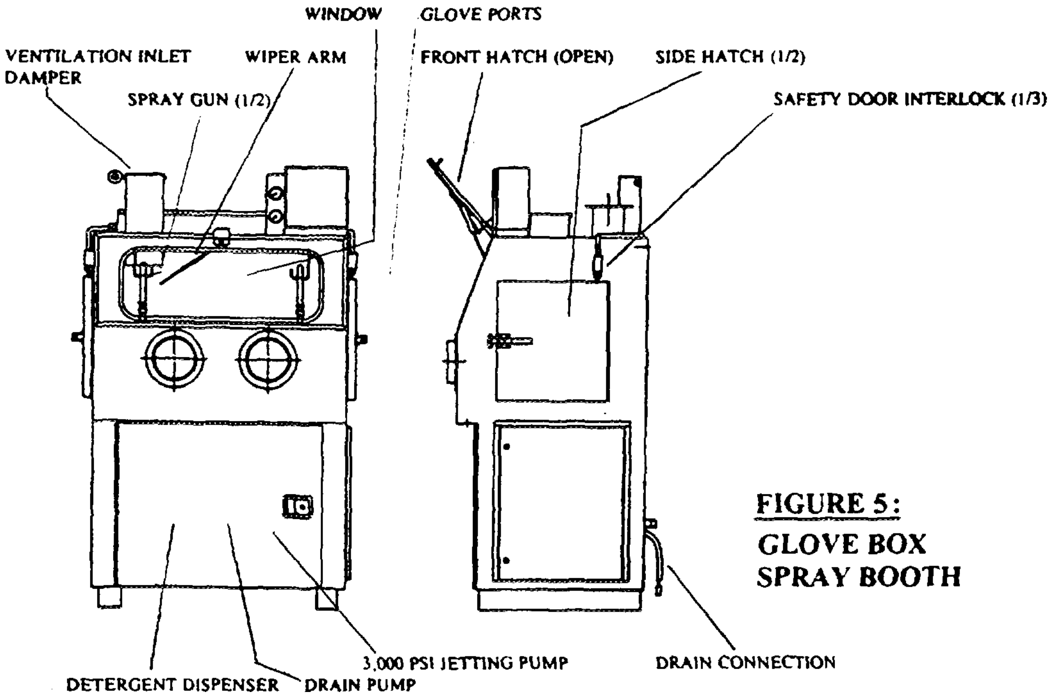


FIGURE 5:
GLOVE BOX SPRAY BOOTH

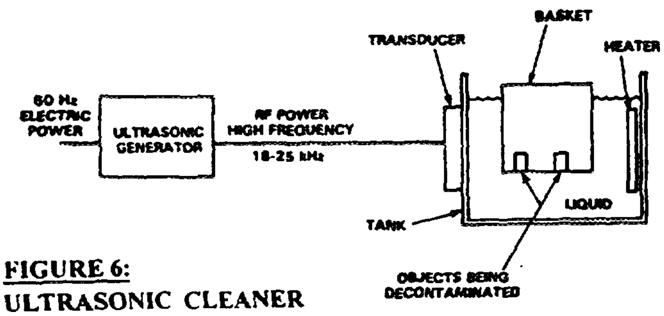


FIGURE 6:
ULTRASONIC CLEANER DETAILS

FIGURE 7: VENTILATED SOLVENT TANK DETAILS

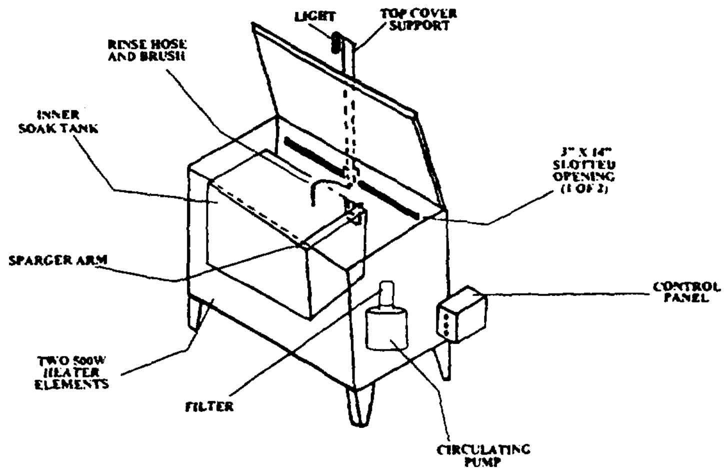


FIGURE 8:
VIBRATORY FINISHER DETAILS

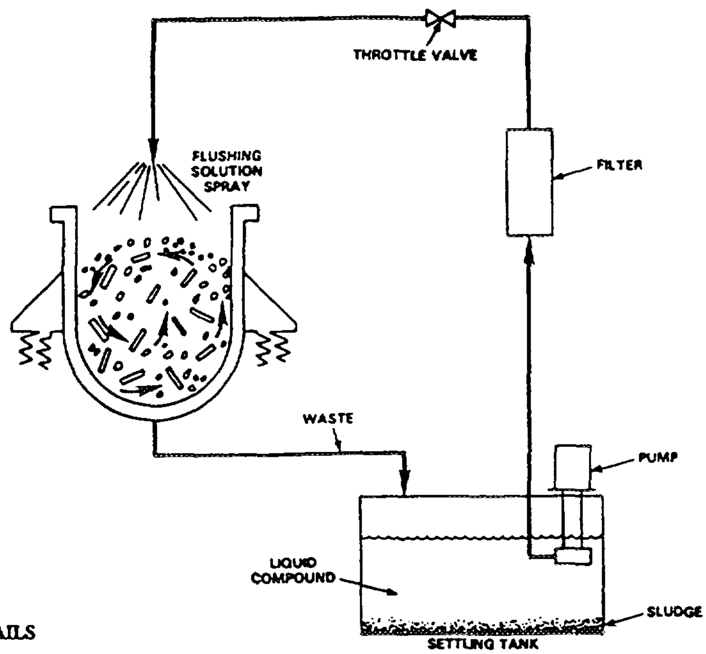


FIGURE 9-A:
VENTILATED BENCH DETAILS

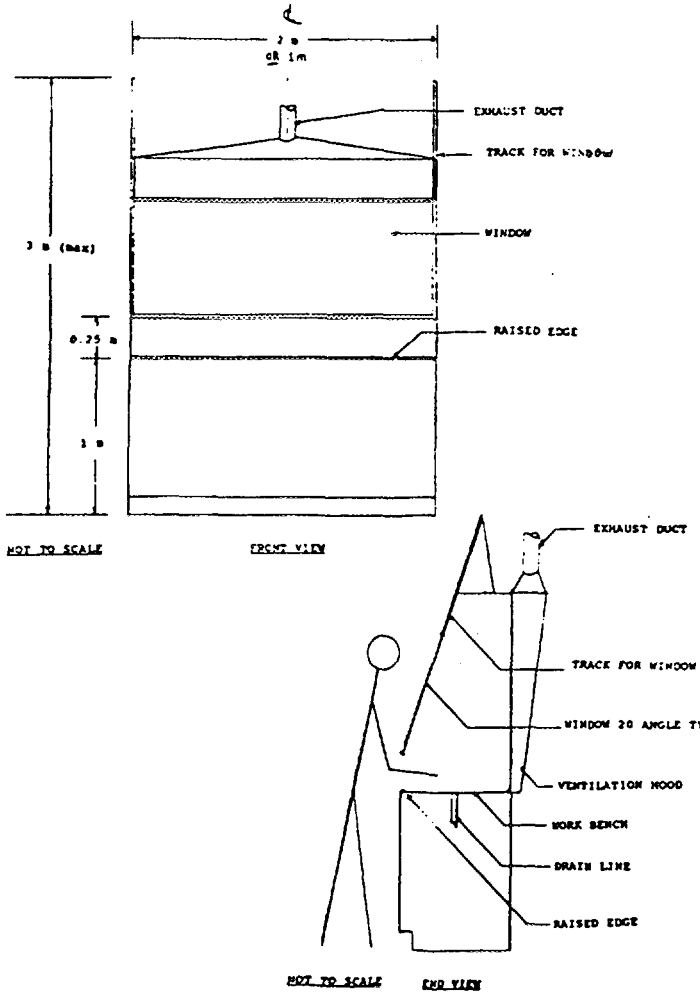
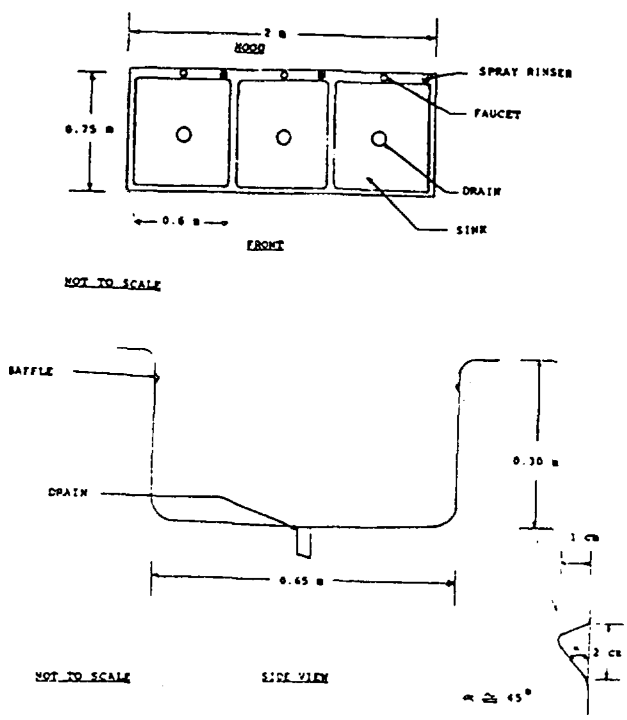


FIGURE 9-B:
VENTILATED SINK DETAILS



NOTE: All other details are identical to 2m ventilated bench units.

CA9700739

DARLINGTON NGD FUEL HANDLING

HEAD EIGHT ACCEPTANCE PROGRAM

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ABSTRACT

Darlington NGD requires eight fuelling machine heads to fuel the 4*932MW reactors. Six heads are used on the three fuelling machine trolleys for normal fuelling operations. A further two heads are required to allow for maintenance and to provide for such reactor face activities as PIPE and CIGAR.

Seven heads were successfully delivered to site from the head supplier. During acceptance testing, stalls on the charge tube screw assembly of the eighth and final head prevented its delivery to site. Replacement of the charge tube screw with a spare screw did not alleviate the problem.

An in depth series of tests were undertaken at site, at the supplier and at the screw sub-supplier to determine the root cause of the problem.

These tests included taking torque measurements under different operating conditions and using different components to assess the effects of the changes on torque levels. An assessment of the effects of changing chemical conditions (particularly crud levels) was also made.

To ensure that the results of the testing were well understood, additional torque testing was also completed on a head and screw assembly at site that was known to work well.

Based on all of the above series of tests, a recommendation was made to re-machine the charge tube screw(s). The original charge tube screw from Head eight was subsequently returned to the sub-supplier for re-work. Follow-up torque measurements and acceptance testing showed that the screw re-work was effective and that Head eight could be successfully delivered to site.

This paper focuses on the results of the head/screw test program. Results of the acceptance testing are also discussed.

INTRODUCTION

The Darlington fuel handling system provides automatic on power fuelling capability to all four reactors. Under normal operating conditions, each reactor unit can be serviced by any one of three fuelling machine (FM) systems.

A fuelling machine system consists of one pair of fuelling machine heads and their associated heavy water and air auxiliary systems all mounted on a transport trolley.

During a typical fuelling operation, one fuelling machine head inserts four fuel bundles into a channel two at a time. This causes the string of 13 bundles in the channel to be shifted and the four end bundles to enter the second fuelling machine head. Either of the fuelling machine heads can insert new fuel or receive irradiated fuel as all the heads are identical (see fig 1).

Each head consists, in part, of a charge tube and a concentric ram which transfers fuel into the reactor under normal operating conditions (310oC and 10MPa). The charge tube has an axial drive (CTA) and a rotary drive (CTR). The ram has an axial drive only.

Figure 2 depicts the layout of a Darlington ballnut/screw arrangement. The charge tube is hollow with the ram positioned inside the charge tube. Both charge tube and ram axial movements are generated through the ball screw/nut arrangement. The charge tube axial drive rotates the charge tube ballnut which is stationary with respect to the fuelling machine. This motion causes the charge tube screw to advance or retract axially depending on the direction of the ballnut rotation. All actions are in heavy water. There is no lubrication (other than the water).

For a ball screw/nut arrangement to work with minimum torque, it is crucial that the balls do not bunch up. Adjacent balls in a screw can come into contact with each other by:

- vibrations
- bending of the screw due to loading
- imperfect screw profiles
- poor deflectors or guides

When any of the above occur, higher torques are required to overcome the bunching up of the balls. An effective ballscrew design will have less potential for ball bunching or if it does have some ball bunching, the balls will separate easily.

DISCUSSION

Six heads are used on the three fuelling machine trolleys for normal fuelling operations. A further two heads are required to allow for maintenance and to provide for such reactor face activities as PIPE and CIGAR.

Seven heads were successfully delivered to site from the head supplier. During final acceptance testing of the eighth head, stalls on the charge tube screw assembly prevented its delivery to site.

Prior to the final acceptance testing, a series of preliminary tests of the head was completed on a test rig that simulated a Darlington fuel channel. During this initial testing, frequent charge tube axial stalls (due to higher than expected torques) were noted. Note that average torques typically seen on an acceptable charge tube are 50-80inch.lbs while peak torques of 150-200inch.lbs were typically seen on Head 8 charge tube.

The fuelling machine was removed from the test rig, disassembled, inspected, cleaned, re-built (with a spare screw/nut assembly) and returned to the test rig for a second series of testing. The original screw/nut assembly was returned to the sub-supplier for evaluation.

During the second series of testing, cold runs (260oC) were successful with no stalls observed. However, when the test rig temperature was increased to 300oC, frequent stalls began to appear. When the drive torque was plotted against time, many torque peaks (as high as twice the average torque) were noted.

Clearly there was a problem with the charge tube assembly.

Although the major efforts were directed at resolving whether the charge screw was the route cause of the problems, it was also felt that the root cause could possible lie with head component misalignment. It should be noted here that Head 8 was at one time overheated when it was used in an unrelated test program.

Ontario Hydro Technology was contracted to perform a series of tests on Head 8. Accelerometer probes were located at several points along the outer housing of the charge tube axial drive (see figure 3[(1)]). Torque profiles were also generated. A series of thirty two tests were completed. Sixteen of the tests were in advance mode and sixteen were in retract mode. These tests were completed under several different conditions including:

- off channel
- on channel, cold and de-pressurized
- on channel, cold and pressurized
- on channel, cold pressurized with two fuel bundles loaded into fuel carrier.
- on channel, hot and pressurized
- on channel, hot pressurized with two fuel bundles loaded into the fuel carrier

Results of this series of tests(2) appeared to indicate that the high torques seen with Head 8 were not caused by:

- the screw touching at the front end of the fuelling machine
- alignment problems at the rear of the fuelling machine
- the input drive or gear box

The testing did appear to indicate that the higher than expected torques were caused by either pitch/thread depth variations from the front to the back of the screw or pre-loading on the ballnut. In addition, there was evidence of double contact with the balls on the screw thread. This double contact indicated that there was some imperfection in the screw thread profile.

In later discussions, test rig crud levels were raised as a possible additional contributing factor since the rig was predominantly carbon steel with minimal oxygen control.

The evidence now clearly pointed to a ballnut/screw problem and not a misalignment problem.

A set of experiments were then conducted to narrow down the problem further. A series of tests were completed on a ballnut/screw arrangement that was located away from the head on a test bench. Seven variables were altered. These variables included:

- changing the screw (two spare screws were available)
- changing the nut (two spare nuts were available)
- changing the screw bearings
- changing the pre-load on the nut (100lbs and 600lbs)
- changing the cleanliness of the screw (by adding fine particulate crud)
- changing ballnut transfer tube
- changing ballnut deflectors

Results of this testing are shown in figure 4. Two parameters appeared to predominate, screw pre-load and rig cleanliness (crud).

Head 8 was then re-built. The pre-load on the screw was reduced from the normal level of 600lbs down to 100lbs. Efforts were undertaken to ensure that the test rig was as clean as possible. The rig was run for several days with purification flows maximized, As much crud as possible was removed from the rig.

With Head 8 now rebuilt using optimum conditions as determined from the bench testing (lower ballscrew pre-load and with a clean test rig), final head acceptance testing recommenced.

Again, higher than expected torques were observed. This was somewhat disappointing since all the evidence from bench test work had indicated that the problem was well understood and that we had completed effective fixes (although bench test work was not completed at normal temperatures and pressures).

It was now evident that Head 8 was not adequate for normal production use. Further work was required on the ballnut/screw arrangement.

The screw/nut supplier was contacted for further input. The original screw from Head 8 (which had been previously returned to the sub-supplier) was inspected more closely. Indications of double contact on the screw thread were evident. In addition the screw was found to be bent by about 13 thousandths of an inch. Although this bending was acceptable for use in normal fuelling machine operation, it was too far out for a precise grinding operation.

A decision was made to straighten the screw and regrind the screw thread. It was recognized that the screw could be badly cracked or possibly broken during the straightening process. However it was equally recognized that we would not be able to use the screw in the current state. The screw was eventually straightened to within 2 thousandths of an inch but could not be straightened further. A decision was then made to machine the screw straight.

Following the straightening of the screw, the thread profile was re-ground. The material removed from the thread profile was mostly removed in the axial direction (about 5 thousandths of an inch). The depth of the thread was only altered in the area where the depth was shallowest (by about 2 thousandths of an inch).

The screw (with its nut) was then returned for re-building into Head 8.

Subsequent acceptance testing with Head 8 with the re-machined screw was successful. No high torques or stops were observed during the acceptance testing. Head 8 was finally delivered to site during the summer of 1995.

CONCLUSION

During final acceptance testing of the eighth head, stalls on the charge tube screw assembly prevented its delivery to site.

An in depth series of tests undertaken at site, at the supplier and at the screw sub-supplier eventually determined the root cause of the problem to be a badly machined screw.

The screw was returned to the manufacture for re-work which included straightening the screw and re-machining the thread profile.

Subsequent testing of Head 8 using the re-machined screw was successful. Head 8 was eventually delivered to site during the summer of 1995 and is presently available for service.
Because the testing program showed the importance of using torque monitoring as a diagnostic tool, future activities are being directed at developing torque sensing equipment for use on the fuelling machines during normal operation.

ACKNOWLEDGEMENT

The authors wish to thank technician staff at GE (Canada) for all their hard work at the test rig which allowed us to get a better understanding of the problems being experienced with Head 8. In particular, the authors wish to thank Art White, Don Foley, Dan Finnegan and Dan Ayotte for their dedication and commitment.

The authors would also like to thank Phil Dale of Ontario Hydro Technologies for his valuable insight which allowed us to better focus in on the route cause of the problem.

REFERENCES

1. Darlington GS-A Fuel Handling System Technical Document.

2. Head 8 Acceptance Testing Minutes of Meeting dated 11 August 1994.

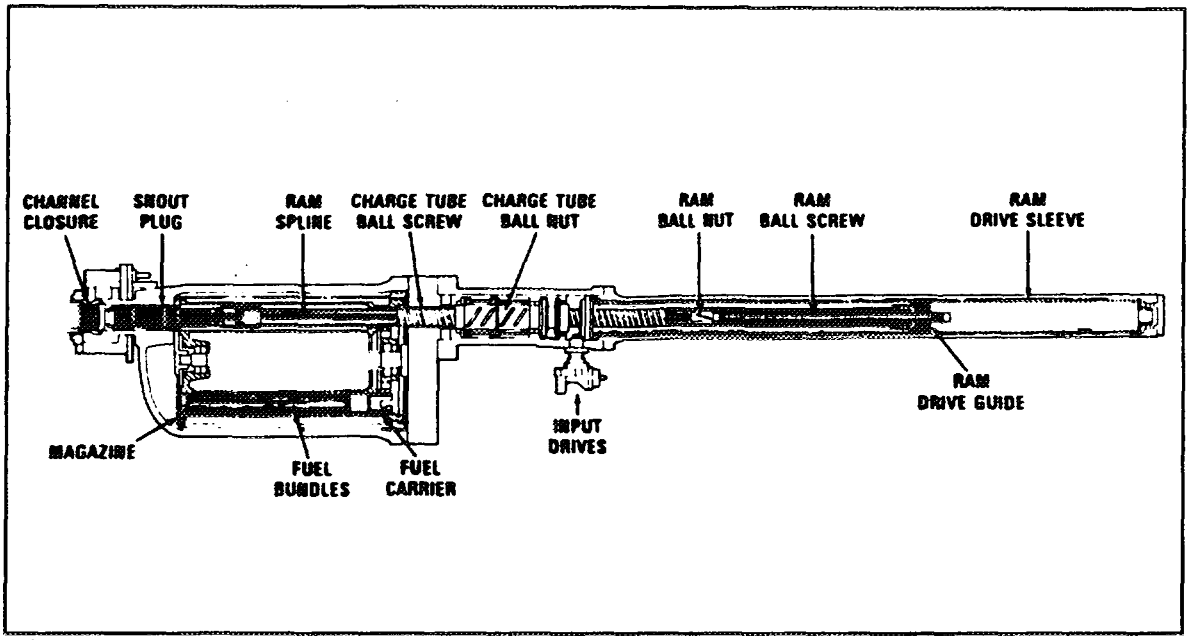

Figure 1: Schematic of Darlington Fuelling Machine.

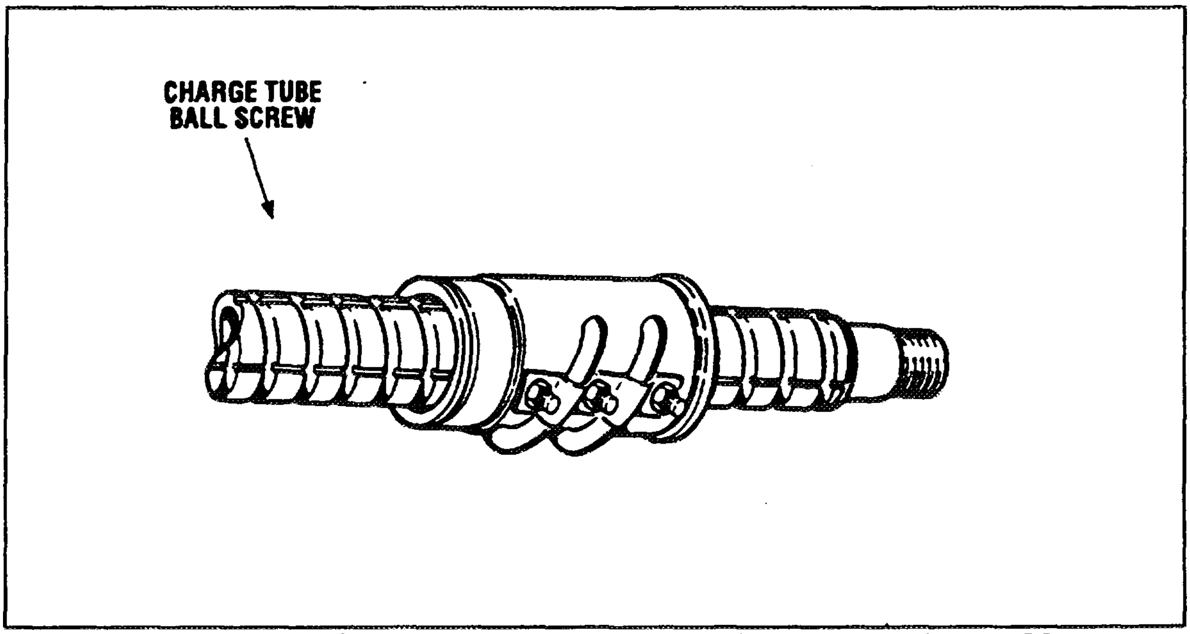

Figure 2: Schematic of a Charge Tube with a Single Ball Nut.

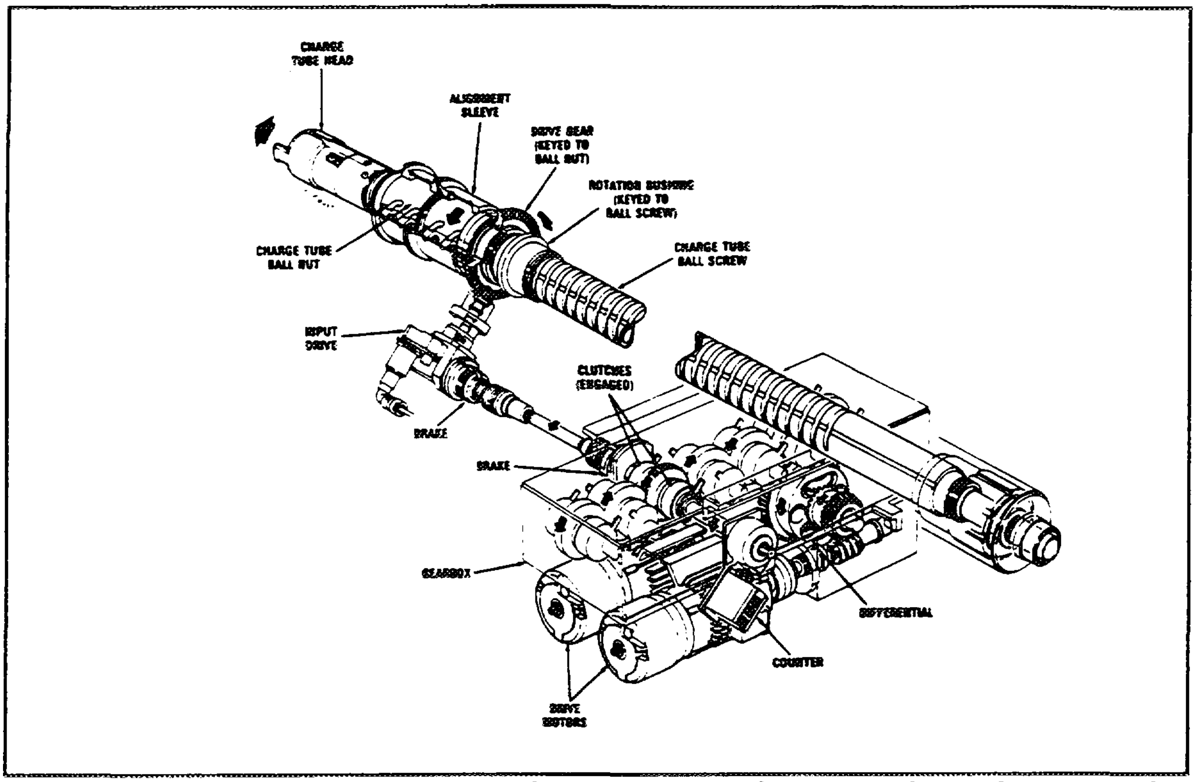

Figure 3: Schematic of Darlington Fuelling Machine Charge Tube Drive Mechanism.

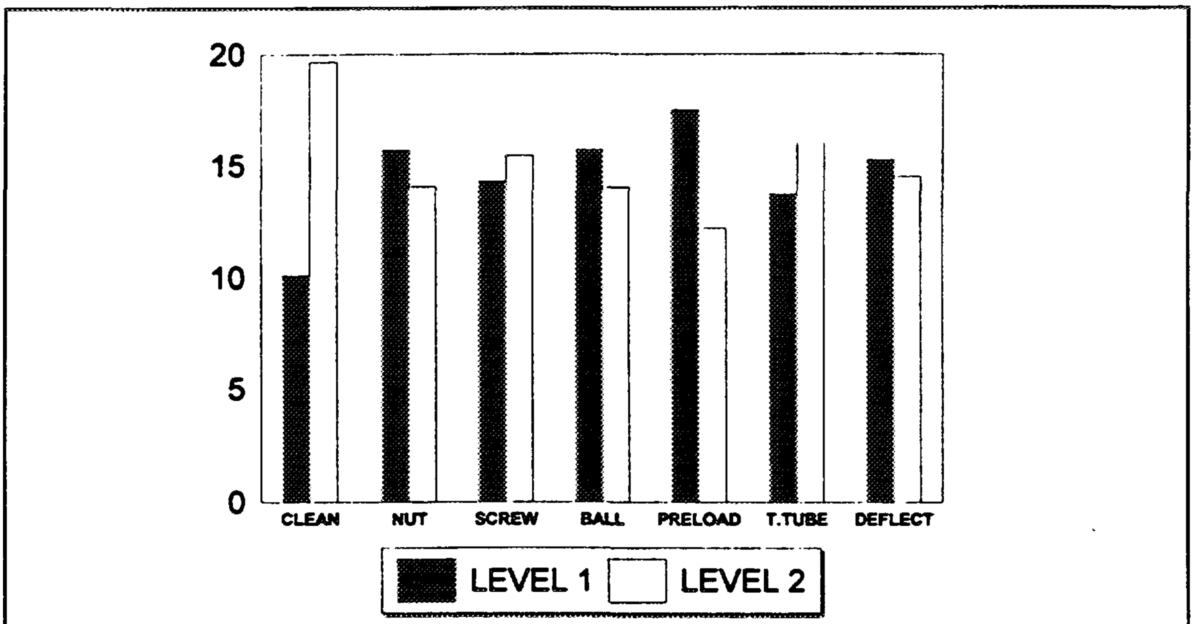

Figure 4: Result of Experiments to Determine Which Major Variable(s) Affect Ballnut/Screw Performance.

FUEL HANDLING SYSTEM OF INDIAN 500 MWe PHWR-EVOLUTION AND INNOVATIONS

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ABSTRACT

India has gained rich experience in design, manufacture, testing, operation and maintenance of Fuel Handling System of CANDU type PHWRs. When design and layout of the first 500 MWe PHWR was being evolved, it has been possible for us to introduce many special and innovative features in the Fuel Handling System which are friendly for O&M personnel. Some of these are:

- Simple, robust and modular mechanisms for ease of maintenance.
- Shorter turn-around time for refuelling a channel by introduction of a transit equipment between Fuelling Machine (FM) Head and light water equipment.
- Optimised layout to transport spent fuel in straight and short path and also to facilitate direct wheeling out of FM Head from Reactor Building to Service Building.
- Provision to operate the FM Head even when Primary Heat Transport (PHT) System is open for maintenance.
- Control-console engineered for carrying out refuelling operations in the sitting position.
- Dedicated calibration and maintenance facility to facilitate quick replacement of FM Head as a single unit.

The above special features have been described in this paper.

1. INTRODUCTION

Pressurised Heavy Water Reactors (PHWRs) built in India have evolved from the CANDU type Douglas Point reactor. The On-power Fuel Handling Systems (FHS) at Rajasthan (RAPS-1&2) were built with Canadian collaboration and are identical to that at Douglas Point. The design at Madras (MAPS-1&2) is similar to that of RAPS-1&2 but manufacture, testing, commissioning and installation were carried out in India. All the subsequent reactors of 220 MWe capacity at Narora (NAPS-1&2), Kakrapar (KAPS-1&2), Kaiga (Kaiga-1&2) and Rajasthan (RAPP-3&4) have FHS which are significantly different from the Rajasthan and Madras reactors. These systems were entirely designed and built in India. Thus, we have gained rich experience in design, manufacture, testing, operation and maintenance of FHS of PHWRs.

Now we have embarked on the detailed design of India's first 500 MWe PHWR to be built at Tarapur (TAPP-3&4). For a 500 MWe PHWR, the refuelling requirement is about two channels per full-power.

day as compared to about one channel per full-power day in case of a 220 MWe PHWR. The increased refuelling demand for 500 MWe reactor requires better availability of FHS. The FHS design should also strive to achieve shorter turn-around time. Further, there is a need to economise on the man-rem requirements during maintenance.

In order to meet the above requirements, the FHS for 500 MWe PHWR has been designed to be simple and rugged. Modular construction has been followed for the various sub-assemblies of Fuelling Machine Head and other equipment of Fuel Transfer System. In addition, a full fledged Calibration and Maintenance Facility has been provided.

When design and layout of the plant were being evolved, it has been possible for us to introduce in the FHS, many unique features which are friendly to operation and maintenance personnel, apart from being more rugged and reliable.

The above features are described in detail in this paper.

2. FUELLING MACHINE (FM) HEAD

FM Head is an important equipment of the FHS. The FM Head consists of three major sub-assemblies, viz., the magazine assembly, the snout assembly and the ram assembly.

2.1 Magazine Assembly

The magazine assembly is made up of a pressure housing and an end cover, which when joined together by a high pressure coupling, form a pressure vessel. A rotary magazine is mounted inside the pressure housing to store new or spent fuel bundles and various other accessories required for carrying out the on-power refuelling operations.

The magazine rotor is made of twelve machined tubes positioned around a central shaft and held in place at each end by two jig bored plates. This rotor assembly is mounted over a central shaft which is bolted to pressure housing at one end and supported in the end cover at the front end. Thus the magazine rotor is a simply supported assembly. The internal gear attached to the rotor is driven by two pinion gears mounted on water lubricated bearings in the rear end of the pressure housing. Each pinion gear, in turn, is driven by an indexing drive gear box through a rotary balanced mechanical shaft seal cartridge. The backlash free indexing drive gear box is driven by an oil hydraulic motor. (Fig 1)

Some special features of this assembly are:

- In the rotor design, welding is completely avoided. Distortion control during welding and subsequent heat treatment, as also the precision machining of the fabricated structure, has been a demanding and difficult operation in earlier design. Thus the 'bolted design' makes it manufacturer-friendly.

- The simply supported rotor design is more rigid and avoids the alignment errors that are possible in case of cantilever mounting used in the earlier design.

- The magazine drive shaft seal has been made as a modular 'cartridge' type unit. Thus, a fully assembled and tested replacement cartridge can be fitted in place of the existing one without dis-assembling too many components.

2.2 Snout Assembly

The snout assembly is attached to the front end of the magazine assembly. It consists of a centre support which carries a bellow like metallic seal for effecting a high pressure joint with the end fitting. A mechanism consisting of four cam actuated jaw segments, a clamping barrel with internal trapezoidal (acme) threads, and a worm gear with matching external trapezoidal threads, are used for moving and clamping on to an end fitting. (Fig.2)

Some special features of this assembly are:

- An oil hydraulic motor is used to drive the mechanism instead of the piston actuated rack and pinion drive used earlier. This enables use of single start acme threads in clamping barrel, giving the required irreversibility to avoid unclamping on loss of oil pressure.

- The snout assembly is modular in construction. This allows removal of the snout assembly without the need for the disassembly of the high pressure coupling between the end cover and the pressure housing (also referred to, some times, as magazine housing).

2.3 Ram Assembly

The ram assembly is mounted at the back end of the magazine assembly. The ram assembly is required for operating the different accessories during fuelling operations and to push new fuel into or receive spent fuel from the channel. In this B-Ram and C-Ram are powered by two separate (but co-axial) rack and pinion drives. Each of these drives consists of a dual rack system driven by two pinions which in turn are driven by two high torque low speed oil hydraulic motors, thus providing a balanced, duplicated drive. Latch ram is also operated by oil hydraulic motor. In addition, a system for emergency movement of B-Ram and C-Ram separately, by using heavy water pressure in the ram housing, has been incorporated. This would be particularly useful in boxing up the channel in abnormal situations (Fig 3).

Some special features of this assembly are :

- The ram assembly is of modular construction, allowing its removal as a single unit from the pressure housing.
- The rack and pinion drive selected is of a robust design and expected to perform better compared to ball screw drive used in the earlier design.
- C-Ram position monitoring scheme is fully outside water environment and hence would be more reliable than the wire rope design used earlier.

3. FUEL TRANSFER (FT) SYSTEM

The FT system requires to perform the basic functions of storing the new fuel in reactor building (RB), exchange of new fuel with spent fuel and transport of spent fuel using the well-proven 'Shuttle Transport System' to Spent Fuel Storage Bay (SFSB), located outside the reactor containment. It comprises of two sets of mechanical equipment, one for the north and the other for the south side of the reactor.

The time taken for the FM Head to discharge spent fuel received from the core, and to accept new fuel for refuelling the next channel, has been reduced by having a common Fuel Transfer Port and incorporating an 'Exchange Programme'. In the exchange programme, while two spent fuel bundles are discharged into the Transfer Magazine (TM), two new fuel bundles are received from the TM into the FM Head. The spent fuel in the TM can be sent to the SFSB simultaneously with the refuelling of next channel. Thus the overall cycle time for refuelling a channel has been reduced to about two and half hours.

All the equipment in the spent fuel path are closed vessels and do not freely communicate with the atmosphere outside the equipment. Thus, spread of activity (mostly from the occasional failed fuel bundles) into the FT Room and other areas in the reactor building is considerably reduced.

TM and the other important equipment of FT system namely Shuttle Transfer Station (STS) are described below;

3.1 Transfer Magazine (TM)

TM (Fig 4) interacts with three different equipment namely New Fuel Magazine, FM Head and Shuttle Transfer Station for the receipt of new fuel bundles, exchange of new fuel bundles with spent fuel bundles and discharge of spent fuel bundles respectively.

It consists of a rectangular stainless steel housing in which a carriage moves linearly up and down. The carriage comprises of six fuel tubes which can be aligned to three ports of the housing to receive / discharge the fuel bundles. The carriage is driven by an oil hydraulic motor through a rotating ball nut and translating ball screw. The position of the carriage is controlled / monitored by duplicate set of potentiometer assemblies. The aligned position of carriage is also confirmed by an independent reed switch assembly.

Some special features of TM are :

- Capability to receive 8 spent fuel bundles from FM Head and load 8 new fuel bundles into the FM Head in one sequence.
- Simple fuel stops for preventing accidental dropping of spent fuel into the housing.
- Drive system and position sensing devices are brought outside the TM Housing to enable easy maintenance. Shielding provided around TM Housing allowing maintenance even when the TM contains spent fuel.
- Provision for cooling of spent fuel bundles stored in it under various conditions.

3.2 Shuttle Transfer Station (STS)

STS (Fig 5) comprises of a shuttle carriage which moves vertically up and down in a cylindrical housing partly filled with light water. It interacts with TM under dry environment to receive the spent fuel into a special carrier tube called 'Shuttle' housed in the carriage. The Shuttle is then lowered and transported to Spent Fuel Receiving Bay through Shuttle Transport Tube (STT) using hydraulic force. The STT is kept straight and short for smooth movement of fuel.

The aligned position of the carriage is achieved against mechanical stops. The position is monitored continuously by a duplicate set of potentiometer assemblies.

Some special features of this assembly are :

- Simple wire rope drive mechanism powered by oil hydraulic motor.
- Fuel shutter for preventing accidental dropping of fuel into the housing.
- Special dampener attached to shuttle carriage to reduce impact force on bundle due to postulated free fall of carriage.

4. ISOLATION OF FM HEAD D2O SYSTEM FROM PHT

As heavy water storage tank is common to FM Head and PHT System, FM Supply pumps cannot be run to meet FHS needs when the PHT system has ben opened for maintenance. This puts a limitation on the preventive maintenance and shutdown maintenance jobs to be taken up on the FHS as heavy water supply would not be available. Therefore, an additional 5 tonnes capacity storage tank together with associated circuits have been provided so that FHS can be operated in an `isolated mode'.

5. FUEL HANDLING CONTROLS

The controls for the FHS have been based on Programmable Logic Controller (PLC). In addition, a sitting control console has been provided for FHS which would help to reduce the operator fatigue. A dedicated Data Acquisition System (DAS) in conjunction with Expert System would be designed to work as Operator Support System (OSS) and as a tool for maintenance and health monitoring of certain critical actuators.

6. CALIBRATION & MAINTENANCE FACILITY (CMF)

In order to meet the enhanced refuelling requirement for 500 MWe, the on-line maintenance activities are required to be minimised. This can be achieved by adopting modular design of the assemblies so that a faulty assembly can be quickly replaced with a well maintained, tested, calibrated and proven reactor-worthy assembly. This approach also avoids any repeat work, which sometimes becomes necessary due to possible maintenance errors in inconvenient locations and radiation fields.

With the above in view, a full fledged off-line Calibration and Maintenance Facility has been provided for the first time, in case of 500 MWe PHWR. This facility has the necessary features for carrying out maintenance, testing and calibration of important components, critical sub-assemblies and complete FM Head with simulated reactor conditions outside the reactor building in a tritium free environment. This would enable to keep an FM Head in the poised condition ready for replacement in a short time whenever required.

In addition, care has been taken to include the following desirable features also in the CMF.

i) Simulating and analysing, when necessary, the mal-operations in the FM Heads as they occur in the reactor.
ii) Providing training facilities for operations and maintenance personnel of the fuel handling unit.

6.1 Layout of CMF

The FM service areas are located below the respective FM Vaults. The two FM service areas in each reactor unit are inter-connected with an FM tunnel which runs below the calandria vault. The FM tunnel permits the movement of FM Head, from any of the two service areas through a single FM airlock, directly to CMF located in the service building (Fig 6). The tunnel has adequate concrete and water shielding to allow passage of FM Head even when the reactor is operating. The CMF is located in between the two reactor buildings and is common for a twin unit station. See Fig 7 for the various facilities provided in the CMF, some of which are described below:

6.2 FM Head Test Facility

Three test channels have been provided for testing of FM Heads. The central channel is similar to a reactor coolant channel. On this channel, two FM heads can be tested in unison either in hot condition or at ambient temperature with either of the two FM Heads as upstream or downstream head. Each of the other two channels can be used for testing an FM Head independently either as upstream or downstream under ambient temperature. The necessary forces (hydraulic drag forces / ram forces of the opposite end FM Head) can be simulated by using a test ram housed in the opposite end of the test channel

6.3 Ram Assembly Test Facility

It is desirable to test a Ram Assembly independently before it is assembled to an FM Head. For this purpose a Ram Assembly Test Rig (RATR) has been provided. A test piston with labyrinth seals is housed inside the RATR to generate forces required to adjust/calibrate the forces of the B-Ram/C-Ram/latch.

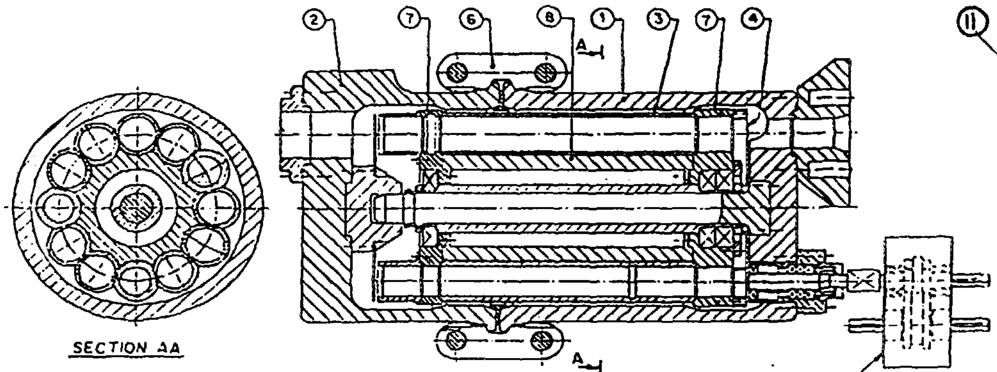
6.4 Other Facilities

Several test rigs have been located in the CMF to independently assemble, test, calibrate and fine tune various sub-assemblies of the FM Head (such as snout assembly, coolant channel closure plugs etc.). Further, provision has been made in the CMF to decontaminate the FM Head before taking up any work on it.

7. CONCLUSIONS

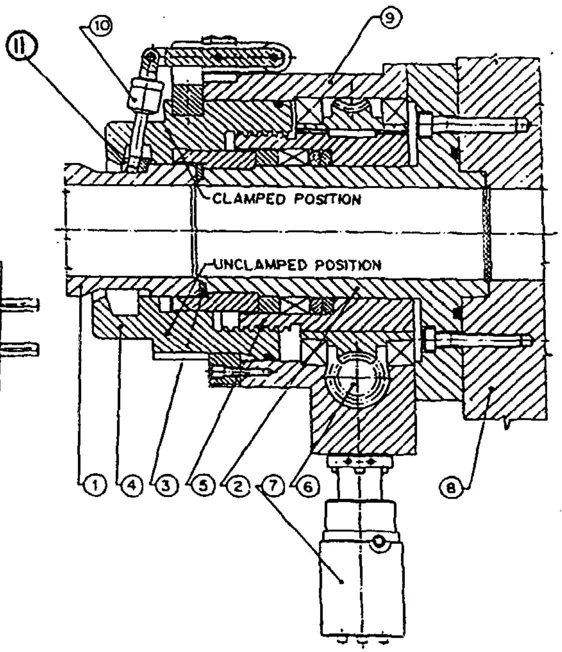
It has been possible for us to briefly cover in this paper only some of the improvements and innovative features of the 500 MWe Indian PHWR FHS design.

It is well recognised that the performance of the on-power FHS play a major role for the success of PHWRs. We hope that these features will contribute in a large way for a troublefree and optimal performance of the FHS.



1. PRESSURE HOUSING.
2. END COVER.
3. MAGAZINE ROTOR ASSY.
4. INTERNAL GEAR
5. MAGAZINE DRIVE INDEXING ASSY.
6. HIGH PRESSURE COUPLING.
7. BEARING HOUSING
8. CENTRAL SLEEVE

FIG. 1 SIMPLIFIED VIEW OF MAGAZINE ASSEMBLY
(500 MWe PHWR)



1. END FITTING.
2. CENTRE SUPPORT.
3. SKINNER SEAL.
4. CLAMPING BARREL.
5. SCREW
6. WORM AND WHEEL DRIVE.
7. SNOUT EMERGENCY LOCK.
8. MAGAZINE HOUSING END COVER
9. OUTER SUPPORT
10. CLAMPING POST
11. WEDGE SEGMENT

FIG. 2 SIMPLIFIED VIEW OF SNOUT ASSEMBLY
(500 MWe PHWR)

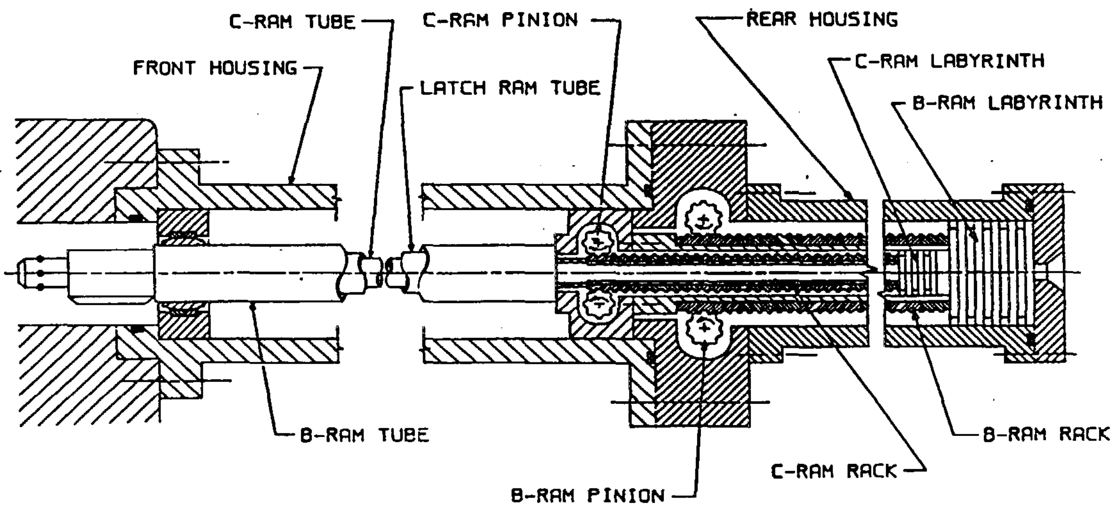


FIG. 3 SIMPLIFIED SKETCH OF RAM ASSEMBLY
(500 MWe. PHWR)

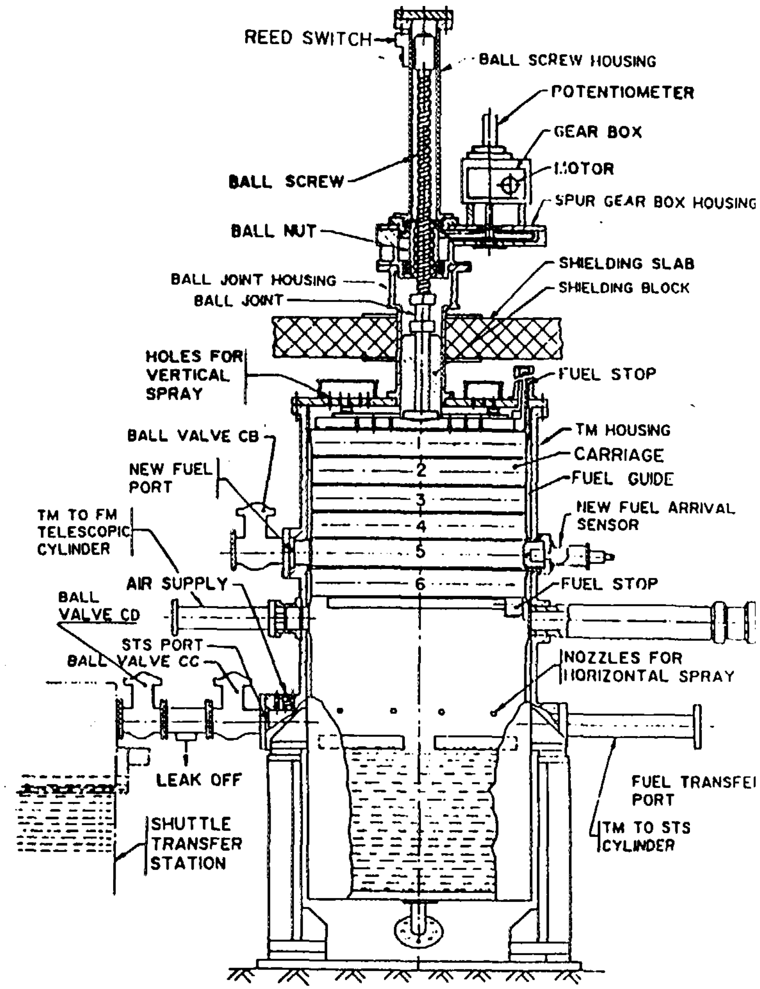


FIG. 4 SIMPLIFIED VIEW OF TRANSFER MAGAZINE
(500 MWe PHWR)

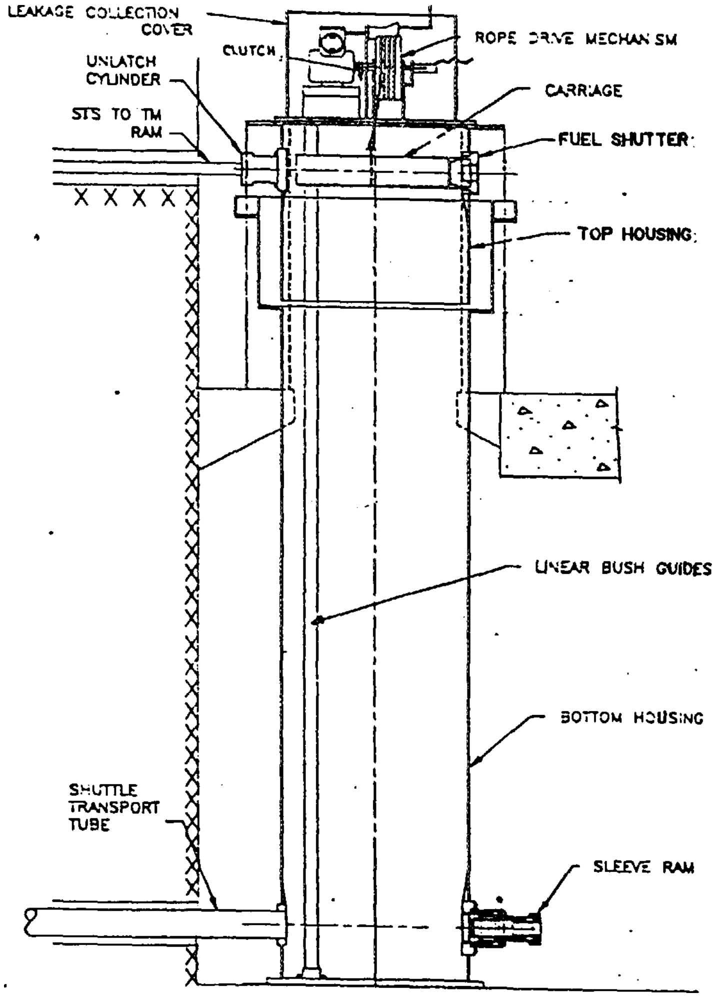


FIG. 5: SIMPLIFIED VIEW OF SHUTTLE TRANSFER STATION
(500 MWe PHWR)

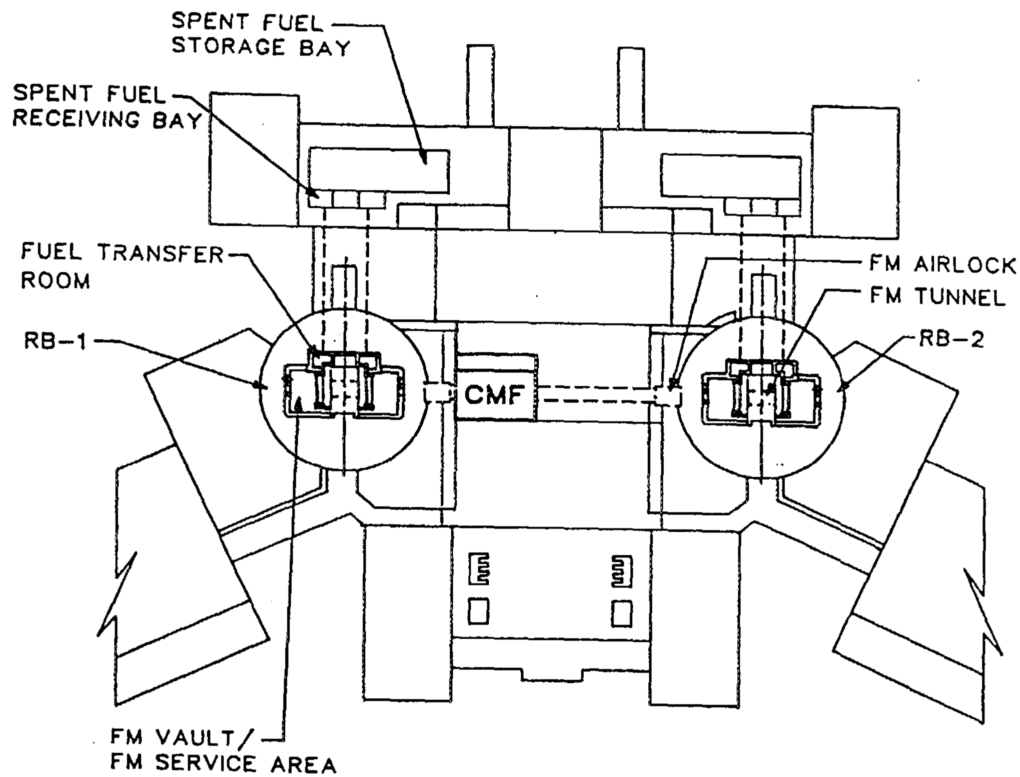
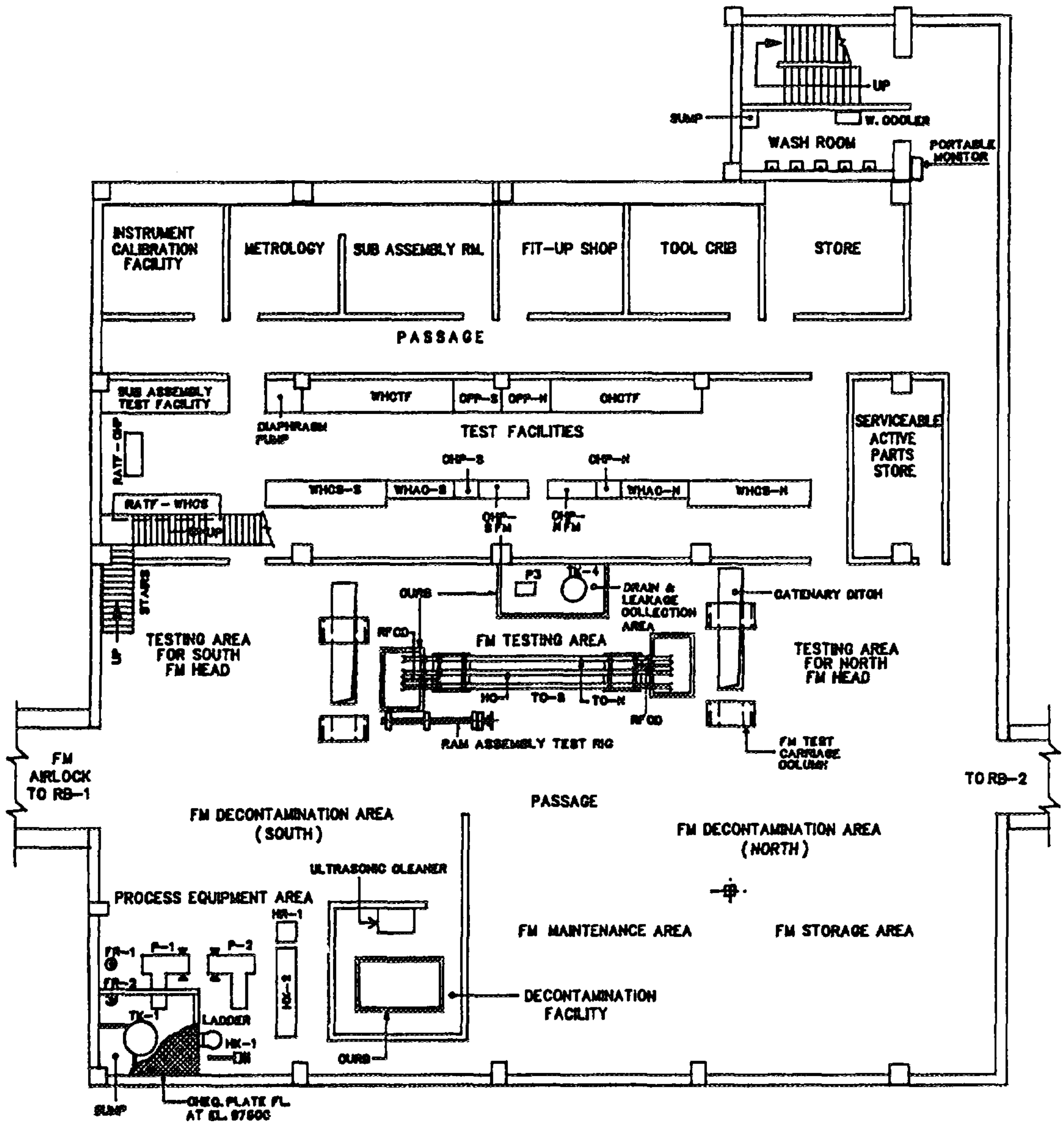


FIG:6 LAYOUT OF FUEL HANDLING EQPT./AREAS
(500 MWe PHWR)



<u>ACRONYMS;</u>

WHCTF = WATER HYDRAULIC COMPONENTS TEST FACILITY
OPP = OIL POWER PACK
OHCTF = OIL HYDRAULIC COMPONENTS TEST FACILITY
WHCS = WATER HADRAULIC CONTROL STATION
WHAC = WATER HYDRAULIC ACTUATOR CABINET
RATF = RAM ASSEMBLY TEST FACILITY
OHP = OIL HYDRAULIC PANEL
TC = TEST CHANNEL
HC = HOT CHANNEL
RFCD = RAM FORCE CALIBRATION DEVICE
P1, P2 = TRIPLEX PUMPS
P3 = CENTRIFUGAL PUMP
HX = HEAT EXCHANGER
HR = ELECTRIC HEATER
FR = FILTER
TK = TANK

FIG:7 CMF LAYOUT AT FLOOR EL. 93.00 M.

ANALYSIS OF FUEL HANDLING SYSTEM FOR FUEL BUNDLE SAFETY DURING STATION BLACKOUT IN 500 MWe PHWR UNIT OF INDIA

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ABSTRACT

Situations of Station Blackout (SBO) i.e. postulated concurrent unavailability of Class III and Class IV power, could arise for a long period, while on-power refuelling or other fuel handling operations are in progress with the hot irradiated fuel bundles being any where in the system between the Reactor Building to the Spent Fuel Storage Bay.

The cooling provisions for these fuel bundles are diverse and specific to the various stages of fuel handling operations and are either on Class III or on Class II power with particular requirements of instrument air. Therefore, during SBO, due to the limited availability of Class II power and instrument air, it becomes difficult to maintain cooling to these fuel bundles. However, some minimal cooling is essential, to ensure the safety of the bundles.

As discussed in the paper, safety of these fuel bundles in the system and/or for those lying in the liner tube region of the reactor end fitting is ensured, during SBO, by resorting to passive means like 'stay-put', 'gravity- fill', 'D_2O- steaming' etc. for cooling the bundles. The paper also describes various consequences emanating from these cooling schemes.

1.0 INTRODUCTION

The on-power refuelling operation is commenced only when Class IV power is available. However, Station Blackout (SBO) i.e. postulated concurrent unavailability of Class IV and Class III electrical power, may occur when refuelling or other fuel handling operations are in progress. Under such situation hot irradiated fuel bundles could be present anywhere in the Fuel Handling System (FHS) and/or in the liner tube region of the downstream reactor end fitting i.e. out of the thermosyphoning flow path of the coolant channel.

For the safety of these fuel bundles, necessary cooling is required in order to avoid their overheating due to the decay heat. However, during SBO, when Class III power is not available and availability of Class II power as well as instrument air is limited, the requirement of cooling these fuel bundles becomes difficult to achieve, since:

a) The normal cooling provisions require substantial electrical power.

b) The fuel handling operations are performed in a number of stages, with specific cooling provisions for each stage.

c) The equipment of the FHS are located at different places from the Reactor Building (RB) to the Spent Fuel Storage Bay (SFSB), Fig.1.

Therefore, in order to assess the effect of SBO on the safety of the irradiated fuel bundles present in the FHS, various scenarios within the system, likely to exist at the time of SBO, have been identified and analysed. Further, a detailed strategy has been developed to provide adequate cooling to the bundles with the minimal power consumption.

2.0 POINTS CONSIDERED IN SBO HANDLING STRATEGY AND COOLING SCHEME FOR FHS:

The SBO handling strategy would greatly be influenced by the exact stage of fuel handling operations being carried out at the time of SBO.

Following general requirements were set up for evolving the SBO handling strategy for FHS :

a) The scheme to be conceptualised for cooling the fuel bundles should make best use of the already available system, the interconnections between FHS and Primary Heat Transport System (PHT) and the inherent cooling capabilities of FHS.

b) In case a cooling scheme can not be evolved with the existing hardware, or, the scheme using the existing circuit is not practicable, modifications in the design should be kept to the bare minimum.

c) Scheme should be general, to the extent possible so as to be applicable for most of the stages of fuel handling operations.

d) The electric power and pneumatic power required should be low.

e) Loss of D_2O and transfer/leakage of heavy water from high pressure circuit of PHT to low pressure circuit should be as low as possible.

f) The FHS should be brought to safe and stable state, at the earliest, requiring minimal operations.

g) Maintaining of the cooling to the spent fuel bundles should be with minimum interventions.

h) Safety of the system should preferably be established with minimum feed backs.

i) Entry to and stay in the RB should be minimal.

3.0 PRINCIPLE ADOPTED FOR COOLING OF FUEL BUNDLES:

The SBO would be declared if Class III power is not restored even after 6* minutes of Class-IV power failure. This implies that the hot irradiated fuel bundles in the downstream liner tube region, if any, or in the FHS would have seen at least 6 minutes of forced water cooling due to the FM supply pump, on Class II power, continuing to operate till the declaration of SBO. The decay power of the fuel bundles at this instance would be almost 3% of the full power. Such hot irradiated fuel bundles if exposed to nearly stagnant air environment, would reach a steady state sheath temperature of the order of 1500 K [1]. The order of the temperature in stagnant steam environment has been found to be the same. In the process of rise in temperature to such values, sheath splitting due to oxidation swelling of fuel, is probable [2], as the oxidising environment could exist due to the thermal and radiation decomposition of water or due to the presence of air.

However, it has been found that as long as such fuel bundles are kept immersed in water, even if the water is boiling without any forced circulation, the clad surface temperature of the fuel bundle would be close to the saturation temperature of water corresponding to the prevailing pressure. This is established through the following calculation procedure:

a) Calculation of sheath temperature for the maximum power fuel element of the maximum power fuel bundle (with 6 minutes of cooling period) assuming the element to be immersed, in isolation, in the boiling pool of water at a saturation temperature of 583 K corresponding to the maximum operating condition at the reactor outlet (The sheath temperature is evaluated to be approximately 5 K higher than the saturation temperature).

b) Verification of the above result, on the extrapolated boiling curve, Fig. 3, obtained by data from Fig. 4 (The sheath temperature is found to be nearly same as that obtained in step 'a' above).

c) Extrapolation of the result for the bundle geometry by estimating the critical heat flux for the bundle with the help of equation 10.112 of [3] (point A in Fig.3) and using the concept depicted in Fig.5 (It is concluded that the sheath temperature of any element in the fuel bundle would be of the same order as that of the single fuel element immersed in water, in isolation, as stated in 'a' above).

From the above calculations it is established that in the system, sheath temperature can go to about 588 K, at the most. This is considered acceptable for the safety of the fuel bundles.

Therefore, in order to dispense with the use of pumps during SBO, in absence of forced circulation, the fuel bundles are required to be kept immersed in water for their safety.

* 6 minutes is reasonable time for the operator to attempt manual connection of Diesel Generator supply to the buses, in case of failure of automatic restoration system.

4.0 POSSIBLE SCENARIOS FOR FHS DURING SBO:

In order to understand the behaviour of the system during SBO, especially from the point of view of safety and cooling of the fuel bundles with the truncated cooling provisions, and also to evolve the cooling scheme for these fuel bundles, following scenarios have been analysed:

A) Fuelling machines (FMs) hydraulically connected to channel (i.e. on-reactor mode) and downstream shielding plug yet to be withdrawn.

B) FMs hydraulically connected to channel (i.e. on-reactor mode) and downstream FM receiving (or has just received) spent fuel bundles.

C) FMs hydraulically isolated from channel (i.e. off-reactor mode/park mode) with spent fuel bundles in one of the FM.

D) FM hydraulically connected to transfer magazine (i.e. fuel transfer mode) with spent fuel bundles in FM and/or transfer magazine (TM) and/or in the tube connecting the FM and TM.

E) Spent fuel bundles under transit storage in TM.

F) A pair of spent fuel bundles in the process of transfer from TM to shuttle transfer station (i.e. dry transfer mode), Fig 2.

G) A pair of spent fuel bundles in the process of transfer from shuttle transfer station (STS) to SFSB.

H) I) Spent fuel bundles in an emergency tube[1] (ET) of Rehearsal Tube Facility (RTF).

ii) FM with spent fuel bundles in hydraulic communication with an ET of RTF.

[1] Among the 3 tubes in the Rehearsal Tube Facility 2 are Emergency Tubes which are same as reactor channel and are meant for storing spent fuel bundles in exigencies

5.0 ANALYSIS OF VARIOUS ALTERNATIVE COOLING SCHEMES CONSIDERED FOR THE IDENTIFIED SCENARIOS :

Various possible cooling schemes for individual scenarios were analysed in detail. Salient features of the analysis for a typical scenario are given in Table 1.

The cooling scheme adopted for various scenarios is summarised in Table 2.

6.0 IMPORTANT RESULTS OF THE ANALYSIS:

6.1 Thermosyphoning Between FHS and PHT

Thermosyphoning between the FHS and the interconnected PHT is possible to be established during off-reactor mode / park mode, fuel transfer mode and during the scenario pertaining to the ET (i.e. Scenario C, D and H respectively as given in Section 4.0). However, owing to the inadequate driving force, thermosyphoning in these cases leads to exposure of the fuel bundles to nearly stagnant steam environment which is unacceptable for fuel bundle safety as discussed in Section 3.0.

6.2 Cooling of Hot Irradiated Fuel Bundles

Cooling of hot irradiated fuel bundles present in the downstream liner tube and in FHS is ensured in line with the principle described in Section 3.0. The fuel bundles present in FM/TM/downstream liner tube region/ET would be cooled as follows:

a) Initially by sensible heating of the equipment and the hold-up water and later by steaming of this water.

b) Venting of steam from FM/TM/ET as the case may be.

c) Make-up of water (lost due to steaming and venting) from
i) the channel, in case FM and channel are in hydraulic communication;

ii) the PHT storage tank (by gravity flow) in all the remaining cases.

The fuel bundles in SFSB and in case present in STS during SBO, would be cooled by the subcooled light water of SFSB/STS.

In case, a pair of spent fuel bundles is undergoing the dry transfer operation at the time of Class IV power failure, the pair would be quickly brought to water environment in TM or STS, depending upon the situation (motive power for this would be available as it would take about 5 minutes). Afterwards the cooling of the pair would be as described above.

6.3 Loss of D_2O Inventory

The escape of D_2O in the form of steam, from the system to the FM vault, in 8 hours (the design target for SBO handling capability) could be at the most 1200 Kg., considering the following:

a) 13 irradiated bundles of maximum power channel (5.5 MWth) with an average cooling period of 10 minutes, out of the thermosyphoning loop, with FM on On-reactor mode.

b) 8 spent fuel bundles of the maximum power channel with an average cooling period of 45 minutes, being in the TM.

In case (a), 700 Kg. from high pressure circuit of PHT and in case (b), 500 Kg. from FT system /low pressure circuit of PHT would escape to FM vault.

6.4 Tritium build up in accessible area (V2 area) of RB as a result of Release of D_2O Steam from the FHS

The steaming from FHS would commence only after at least one hour from the declaration of SBO, when equipment containing the fuel bundles and the water inventory submerging the fuel bundles in these equipment, get heated-up. Therefore, build-up of tritium in FM vault and in other high enthalpy areas (V1 area) may occur only after the first hour of SBO. Due to the stipulated leak paths between V1 and V2 areas, the tritium build-up in V2 area is conservatively evaluated to be about 22 DAC at the end of 4 hours and about 70 DAC at the end of 8 hours, Fig. 8. This is considered acceptable under SBO conditions.

7.0 CONCLUSION

A detailed SBO handling strategy for FHS, has been developed, to bring the system to stable state with minimal energy consumption.

The safety of the irradiated fuel bundles present anywhere in the FHS, is ensured during SBO, by resorting to the inherent, passive cooling capability of the system. However, this will eventually result in some escape of D_2O and subsequent build-up of tritium.

REFERENCES

1. G.PAL, S.G.MARKANDEYA and V.VENKATRAJ, "Analysis of the Transient Behaviour of 37 Rod Fuel Cluster of 500 MWe PHWR During Spent Fuel Handling Operation", Proceedings of TENTH National Heat & Mass Transfer Conference; paper no. III-14; Srinagar, India, 1989.

2. J.NOVAK and G.MILLER, "Dry Fuel Handling: Station Experience and Ontario Hydro (CNS) Program", Proceedings of International Conference on CANDU Fuel, pp 322 -345, Chalk River, Canada, 1986.

3. ROBERT H. PERRY and DON GREEN, "Perry's Chemical Engineers Handbook", Sixth Edition, McGraw Hill.

4. COULSON J.M. and RICHARDSON J.F. "Chemical Engineering, Volume 1, Fluid Flow , Heat Transfer and Mass Transfer", Third Edition, Oxford Pergamon Press, 1977.

5. WARREN L. McCABE , JULIAN C. SMITH and PETER HARRIOT, "Unit Operations of Chemical Engineering", Fourth Edition, McGraw Hill.

6. ERNST U. SCHLUNDER et al, "HEDH- Heat Exchanger Design Handbook, Volume 2, Fluid Mechanics and Heat Transfer", Hemisphere Publishing Corporation, 1986.

TABLE 1

ANALYSIS OF VARIOUS COOLING SCHEMES TO EVOLVE THE SCHEME TO BE ADOPTED FOR HANDLING SBO FOR THE SCENARIO OF,

"FUELLING MACHINES HYDRAULICALLY CONNECTED TO CHANNEL AND DOWNSTREAM FM RECEIVING/RECEIVED SPENT FUEL BUNDLES"

Highlights of Alternative Schemes considered for handling SBO :

I. SWITCH OFF ELECTRICAL LOADS and STAY-PUT:
   - Scheme Leads to exposure of fuel bundles to stagnant steam environment, Fig.6.

II. [1]PUSH THE IRRADIATED FUEL BUNDLES PRESENT IN LINER TUBE REGION, BACK IN TO THE THERMOSYPHONING FLOW PATH OF THE CHANNEL :
   - Feasible only when 17 or less fuel bundles in the channel.
   - Operation of fuel stop will be required needing instrument air.
   - Scheme takes nearly 10 min. for execution and consumes 30 kwh energy.
   - Bundles will stay in thermosyphoning flow as the drag force on them is negligible.

III. [1]BOX-UP THE CHANNEL AT DOWNSTREAM END FITTING :
   - Possible only when 17 or less fuel bundles in the channel.
   - Guide sleeve to be removed, seal plugs to be installed.
   - Scheme takes nearly 10 min. for execution and consumes 30 kwh energy.

IV. CIRCULATION OF WATER IN FM BY FUEL TRANSFER (FT) D2O PUMP :
   - Pump would be required for almost entire duration of SBO.
   - Heat sink would be required for removal of decay heat from FT heat exchanger.

V. INJECT LIGHT-WATER INTO THE FM :
   - Scheme would downgrade entire PHT.

VI. VENT THE STEAM FROM FM and MAKE-UP THE WATER IN FM BY CHANNEL WATER.

   a) VENT BY VENT VALVE
   - Constant monitoring of D2O level in FM and constant adjustment of vent valve will be required.
   - Scheme leads to exposure of hoses to steam.

   b) VENT BY AIR TRAP, FIG. 7
   - Passive scheme, needs only valving-in of the trap.
   - No exposure of hoses to steam.

SCHEME ADOPTED : VENTING BY AIR TRAP AS PER VI(b) ABOVE IN LINE WITH THE POINTS STATED IN SECTION 2.0 AND THE PRINCIPLE OF COOLING DESCRIBED IN SECTION3.0.

[1] The scheme has been worked out with the assumption that no operation is required to be carried out on the upstream FM. Hence, the scheme is applicable only when 17 bundles are existing in the channel as 13 irradiated bundles, originally in the core, would be brought back, allowing additional 2 pairs to be residing in the space between the FM and the channel.

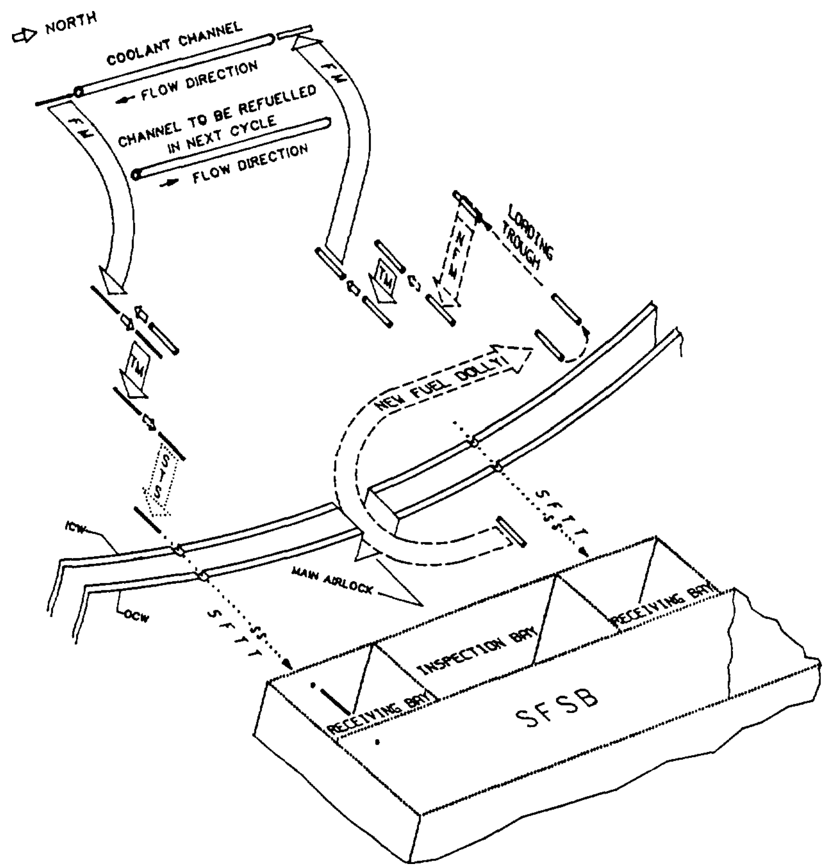
TABLE 2

SUMMARY OF COOLING SCHEMES FOR VARIOUS SCENARIOS[*]

ON-REACTOR MODE, DOWNSTREAM FM RECEIVING OR RECEIVED FUEL BUNDLES SCENARIO B[#]	OFF-REACTOR MODE, FM WITH EIGHT SPENT FUEL BUNDLES SCENARIO C[#]	FUEL TRANSFER MODE FUEL BUNDLES IN FM AND/OR IN TM SCENARIO D[#]	FUEL BUNDLES UNDER TRANSIT STORAGE IN TM SCENARIO E[#]	PAIR OF FUEL BUNDLES UNDERGOING DRY TRANSFER SCENARIO F[#]	PAIR OF FUEL BUNDLES IN SHUTTLE TRANSFER STATION SCENARIO G[#]	FUEL BUNDLES IN EMERGENCY TUBE, OR FM WITH SPENT FUEL BUNDLES IN COMMUNICATION WITH AN EMERGENCY TUBE OF REHEARSAL FACILITY SCENARIO Hi[#] & Hii[#]
VENTING OF FM BY AIR-TRAP (ON VENTING THE STEAM, THE LIQUID D2O WOULD BE INGRESSED INTO FM FROM THE CHANNEL & HENCE KEEPING THE FUEL BUNDLES IN FM IMMERSED IN LIQUID D2O)	SAME AS THAT FOR SCENARIO B (ON VENTING, THE LIQUID D2O WOULD COME INTO THE FM FROM PHT STORAGE TANK DUE TO GRAVITY & HENCE KEEPING BUNDLES IN FM IMMERSED IN LIQUID D2O)	SAME AS THAT FOR SCENARIO B (AFTER VENTING, THE D2O IN FM & TM WOULD BE SUFFICIENT TO KEEP THE BUNDLES SUBMERGED IN LIQUID D2O. HOWEVER, WHENEVER REQUIRED BASED ON TM LEVEL TRANSMITTER SIGNAL, THE TM MAY BE FILLED BY GRAVITY FLOW FROM PHT STORAGE TANK)	FILLING UP OF TM BY GRAVITY FLOW FROM PHT STORAGE TANK & SIMULTANEOUS VENTING OF TM BY AIR TRAP.	THE PAIR OF FUEL BUNDLES TO BE BROUGHT TO WATER ENVIRONMENT EITHER BY RETRACTING THE PAIR TO TM OR BY TAKING THE PAIR TO STS DEPENDING UPON THE SITUATION. AFTERWARDS COOLING WOULD BE AS PER EITHER THE PREVIOUS SCENARIO OR THE NEXT SCENARIO AS THE CASE MAY BE.	FOR KEEPING THE BUNDLES IMMERSED IN WATER ENVIRONMENT, SUFFICIENT QUANTITY OF H2O ABOVE THE PAIR OF BUNDLES WOULD BE AVAILABLE.	VENTING OF EMERGENCY TUBE BY AN AIR-TRAP (ON VENTING THE TUBES, BUNDLES WOLD BE KEPT IMMERSED IN D2O BY FILLING THE TUBE DUE TO GRAVITY FLOW IN THE MANNER SAME AS OFF-REACTOR MODE).

[*] Scenario A[#] does not have cooling requirements hence not indicated here.

[#] For description of title of scenarios, refer to section 4.0.



NOTES :-

1. THE REFUELLING IS DONE IN THE DIRECTION OF FLOW. NORMALLY EIGHT FUEL BUNDLES ARE REFUELLED IN ONE REFUELLING OPERATION.

2. SHUTTLE TRANSFER STATION (STS) ON NORTH SIDE & NEW FUEL MAGAZINE ETC. AT SOUTH SIDE NOT SHOWN FOR CLARITY.

3. NORMALLY NEW FUEL BUNDLES FROM TRANSFER MAGAZINE (TM) ARE EXCHANGED WITH SPENT FUEL BUNDLES IN FM, AS SHOWN IN SOUTH SIDE, WHEN FM IS IN COMMUNICATION WITH TM. FOR CLARITY, NEW FUEL LOADING ROUTE TO FM IS SHOWN HERE ON NORTH SIDE ONLY.

4. FROM TM, AT A TIME, ONE PAIR OF SPENT FUEL BUNDLES IS DISCHARGED TO BAY VIA STS. TRANSFER OF THE PAIR FROM TM TO STS IS DONE IN DRY ENVIRONMENT AS SHOWN IN SOUTH SIDE AND DETAILED IN FIG. 2.

ACRONYMS :-

FM - FUELLING MACHINE
ICW - INNER CONTAINMENT WALL OF REACTOR BUILDING
NFM - NEW FUEL MAGAZINE
OCW - OUTER CONTAINMENT WALL OF REACTOR BUILDING
SFSB - SPENT FUEL STORAGE BAY
SFTT - SPENT FUEL TRANSPORT TUBE
STS - SHUTTLE TRANSFER STATION
TM - TRANSFER MAGAZINE

LEGEND :-

NEW FUEL
SPENT (IRRADIATED) FUEL
OPERATION IN D_2O ENVIRONMENT
OPERATION IN DRY ENVIRONMENT
OPERATION IN H_2O ENVIRONMENT

FIG. 1 SCHEMATIC OF MAJOR FUEL HANDLING OPERATIONS

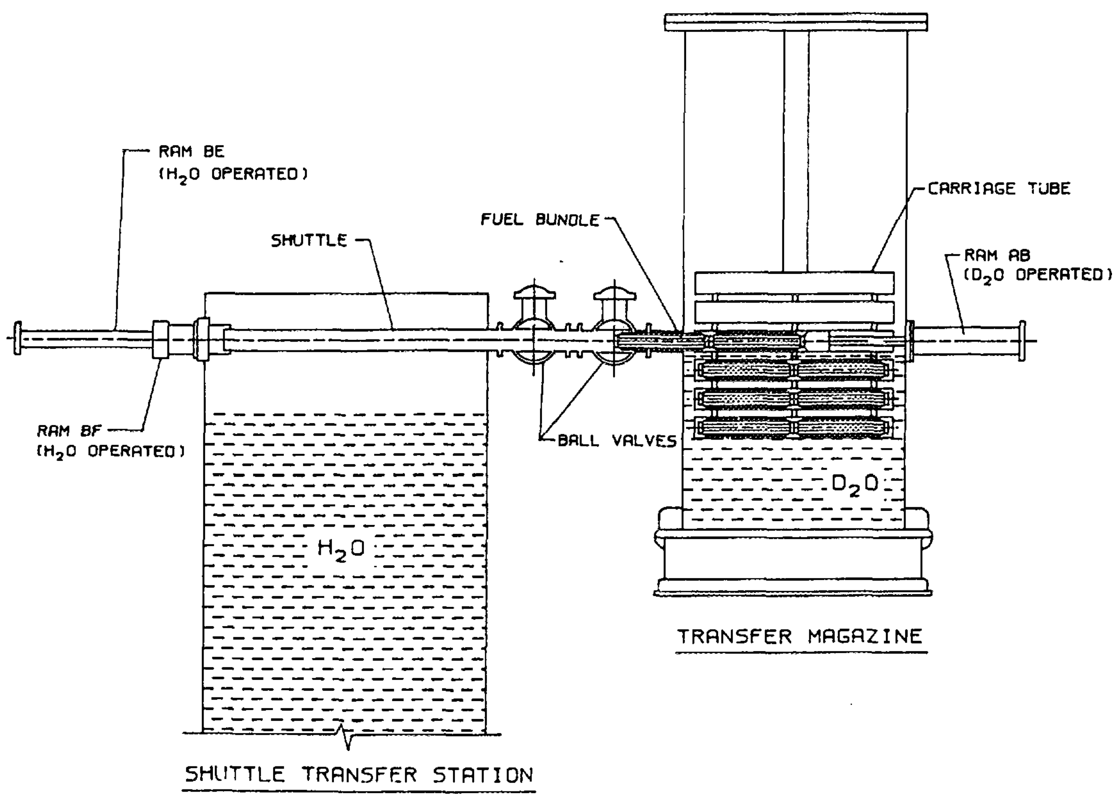


FIG. 2 DRY TRANSFER OPERATION

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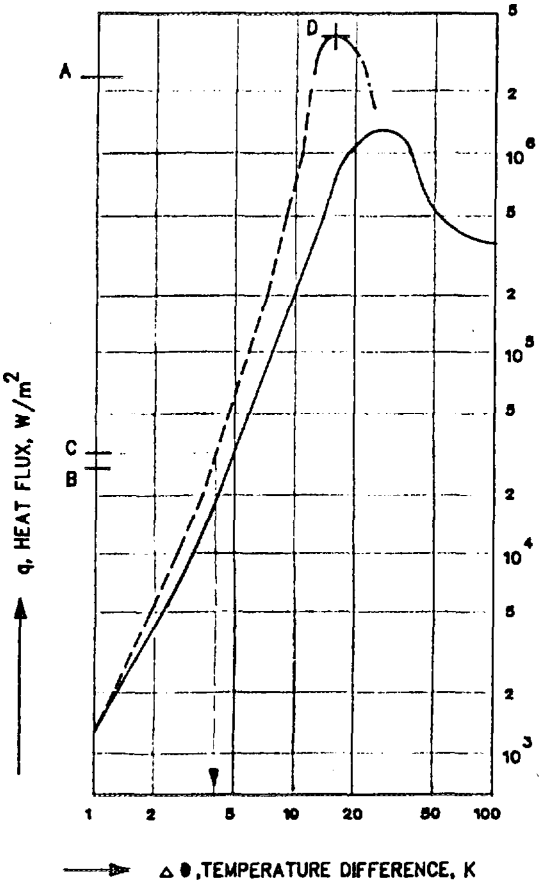


LEGEND :-

—— = STANDARD CURVE FOR WATER BOILING AT 373 K

---- = EXTRAPOLATED CURVE, FOR WATER BOILING AT 583 K, BY EVALUATING POINT D FROM FIG. 4 (EVALUATION IN FIG. 4 IS SHOWN IN CHAIN DOTTED LINE).

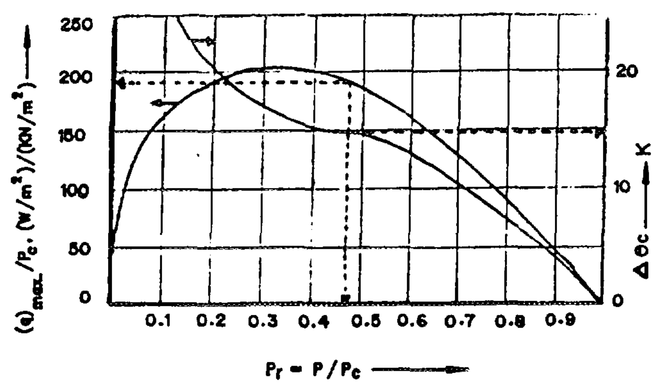
C = HEAT FLUX OF MAXIMUM POWER FUEL ELEMENT OF THE MAXIMUM POWER FUEL BUNDLE.

B = HEAT FLUX OF MAXIMUM POWER ELEMENT CORRESPONDING TO THE MAXIMUM POWER FUEL BUNDLE.

A = THE CRITICAL HEAT FLUX OF THE FUEL BUNDLE EVALUATED FROM EQUATION 10.112 OF [3].

FIG. 3 EFFECT OF TEMPERATURE DIFFERRENCE (BETWEEN THE SURFACE & POOL WATER) ON HEAT FLUX FROM THE SURFACE, KEPT SUBMERGED IN BOILING WATER.

(ref. FIG. 7.37 OF [4])



(P = OPERATING PRESSURE, P_c = CRITICAL PRESSURE FOR WATER)

FIG. 4 EFFECT OF PRESSURE ON 'PEAK HEAT FLUX IN THE BOILING CURVE FOR WATER i.e. (q) max' AND 'COR--RESPONDING TEMPERATURE DIFFERENCE, i.e. Δ θ c'

(ref. FIG. 7.40 OF [4] & FIG. 13.7 OF [6])

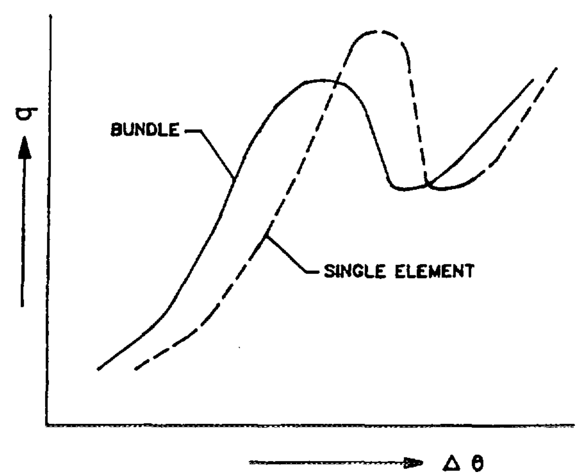


FIG. 5 CONCEPTUAL COMPARISION OF BOILING OUTSIDE BUNDLE WITH THAT OUTSIDE SINGLE ELEMENT

(ref. FIG. 5 IN SECTION 2.7.5 OF [6])

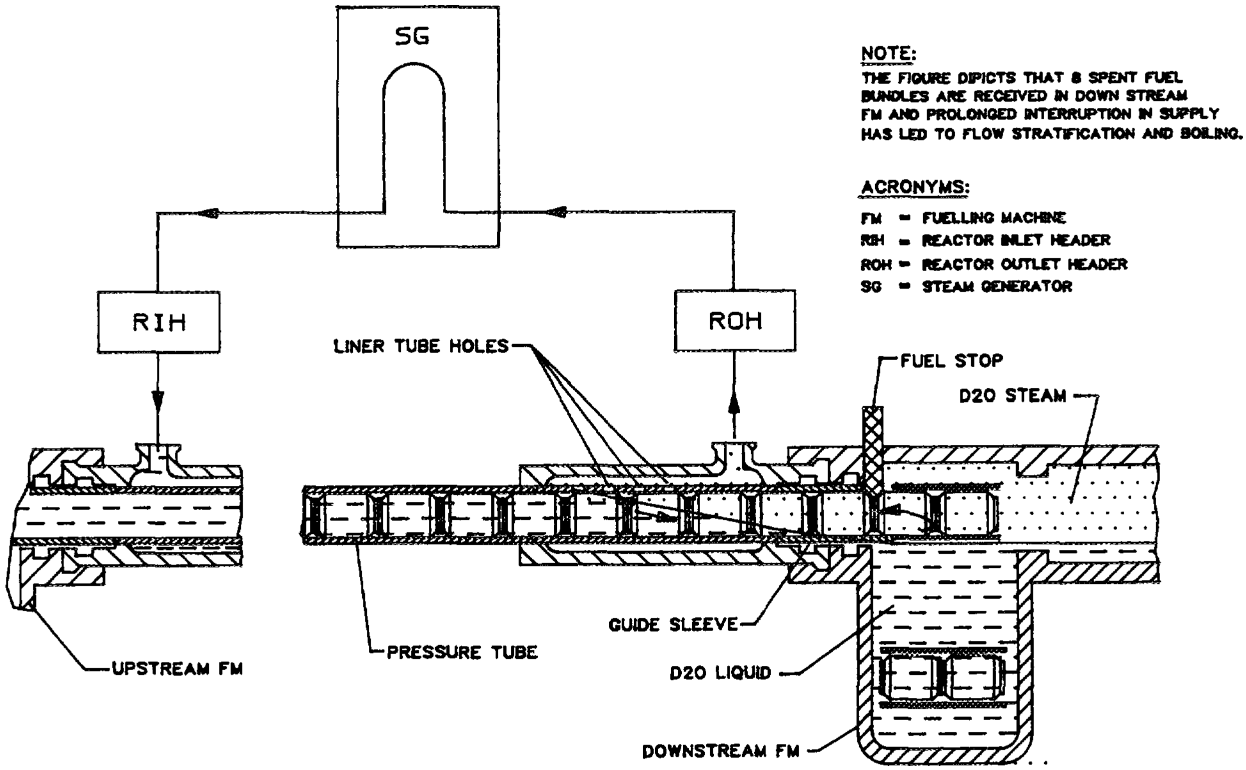


FIG. 6 FLOW STRATIFICATION IN FUELLING MACHINE

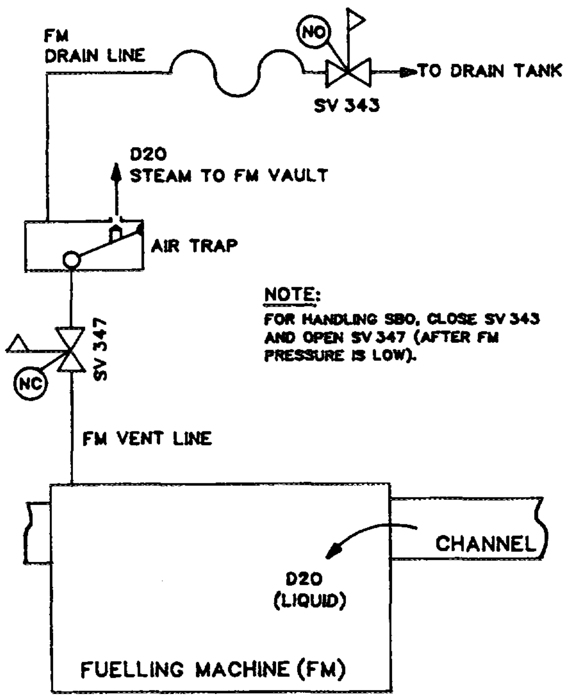


FIG. 7 VENTING OF FM BY AIR-TRAP

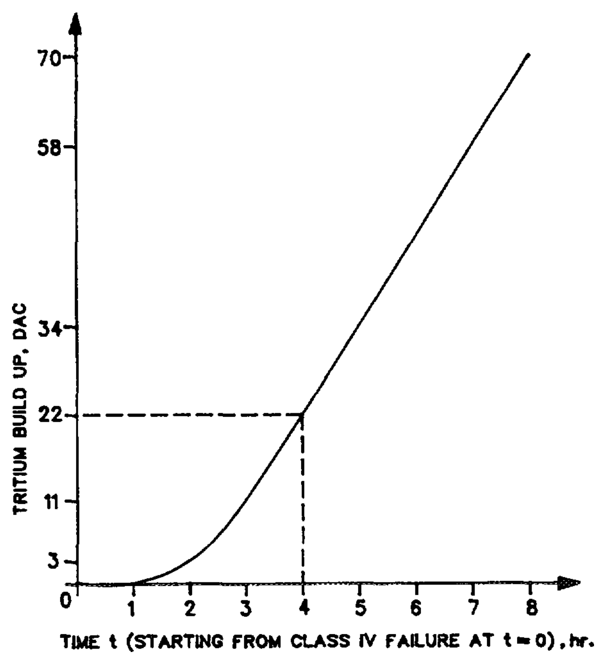


FIG. 8 TRITIUM BUILD UP IN V2 AREA IN 8 Hrs. OF SBO

(CORRESPONDING TO 1200 Kg. D20 RELEASED FROM FHS IN V1 AREA)

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EVOLUTION OF THE DESIGN OF FUEL HANDLING CONTROL SYSTEM IN 220 MWe INDIAN PHWRs

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ABSTRACT

Following two CANDU type reactors at Rajasthan (RAPS-1&2), three nuclear power stations, each of two units of 220 MWe has been in operation at Rajasthan (RAPS-1&2). Madras (MAPS-1&2). Narora (NAPS-1&2) and Kakrapar (KAPS-1&2). Two more stations, also of 220 MWe capacity, are under construction at Rajasthan (RAPP-3&4) and Kaiga (Kaiga-1&2). These are natural uranium fuelled pressurized heavy water cooled and heavy water moderated reactors (PHWRs). The two units at Rajasthan viz RAPS-1&2, were built with the technical collaboration with Canada and the rest of the units have been designed and built indigenously incorporating a number of modifications, particularly in the on-power refuelling system. The evolution of the design of the Fuel Handling Control systems of these reactors, taking into consideration operational needs, safety aspects and maintainability are highlighted in this paper.

A combination of hydraulic and electronic control has been provided to enable the operations. In RAPS-1&2, hardwired electronic controls were provided, while in MAPS-1&2, the hardwired system was improved. From NAPS onwards, a computerized control system with hardwired interlock logic has been provided. New devices like coarse-fine potentiometers, special oil filled potentiometer assembly, rectilinear potentiometers etc., were specified from NAPS onwards. Positioning logic is computerized providing flexibility and expandability. Digital panel meters and indicating lamps have been provided for manual mode operations, while CRT monitors help in computer mode operations.

Hydraulic controls which comprise D_2O hydraulics, H_2O hydraulics and oil hydraulics have been improved from NAPS onwards. Hydraulic panels have been relocated in accessible areas to reduce manrem and for better maintainability. All electric drives including X and Y drives were modified as hydraulic drives for better control. New types of valves have been employed.

The paper highlights the details of the changes incorporated.

1. INTRODUCTION

The pressurised Heavy Water reactors (PHWRs) with natural uranium fuel require regular and frequent on-power refuelling. In Indian PHWRs of 220 MWe, a complex Fuel Handling System mainly comprising two Fuelling Machines and a Fuel Transfer System is provided to enable refuelling. An elaborate control system with the following main features has became necessary for enabling automatic and remote manual operations.

a) Automatic Sequential Logic
b) Safety interlock logic
c) Continuous feedback process control
d) Accurate positioning at precalibrated positions.
e) Torque, Force and speed control for the drives.

f) Manual operability and system status indicators.

It has also been considered desirable to have an operational logging system in addition to the above.

These requirements have been met by providing suitable Electrical and Hydraulic controls. The Hydraulic controls meet the requirement (e) mentioned above whereas the rest of the requirements are met by Electrical controls. The two CANDU type reactors at Rajasthan Atomic Power Station (RAPS-1&2)and the two reactors at Madras Atomic Power Station (MAPS-1&2) were provided with hardwired Electrical control systems, whereas in subsequent reactors at Narora (NAPS-1&2) and Kakrapar (KAPS -1&2) computerised control systems have been provided. The above reactors are in operation. Four units, two at Kaiga (Kaiga- 1&2) and two at Rajasthan (RAPP-3&4) are under construction.

2. ELECTRICAL CONTROLS

The Fuel Handling System is provided with extensive electrical controls to enable automatic sequential operation while ensuring safety , reliability and maintainability.

2.1 CONTROL SCHEMES

In RAPS-1&2 and MAPS-1&2, a totally hardwired system was provided. Discrete component logic units, employing Germanium transistors were employed in RAPS-1&2, whereas these were improved using silicon transistors in MAPS-1&2. The positioning controls employed hardwired trip units.

From NAPS onwards, a computerised system was adopted in view of the following advantages:

I. Better man-machine communication with the help of CRT messages.
II. Flexibility in positioning, interlock and operational program logic provided by software, compared with wiring changes that were needed in previous projects.
III. On-line diagnostic capability.

IV. Possibility of performance evaluation by suitable data logging and analysis.

The computerised system provided in NAPS-1&2 and KAPS-1&2, employed a 16 bit supervisory master computer with two separate 8-bit microcomputers. The master computer analyses the operational demands, as entered by the operator and initiates the various tasks that the microcomputers have to execute. The software is organised in such a way that a full auto mode operation, including automatic sequential operation, is possible with both master and microcomputers being functional, whereas, a degraded or semi-auto mode is possible with only the microcomputers being functional.

A manual mode is also provided, alongwith a full fledged control console, so as to enable completion of operations to bring the system to a safe state, in the event of any failures of the computerised system.

The application software for the computerised system is written in a specially developed higher level language called Process Control Language (PCL). A compiler has been provided with the master computer, so that the codes, which are compiled off-line, can be stored on the master computer disks, and transferred to microcomputers as required. The transferred codes are interpreted by an Interpreter program in the micro computer. It is possible to calibrate the various discrete positions, by bringing the drive to the required position and invoking the calibration program. The microcomputer enables automatic positioning and also checks the safety interlocks. No command is issued to the field device unless the interlocks are satisfied, as checked by the microcomputer and also by a hardwired safety interlock logic, which does not depend on the computerised system. This hardwired IC-based safety interlock logic also ensures safety for the commands issued by the operator in the manual mode. During the sequential operation, if any of the permissives are not available, these are indicated on the CRT monitors on the console by proper messages.

On the control console, digital panel meters have been provided to enable positioning of the drives, in cases, where microcomputers develop faults during operation. The operator will be required to refer to the calibration record in such cases.

The safety interlock logic in NAPS-1&2 employs TTL integrated circuits, as compared with discrete components like silicon transisters in MAPS-1&2. CMOS logic units are employed in RAPP-3&4 and Kaiga-1&2. Comparator units are employed to generate logic signals based on analog field signals. These logic signals are used in interlock logic.

While in RAPP-3&4/Kaiga-1&2, the control philosophy is mostly comparable to that of NAPS, a few improvements have been incorporated in design. Arithmetic capability to some extent has been added in the PCL. A 16-bit microcomputer will be employed, thereby enhancing the computing power considerably. This will enable limited use of automatic sequential operations even without the supervisory computers. Positioning can be done based on the signals from either the primary devices or back-up devices, comparator units will have the set point potentiometers located independently, so that replacement of the defective comparater units does not require adjustment of set points. A PC-based operational logging sub system will also produce a summary print out apart from storing a detailed log in the computer.

2.2 SENSORS

In comparison with RAPS/MAPS, a coarse-fine potentiometer assembly has been employed as a position monitoring sensor for many drives (e.g. B-ram of Fuelling Machines)from NAPS onwards. This has a ten turn potentiometer with a single turn continuous rotation potentiometer on a common shaft, thereby enhancing the resolution and positioning accuracy with a good signal to noise ratio. At certain places where two Linear Variable differential transformers (LVDT) were employed in RAPS/MAPS, a single long stroke LVDT has been employed from NAPS onwards. For sensing small movements, rectilinear potentiometers have also been used for better compatibility with the computerised controls. Oil filled potentiometer housing for C-Ram has been modified to utilise a single turn continuous rotation potentiometer, thereby reducing probability of potentiometer failures resulting from Ram'C rope breakages. Further this type of design has simplified the procedure to replace the failed assemblies.

3. HYDRAULIC CONTROLS

Hydraulic controls have been provided to ensure that the hydraulic cylinders and hydraulic drive motors function to meet the requirements of the design of mechanical equipment. Hydraulic controls can be categorised under Water Hydraulic controls and Oil hydraulic controls.

3.1 WATER HYDRAULIC CONTROLS

Water Hydraulic controls employ Heavy water or light water as the fluid medium, so as to ensure compatibility with the process. The water hydraulic cylinders require the force and speed to be controlled. Solenoid operated directional valves are employed to enable the advancing and retracting of the cylinders. In general, Differential Pressure Regulating Valves with the outlet pressure referenced to the operating pressure of the associated equipment, are employed to provide controlled forces. However, control valves are employed for C-Ram force control, so that different forces can be selected on program commands. Speed control is achieved by providing throttling valves or fixed orifices.

Designs of most of the stations as regards water hydraulic controls are similar. However from NAPS onwards the layout of the valve panels has been changed. Control devices for new actuators have been accommodated and also catenary hoses have been standardized. Fuelling Machine (F/M) valve panels have been rearranged such that the structural frame work meets seismic requirements. All catenary hoses have been provided with either in-line excess flow check valves or flow limiting orifices, to prevent or limit the D_2O escape during failures of hoses.

3.2 OIL HYDRAULIC CONTROLS

These controls enable speed and torque control, in addition to direction of movement for the various oil hydraulic drives. Many of the Electric drives, provided in the design of RAPS-1&2 and MAPS-1&2 were changed to oil hydraulic drives from NAPS onwards. For example, the drive motors of X and Y motion drives of Fuelling Machines have now been made oil hydraulic from NAPS, whereas these were electric motors in RAPS-1&2 and MAPS-1&2. Similarly New Fuel Magazine and Transfer Magazine drives of Fuel Transfer System are now oil hydraulic. The design philosophy of speed control by using Pressure Compensated Flow Control Valves and force/torque control by pressure regulating valves has been extended to the new drives. In certain cases relief valves have been provided for force control. Also additional devices like counterbalance valves (CBV) have been provided. For example, for F/M Y-motion drive control, CBVs are provided to prevent uncontrolled descent whenever brakes are released.
In RAPS-1&2 and MAPS-1&2, F/M oil power pack units and valve panels were provided on the

Fuelling Machines. However from NAPS onwards, these have been shifted to accessible areas, by employing increased number of catenary hoses. This has resulted in better maintainability and reduction in manrem. Oil coolers with chilled water flow have been provided to limit the temperature rise of oil, thereby improving the performance.

4. SAFETY ASPECTS

The safety of Fuel Handling System will be ensured mainly if the accidental unclamping is prevented and coolant to the spent fuel bundles is made available at all the times. However, additionally, it is considered necessary to prevent hot water ingress into the Fuelling machines and to provide sufficient interlocks for preventing mechanical damages resulting in difficulties for retrieval apart from leading to manrem consumption. Control system design has been improved in these respects. Interlocks are checked both in the hardware and software from NAPS onwards. Autoinitiated action based on redundant timers have been incorporated so as to ensure coolant during dry transfer. In RAPP-3&4/Kaiga-1&2 devices which are important for safety has been identified and it has been ensured that fail-safe feature is provided to a large extent. For safety related functions, back-up devices are provided. The cable routes have been made diverse for back-up devices to prevent common cause failures to the extent practically feasible. In the computerised systems, watch dog timers are provided. Any power supply failure leads to a "Tripped state" which has been made safe. In this state, all bistable valves retain the last energised condition, while most of the oil hydraulic drives come to a halt. The failure of set point power supply for F/M magazine pressure control leads to closure of return line control valves, preventing hot water ingress, if the Fuelling Machine is in hydraulic communication with the coolant channel.

5. CONCLUSIONS

As can be seen from above, NAPS design which is computerised has also been improved with regard to operability, man-machine interface and state-of-art technology. Safety has been of prime concern in addition to operating features. It may be noted that the performance of the system at NAPS/KAPS has been reasonably good.In RAPP-3,4/Kaiga-1&2, in addition to the above mentioned features, maintainability has also been given importance and it is expected that RAPP-3&4/Kaiga-1&2 will give better performance.